2.14 Self-Luminous Products

2.14.1 Introduction

In 10 CFR 30.19, any person is exempt from licensing requirements to the extent that person receives, possesses, uses, transfers, owns, or acquires tritium (³H), ⁸⁵Kr, or ¹⁴⁷Pm in self-luminous products; however, the exemption does not apply to products intended primarily for frivolous purposes or in toys and adornments. This exemption was proposed on June 21, 1968 (33 FR 9198), and issued as a final rule on June 6, 1969 (34 FR 9025).

Requirements for a license to manufacture, process, produce, or initially transfer self-luminous products containing ³H, ⁸⁵Kr, or ¹⁴⁷Pm and intended for use under 10 CFR 30.19 are contained in 10 CFR 32.22. Conditions of licenses are also included in 10 CFR 32.25; these include requirements for quality control, labeling, and reporting of transfers.

Section 32.22 specifies that the license applicant must demonstrate that the product will meet certain safety criteria set forth in 10 CFR 32.23. These safety criteria are the primary factors in controlling the radiation doses associated with this exemption and are described below:

- In normal use and disposal of a single exempt unit, it is unlikely that the annual dose from external exposure, or the annual dose commitment resulting from intake of radioactive material, to a suitable sample of the group of individuals expected to be the most highly exposed to radiation or radioactive material from the product will exceed 0.01 millisievert (mSv) (1 mrem) to the whole body, the head and trunk, active blood-forming organs, the gonads, or the lens of the eye: 0.15 mSv (15 mrem) to the hands and forearms, feet and ankles, or localized areas of the skin averaged over areas no larger than 1 cm²; and 0.03 mSv (3 mrem) to any other organs.
- In normal handling and storage of the quantities of exempt units that are likely to accumulate in one location during marketing, distribution, installation, and servicing of the product, it is unlikely that the annual dose from external exposure, or the annual dose commitment resulting from intake of radioactive material, to a suitable sample of the group of individuals expected to be the most highly exposed to radiation or radioactive material from the product will exceed 0.1 mSv (10 mrem) to the whole body, the head and trunk, active blood-forming organs, the gonads, or the lens of the eye; 1.5 mSv (150 mrem) to the hands and forearms, feet and ankles, or localized areas of the skin averaged over areas no larger than 1 cm²; and 0.3 mSv (30 mrem) to any other organs.
- It is unlikely there will be a significant reduction in the effectiveness of the containment, shielding, or other safety features of the product from wear and abuse that is likely to occur in normal handling and use of the product during its useful life.
- In use and disposal of a single exempt unit, or in handling and storage of the quantities of exempt units that are likely to accumulate in one location during marketing, distribution, installation, and servicing of the product, the probability is low that the containment, shielding, or other safety features of the product would fail under such circumstances that an individual would receive an external dose or internal dose

commitment in excess of 5 mSv (0.5 rem) to the whole body, the head and trunk, active blood-forming organs, the gonads, or the lens of the eye; 75 mSv (7.5 rem) to the hands and forearms, feet and ankles, or localized areas of the skin averaged over areas no larger than 1 cm², and 15 mSv (1.5 rem) to any other organs. The probability is negligible that an individual would receive an external dose or internal dose commitment in excess of 150 mSv (15 rem) to the whole body, the head and trunk, active blood-forming organs, the gonads, or the lens of the eye; 2 Sv (200 rem) to the hands and forearms, feet and ankles, or localized areas of the skin averaged over areas no larger than 1 cm²; and 0.5 Sv (50 rem) to any other organs.

These criteria for failure of safety features have the effects of establishing an upper limit on the amount of radioactive material in an exempt unit and of establishing a need to provide some degree of containment for the material even under accident conditions. Furthermore, the probabilities associated with radiation dose limits for accidental exposures ensure that the risk to individuals would be very small.

In the last set of safety criteria for accidental exposures, it is intended that as the magnitude of the potential dose increases above that permitted under normal conditions, the probability that any individual will receive such a dose must decrease. The probabilities of failure of safety features in products are expressed in general terms to emphasize the approximate nature of the required estimates, but the following values may be used as guides in estimating compliance with the safety criteria:

- A probability of failure is "low" if there is not more than one failure per year for each 10,000 exempt units distributed.
- A probability of failure is "negligible" if there is not more than one failure per year for each 1 million exempt units distributed.

Therefore, the safety criteria for self-luminous products containing ³H, ⁸⁵Kr, or ¹⁴⁷Pm used under 10 CFR 30.19 provide: (1) radiation dose limits for individuals from normal handling, storage, use, and disposal of these products, and (2) radiation dose limits for individuals in conjunction with associated probabilities of occurrence for accidents. The safety criteria do not include any limits on collective dose.

2.14.2 Description of Items

A variety of self-luminous products that have existed at one time or another could potentially cause exposure to consumers (NCRP 95, NUREG/CR–1775). These products include wristwatches, telephone dials and push buttons, night map and document readers, leveling bubbles, marine compasses, gun sights, and aircraft and building exit signs. Some of these products were distributed for use under general license and some were distributed for use under exemptions other than 10 CFR 30.19. The only items to be distributed for use in significant numbers under this exemption have been wristwatches and gun sights containing sealed tubes of ³H gas.

2.14.2.1 Wristwatches

Electronic digital watches started out as a gimmick consumer item in the early 1970s (NUREG/CP-0001; Ristagno). Two competing technologies were the light-emitting diode (LED), which offered a time readout on demand, and the liquid crystal display (LCD), which offered a continuous time readout. The reason for the LED time-on-demand readout was the high drain on the batteries used to power the watch. Because of the short battery life expectancy of the LED, the LCD became the dominant technology. The battery life expectancy was anywhere from 3 to 9 months for the LED and 18 to 24 months for the LCD. The LCD was visible under normal ambient lighting conditions, but it became marginally visible at twilight and invisible at night. Thus, backlighting with either miniature incandescent light bulbs or self-luminous gaseous tritium light sources (GTLSs) was used to make the LCD readable at night.

The self-luminous LCD wristwatches contained about 7.4 gigabecguerel (GBg) (200 millicurie (mCi)) ³H with a nominal composition, according to watch manufacturers, of elemental ³H ≥99% and tritium oxide ≤1% in two sealed pyrex (borosilicate) glass tubes (NUREG/CR-0215). A typical tube was approximately 2.0 cm long, with a 0.3-cm by 0.08-cm elliptical cross-section and a 0.02-cm-thick wall. The tubes were coated on the inside with an inorganic phosphor. Light was produced when the phosphor was activated by beta particles emitted during the decay of the ³H in the sealed tube. The sealed tubes were well protected in an assembled watch and were not easily accessible. Both tubes were bonded to a metal tray with a shockabsorbent silicone adhesive. The tray was sealed to a transflector-display panel, and the entire assembly was encapsulated in a plastic collar. A printed circuit board was affixed to the plastic collar, covering the bottom of the tray, and the entire module was then inserted into the watch case as an integral unit. About 40,000 LCD watches backlighted by GTLSs were distributed during 1976 (NUREG/CP-0001; Ristagno), and a total of several million were distributed during the late 1970s and early 1980s (NUREG/CR-1775). Data reported by industry sources have indicated that LCD watches backlighted by GTLSs were produced for only a few years during the early 1980s (NCRP 95).

The self-luminous wristwatches now being distributed in the United States contain 14 to 15 tiny GTLSs to light the dial and are imported mainly from Canada and Switzerland. In a typical Swiss watch design (Yeaple, 1989), 14 GTLSs containing a total of about 1.9 GBq (50 mCi) of ³H gas (HT) are used to provide good readability in any light. They are made of pyrex glass and sealed with a laser process. Twelve of the GTLSs are 0.25 cm long with outer diameters of 0.095 cm, and the other two are 0.41 cm and 0.66 cm long with outer diameters of 0.07 cm. The two longer GTLSs are bonded to the minute and hour hands so they do not hard set using elastomeric adhesives applied in a double layer, and the shorter GTLSs are placed at the 12-hour indexes on the dial. They are held in place by clamping pressure created when a plastic holder ring having 12 U-shaped channels for the GTLSs is press fitted into the back of the dial. The ring is then adhesively bonded to the dial. The wristwatch is shockproof and waterproof to a depth of 30 meters. In a typical Canadian watch design, 15 tiny GTLSs containing a total of about 1.1 GBq (30 mCi) of ³H are used, with the extra GTLS being located on the bezel of the watch. GTLSs containing ¹⁴⁷Pm do not appear to be in current use.

2.14.2.2 Gun Sights

Self-luminous night sights are relatively new for firearms, having been introduced in the mid-1980s (Kasler, 1992). On handguns, the total amount of ³H contained in the three tiny

GTLSs of a typical three-dot set is approximately 1.9 GBq (50 mCi). The GTLSs glow with a distinctive green color that quickly draws the eye to the proper sight alignment in low-light situations. Night sights are currently standard issue with several Federal law enforcement agencies, many State police departments, and hundreds of municipal and county departments. Statistically, about 80% of law enforcement shooting encounters take place in low-light or no-light situations (Petty, 1992). Homeowners may also confront intruders during evening hours, and anyone who may have to fire in low light can benefit from the use of night sights.

The GTLSs for some night sights are manufactured in the United States, but most are imported from either Switzerland or Israel. In a typical Swiss sight assembly for a handgun, the GTLSs consist of three pyrex tubes about 0.5 cm in length, 0.1 cm in outside diameter, and 0.02 cm in wall thickness. Each tube is coated with an inorganic phosphor on the inside, filled with HT, sealed with a laser, and leak tested. The next step in the assembly process is to shock mount the GTLSs inside small aluminum tubes using silicone rubber cement. This provides shock isolation to protect the GTLSs from weapon recoil. The viewing end of the aluminum tubes are then sealed with polished industrial sapphires. Clarity of the sapphire provides maximum brightness and its hardness (second only to a diamond) also provides protection for the GTLSs. Finally, the aluminum tubes are precision turned on a lathe to provide a controlled fit into holes drilled in the steel sight assembly of the weapon, and they are sealed in place using a modified cyanoacrylate adhesive (super glue). In a typical Israeli sight assembly for a handgun, the GTLSs are entirely encased in a synthetic bonding material that completely seals the GTLSs and makes them impervious to gun-cleaning solvents. Neither the Israeli nor the Swiss sights are damaged by wiping and cleaning with conventional gun-cleaning solvents (Petty, 1992).

Self-luminous sights are also available as "bow pin" sights for archery bows and as night scopes for rifles and other large firearms. A bow pin sight typically uses a single GTLS containing 0.19 to 0.37 GBq (5 to 10 mCi) of ³H, and a night scope for a large firearm typically uses one to three GTLSs containing 3.0 to 3.7 GBq (80 to 100 mCi) of ³H per GTLS. The distribution of bow pin sights and scopes for large firearms is small when compared to the number of gun sights being distributed for handguns.

2.14.3 Summary of Previous Assessments

....

Assessments of radiological impacts on the public from self-luminous wristwatches containing HT were performed by McDowell-Boyer and O'Donnell (NUREG/CR–0215) and Buckley et al. (NUREG/CR–1775) and the results were summarized by the National Council on Radiation Protection and Measurements (NCRP) (NCRP 95). The analysis of Buckley et al., was based in large part on the work of McDowell-Boyer and O'Donnell.

McDowell-Boyer and O'Donnell calculated doses from the escape of HT into air using either CONDOS (Computer Codes, O'Donnell et al., 1975) or AIRDOS-II (Computer Codes, Moore, 1977). For calculational purposes individuals were divided into three groups: (1) wearers of the wristwatches, (2) bystanders, and (3) distant persons. The CONDOS code was used in the individual dose calculations for wearers of the watches and bystanders. Bystanders were individuals who work near or associate with wearers of the watches. Three such individuals were assumed to be near each wearer. For distant persons, the individual and collective doses were calculated using the AIRDOS-II code. The ³H leakage was considered to be a ground-level release that was dispersed into the atmosphere, then absorbed through the skin

and inhaled by the distant persons within an 80-km radius of the watch wearer of the watch. An annual distribution of 1 million watches and a 10-year effective lifetime for the watches were used in the estimation of collective doses.

The exposure scenarios for routine use of self-luminous wristwatches that were considered include:

- exposures to workers and members of the public during distribution and transport;
- exposures to wearers of watches, bystanders near wearers, and other members of the public;
- exposures to watch repairmen; and

....

• exposures to members of the public from storage of discarded watches in the home and disposal in landfills or by incineration.

The exposure scenarios considered involving accidents or misuse of self-luminous wristwatches include:

- exposure following breakage of a watch in the home;
- exposure to a watch repairman following breakage in a repair shop; and
- exposure following a fire in a warehouse.

In the dose analyses for the routine use scenarios, a leakage rate of ³H from wristwatches containing up to 7.4 GBq (0.2 Ci) of HT was assumed, on the basis of available data, to be 1.9 KBq/day (50 nCi/day) (i.e., 10 ppb/h), and all ³H was assumed to be in the oxide form (i.e., tritiated water vapor (HTO)). The latter assumption probably resulted in overestimates of dose, because some of the ³H would be in elemental form (i.e., HT) and the dose per unit activity is much less for HT than for HTO.

In the dose analyses for storage in homes, the decay of ³H over the previous 10 years was considered and the leakage rate was reduced to 1.1 KBq/day (30 nCi/day) (i.e., 10 ppb/h). All ³H was also assumed to be converted to HTO.

In the dose assessments for accident and misuse scenarios, instantaneous release of 7.4 GBq (0.2 Ci) of ³H from watches was considered, and the ³H was assumed to be 99% in HT form and 1% in HTO form. Doses resulting from accidents and misuse were also estimated by assuming that all of the ³H was HTO, but these results will not be discussed here because most of the ³H from an instantaneous release would be in the HT form.

The dose estimates obtained by McDowell-Boyer and O'Donnell (NUREG/CR-0215) for the routine use scenarios are summarized as follows:

 During distribution, truck drivers who pick up shipments could receive an annual dose equivalent of 0.003 mSv (0.3 mrem), but other parcel system workers likely would receive annual dose equivalents less than 1×10⁻⁴ mSv (<0.01 mrem). Some marketing employees could receive an annual dose equivalent of 0.0013 mSv (0.13 mrem), but most workers likely would receive annual dose equivalents of less than 5×10^{-5} mSv (<0.05 mrem). Annual dose equivalents to individual customers probably would be less than 1×10^{-7} mSv (< 1×10^{-5} mrem). The collective dose equivalents from the annual distribution of 1 million wristwatches was estimated to be about 0.1 person-Sv (10 person-rem).

- During routine use, wristwatch wearers could receive annual dose equivalents in the range of 3×10^{-5} to 2×10^{-4} mSv (0.003 to 0.02 mrem), depending on the location of the wearer. The same doses could be received by bystanders who work near, live with, or associate with the watch wearers. Other individual members of the public who are distant from wearers could receive annual dose equivalents of 3×10^{-8} mSv (3×10^{-6} mrem) or less. Annual collective dose equivalents from use of the wristwatches were estimated to total 2.3 person-Sv (230 person-rem) (i.e., about 0.65 person-Sv (65 person-rem) to wearers, 1.6 person-Sv (160 person-rem) to bystanders close to the wearer, and 0.01 person-Sv (1 person-rem) to distant persons).
- During repair, watch repairmen could receive annual dose equivalents in the range of 3×10⁻⁴ to 6×10⁻⁴ mSv (0.03 to 0.06 mrem), depending on the size of the repair shop, and the annual collective dose equivalent could be about 0.01 person-Sv (1 person-rem).
- During storage of discarded wristwatches in the home, individuals in the home could receive annual dose equivalents as high as 3×10⁻⁵ mSv (0.003 mrem), and distant persons could receive annual dose equivalents 2×10⁻⁸ mSv (2×10⁻⁶ mrem) or less. The annual collective dose equivalent from storage could be about 0.5 person-Sv (50 person-rem). If half of the watches are disposed of in landfills after storage and the other half are burned, maximally exposed individuals could receive annual dose equivalents of 0.001 mSv (0.1 mrem) near a landfill and 0.2 mSv (20 mrem) near a location where watches were burned at ground level, no longer a common practice. Burning in an incinerator would not produce individual doses higher than those from burning at ground level. The annual collective dose equivalents could be about 0.01 person-Sv (1 person-rem) from disposal in landfills and about 2 person-Sv (200 person-rem) from incineration and burning at ground level.
- The total annual collective dose equivalent from all normal use scenarios thus would be 5 person-Sv (500 person-rem). About half of the collective dose would be received during use of wristwatches, and most of the rest results from incineration or burning at ground level.

The total collective dose estimated by McDowell-Boyer and O'Donnell is about a factor of 15 less than the value of 77 person-Sv (7700 person-rem) adopted in NCRP 95 on the basis of results given by Buckley et al. (NUREG/CR–1775). This difference results almost entirely from the difference in estimated collective dose from disposal of wristwatches in landfills or by incineration. For these scenarios, Buckley et al. used more conservative assumptions in the dose analysis than those used by McDowell-Boyer and O'Donnell.

The dose estimates obtained by McDowell-Boyer and O'Donnell for the accident and misuse scenarios are summarized as follows:

- For breakage of a single wristwatch in a home, the dose equivalent to an individual over the next 24 hours would be 0.005 mSv (0.5 mrem).
- For breakage of a single watch in a small repair shop, the dose equivalent to a repairman over the next 10 hours would be 0.1 mSv (10 mrem).
- For a fire in a warehouse containing 60 wristwatches, and assuming no ventilation of air for 15 minutes, the dose equivalent to a worker in the warehouse during that time would be 0.01 mSv (1 mrem).

Thus, it is concluded from the published analyses that the individual and collective effective dose equivalents (EDEs) to distant persons are extremely small and can be ignored in the case of routine use and storage of old watches in homes. Also, it is concluded that (1) the individual and collective doses for other exposure pathways should be recalculated for the current annual distribution and ³H activity in self-luminous wristwatches, (2) exposures from accidents, disposal, and distribution of the watches should be reevaluated using the generic methodologies in Appendixes A.1, A.2, and A.3 of this report, and (3) the ³H intake through skin in contact with the watch case should be considered in estimating individual and collective doses to wearers during routine usage. The ³H intake through skin in contact with a watch case has been discussed in several recent articles by Brunner et al. (1996), Turvey (1996), Thüler (1996), and Beyer et al. (1996).

For gun sights containing HT, there are no known previous analyses of the radiological impacts, and a totally new assessment is needed for this self-luminous product.

2.14.4 Current Assessment for Wristwatches

Table 2.14.1 presents results of the current assessment of potential radiation doses for an assumed annual distribution of 100,000 wristwatches initially containing 1.9 GBq (50 mCi) of ³H each. The effective lifetime of the watches is assumed to be 10 years, and the leakage rate of ³H from the GTLSs in the watches is assumed to be 10 ppb/h under normal circumstances (NUREG/CR–0215). For ³H, the only important modes of exposure are inhalation and absorption though the skin. However, the chemical form of the ³H at the time of exposure is very important with respect to the dose calculations. Doses from exposure to HT are estimated to be about 10,000 times less than those from exposure to the same concentration of HTO in air (see, for example, ICRP 68, Table C.1).

The ³H in the GTLSs is generally more than 99% HT and less than 1% HTO. Thus, the ³H that leaks from the GTLSs will mainly be in the HT form, but it will contact many surfaces that catalyze oxidation before leaking into air (NUREG/CR–0215). For this reason, a conservative approach is taken. For exposure to airborne levels, it is assumed that ³H leaks from the GTLSs as HTO under normal circumstances. This assumption may overestimate dose particulary for persons in enclosed spaces (i.e., houses), but it should have less of an effect on exposure from distant releases in the outdoors (i.e., from disposal). For skin uptake and exposure from direct contact with surfaces containing GTLSs, special dosimetry has been developed (see Section 2.14.4.2.1). Since accidents and misuse may involve an instantaneous release of the ³H in the watches, no assumptions are made concerning elemental ³H conversion since contact

with other surfaces that catalyze oxidation would be less likely and room ventilation would rapidly vent the HT (NUREG/CR-0215).

2.14.4.1 Distribution and Transport

....

The potential radiation doses from distribution and transport of the self-luminous watches are estimated using the generic methodology of Appendix A.3. In applying this methodology, it is assumed that local parcel-delivery drivers in large trucks pick up the watches from suppliers and take them to local terminals, where they are shipped by semi-truck to other local terminals for delivery to individual buyers or stores. Also, it is assumed that each shipment passes through an average of four regional terminals before reaching its final destination.

It is assumed further that (1) the radiation dose to workers at both local terminals and regional terminals are the same as those estimated for workers in a large warehouse, (2) a local parceldelivery driver could pick up an average of 200 watches per day (250 day/yr) from a single supplier, (3) 50% of the watches are shipped directly to individual buyers and 50% are shipped directly to stores, (4) retail clerks in some large department stores could be exposed continuously to an average of 10 watches during the year, and (5) the leakage rate from the watches is 10 ppb/h or 100 times less than the value of 1 ppm/h used in the development of the generic methodology in Appendix A.3. The direct shipment of small, expensive items, such as watches to retail stores, is commonly used to minimize possibility of theft during distribution.

Based on these assumptions and generic methodology in Appendix A.3, the individual EDE could be 0.001 mSv (0.1 mrem) for a local parcel-delivery driver who is assumed to pick up 1000 self-luminous watches from the same supplier each week (50 wk/yr). Individual doses to other truck drivers, terminal workers, retail clerks, store customers, and members of the public along truck routes would be less. The total collective EDE from 1 year's distribution of 100,000 self-luminous watches containing 1.9 GBq (50 mCi) of ³H per watch could be 1×10^{-2} person-Sv (1 person-rem), due almost entirely to exposure to store clerks and shoppers.

2.14.4.2 Routine Use

Two modes of exposure can occur during routine use of self-luminous wristwatches containing ³H. One mode is exposure to airborne releases of ³H from the wristwatches, and the other mode is exposure from skin contact with the case of the wristwatch. The latter mode of exposure applies only to the wristwatch wearer, whereas the first mode of exposure applies to the wristwatch wearer as well as coworkers and other family members.

The following parts of this section discuss individual doses to wearers from skin contact with the wristwatches (Section 2.14.4.2.1); individual doses to wearers, coworkers, and other family members due to airborne releases during routine use (Section 2.14.4.2.2); individual doses due to storage of old watches in homes (Section 2.14.4.2.3); and total individual and collective doses from all types of exposures during routine use and storage of watches in homes (Section 2.14.4.2.4).

2.14.4.2.1 Skin Contact With the Wristwatch Case

The importance of this pathway of exposure is noted from a publication by Brunner et. al (1996). In the study by Brunner et al., release rates from 82 different plastic case watches were

determined. The observed release rates ranged from 110 to 160,000 Bq/day (3 to 4300 nCi/day) with a median of 16,000 Bq/day (430 nCi/day); mean of 24,400 Bq/day (660 nCi/day). The reported cumulative frequency distribution of the release rates was consistent with a log normal distribution and, therefore, the central value is given by the median. The watches contained 200 to 300 MBg (5.4 to 8.1 mCi) and the fractional release rate is about 3×10⁻⁶ per hour, or 3 ppm/h. The concentration of ³H in the urine of 108 individuals wearing such watches also was measured. Observed concentrations ranged from minimum detectable activity (MDA) to about 1100 Bq/L (30 nCi/L) with a median of 99 Bq/L (2.7 nCi/L); mean of 197 Bq/L (5.3 nCi/L). Exposure time of subjects ranged from 2 to 24 h/day. Comparing the cumulative frequency distribution of the data for tritium release rate from the watches with the measured concentration in urine provides a means for estimating the fractional uptake of tritium by individuals wearing watches. If it is assumed that 50% of the tritium uptake is excreted in the urine (ICRP 23), then the fractional uptake can be calculated, based on a standard excretion rate of 1.4 liters per day, by assuming equilibrium conditions between uptake and excretion. Based on the data for the median, 75%, and maximum values for watch leak rate and for concentration in urine as reported by Brunner et al. (1996), the corresponding uptake can be calculated to be approximately 2%. While the chemical form and actual amounts of tritium uptake via skin absorption versus inhalation are unknown, this 2% absorption is not inconsistent with that measured by Eakins et. al., (1974) for exposure to skin to metal foils containing absorbed tritium.

The dosimetry for tritium uptake from watches is not well established. Johnson and Dunford (1984) modeled the uptake, transfer, and elimination of tritium via skin absorption based on the data from study conducted by Eakins et. al. Their modeling yielded dose factors for the skin dose equivalent per becquerel uptake of 4.5×10^{-8} Sv (1.7×10^5 rem/Ci) and 4.9×10^{-9} Sv (1.8×10^4 rem/Ci) over a 40 cm² skin area for maximum and minimum skin retention times, respectively. As acknowledged in their study, much uncertainty surrounds the modeling and assumptions for the dosimetry for skin uptake of tritium, and at best these dose factors provide a crude estimate for approximating potential dose significance. For purposes of this study, the decision was made to use the maximum skin retention dose factor of 1.8×10^{-3} mSv-cm²/Bq (6.7×10^3 mrem-cm²/ μ Ci), derived from the Johnson and Dunford study by normalizing the uptake for the exposed skin area. This factor provides what is believed to be an upper bound assessment, based on the limited data. It should be recognized that a more probable dose could be an order of magnitude (or more) lower.

To estimate the potential radiation doses due to skin absorption of ³H from a self-luminous wristwatch initially containing 1.9 GBq (50 mCi) of ³H, the analysis proceeds as follows:

- First, the ³H leakage from the self-luminous watch is (1.9 GBq (50 mCi)) × (10 ppb/h) × (16 h/day) or 300 Bq/day (0.008 μ Ci/day), where 10 ppb/h is the assumed rate of HTO leakage from the self-luminous watch (see Section 2.14.4).
- Second, the intake of HTO through the skin in contact with the case of the watch is (300 Bq/d (0.008 μ Ci/d)) × (0.02), or 6 Bq/day (1.6×10⁻⁴ μ Ci/day), where 0.02 is the fractional absorption of ³H released from the watch by the skin in contact with the case of the watch (see above).
- Third, the annual dose equivalent from the HTO to the skin in contact with the case is (6 Bq/d ($1.6 \times 10^{-4} \mu$ Ci/day)) × (365 d/yr) × (1.8×10^{-3} mSv-cm²/Bq

 $(6.7 \times 10^3 \text{ mrem-cm}^2/\mu\text{Ci})) \div (10 \text{ cm}^2)$, or about 0.4 mSv (40 mrem) over an area of 10 cm². The area of skin in contact with the wristwatch is assumed to be about 10 cm² (Turvey, 1996).

The average annual dose equivalent to the skin of the whole body from the distributed wristwatch source is calculated by multiplying the localized skin dose by the ratio of the exposed area of 10 cm² to the total skin area of the body of 1.8×10^4 cm² (ICRP 23) ((0.4 mSv (40 mrem)) × (10 cm²/1.8×10⁴ cm²), or about 2×10⁻⁴ mSv (0.02 mrem). The contribution of this skin dose equivalent to the annual EDE is (2×10⁻⁴ mSv (0.02 mrem) × (0.01), or less than 1×10^{-5} mSv (<0.001 mrem), where 0.01 is the organ weighting factor for skin of the whole body (ICRP 60).

The annual EDE to the internal organs of the body for the absorption of HTO through the skin in contact with the case of the watch is (6 Bq/day ($1.6 \times 10^{-4} \,\mu$ Ci/day)) × ($365 \,day/yr$) × ($1.7 \times 10^{-8} \,m$ Sv/Bq ($6.4 \times 10^{-2} \,m$ rem/ μ Ci)), or $4 \times 10^{-5} \,m$ Sv ($0.004 \,m$ rem), where $1.7 \times 10^{-8} \,m$ Sv/Bq ($6.4 \times 10^{-2} \,m$ rem/ μ Ci) is the dose conversion factor for either absorption through the skin or ingestion of ³H (see Section 2.1).

In summary, the annual dose equivalent to a small area of skin is estimated to be 0.4 mSv (40 mrem) over an area of 10 cm² in contact with the wristwatch. The skin dose due to the distributed wristwatch source of ³H makes a negligible contribution to the annual EDE, and the total annual EDE to a wearer from skin absorption of ³H in contact with the case of a self-luminous watch is estimated to be 4×10^{-5} mSv (0.004 mrem).

The collective EDE of 1 year's distribution of 100,000 wristwatches initially containing 1.9 GBq (50 mCi) of ³H each could be 0.004 person-Sv (0.4 person-rem) during the first year of routine use. The total collective dose is estimated to be about 0.03 person-Sv (3 person-rem) over the 10-year useful life.

2.14.4.2.2 Airborne Releases from Wristwatch

Because watches may be worn in a variety of ways during routine use, the following two scenarios have been chosen to indicate the potential dose from airborne releases of ³H from self-luminous wristwatches.

Scenario I. A wearer spends 12 hours at home each day (4380 h/yr) and exposes three other family members to airborne releases of ³H from the wristwatch. The home has an enclosed volume of 450 m³ and a ventilation rate of 1 volume change per hour. The equilibrium concentration of HTO in the air of the home is approximately 37 MBq/m³ (1 pCi/m³) and the breathing rate of the individuals is 0.9 m³/h. Thus, the annual EDE to the wearer and to other family members would be less than 1×10^{-5} mSv (<0.001 mrem), assuming the other family members are exposed over the same 12 h/day as the wearer.

Scenario II. A wearer works 8 h/day (2000 h/yr) in an office or shop and exposes two coworkers to airborne release of ³H from a wristwatch. The office or shop has an enclosed volume of 34 m³ and a ventilation rate of 1 volume change per hour. The equilibrium concentration of HTO in the air of the office or shop is approximately 0.55 Bq/m³ (15 pCi/m³) and the breathing rate of the individuals is 1.2 m³/h. Thus, the annual EDE to the wearer and

two coworkers would be less than 1×10^{-5} mSv (<0.001 mrem), assuming the coworkers are exposed over the same 8 h/day as the wearer.

To estimate the collective dose from airborne releases of ³H during routine use of the self-luminous watches, it is assumed that 50% of the wearers work outdoors or in areas with high ventilation rates so that their annual EDEs at work are essentially zero, and the other 50% work in small offices or shops with low ventilation rates so that their annual individual EDEs at work are similar to those in Scenario II. While the estimated individual doses are presented as less than values, the calculated values used for estimating collective dose are 4×10^{-6} mSv (4×10^{-4} mrem) for Scenario I and 5×10^{-5} mSv (0.005 mrem) for Scenario II. Thus, the collective EDEs from Scenarios I and II would total about 0.009 person-Sv (0.9 person-rem) for the first year and 0.07 person-Sv (7 person-rem) over 10 year useful life.

2.14.4.2.3 Total Individual and Collective Doses

For a 16-hour-per-day wearer of a self-luminous wristwatch containing 1.9 GBq (50 mCi), the annual dose equivalent to skin from routine use could be 0.4 mSv (40 mrem) over an area of 10 cm² in contact with the wristwatch. The annual individual EDE to such a wearer from routine use could be approximately 4×10^{-5} mSv (0.004 mrem), essentially all due to absorption of ³H through the skin in contact with the case of the watch (see Section 2.14.4.2.1). The individual dose to coworkers and other family members is significantly less.

The total collective dose equivalent to both wearers and other members of the public from routine use of 1 year's distribution of 100,000 watches containing 1.9 GBq (50 mCi) of ³H could be 0.1 person-Sv (10 person-rem). Of this total, 0.03 person-Sv (3 person-rem) is due to exposure to the wearer by absorption of ³H through the skin in contact with the case of the watch (see Section 2.14.4.2.1), and 7×10^{-2} person-Sv (7 person-rem) is due to airborne releases from the watch while at work or at home (see Section 2.14.4.2.2).

2.14.4.3 Watch Repair

Watch repairmen may adjust or replace batteries in the watches. Other repair operations are not considered because work involving batteries is likely to account for the majority of repairs. It is unlikely that ³H will be released catastrophically from the watches during repair, but normal leakage of ³H from the watch will continue. The unlikely breakage of GTLSs in a watch shop is considered in Section 2.14.4.5.

For a repairman at a watch repair shop, the potential doses were estimated using the following scenarios:

- the shop had an enclosed volume of 34 m³ and a ventilation rate of 1 volume change per hour, and the watch repairman was exposed to airborne leakage of ³H from a watch for 1 day (8 hours) between the time the watch was received at the shop and returned to the owner;
- the average time for adjustment and battery replacement was 10 minutes, and the repairman was exposed during this time to airborne releases of ³H into a small hemispherical air space with a radius of 1.5 meters, a volume of 7 m³, and a ventilation rate of 1 volume change per hour; and

 each watch was handled for 10 minutes during watch repair, and 2% of the ³H escaping from the watch was absorbed through a skin area of 3 cm² on the ends of the repairman's fingers.

Based on these assumptions, a breathing rate of 1.2 m³/h for light activity, and the repair of 100 watches per year, the EDE to the watch repairman from all exposure pathways could be 1×10^{-5} mSv (0.001 mrem), and the dose equivalent to the skin could be 0.004 mSv (0.4 mrem) over an area of 3 cm² in contact with the watches.

If the watches are serviced every 2 years mainly to have batteries replaced, the collective EDE to repairmen could be 7×10^{-6} person-Sv (7×10^{-4} person-rem) for servicing one half the watches on an annual basis and 5×10^{-5} person-Sv (0.005 person-rem) over the 10-year useful life of the 100,000 watches distributed annually.

2.14.4.4 Disposal of Watches

Half of the watches are assumed to be disposed of as ordinary trash after 10 years of use, and the other half are assumed to be used for 10 years, then stored in homes for an additional 5 years before disposal. The total amount of ³H in 1 year's distribution of 100,000 watches is estimated to be about 92 TBq (2.5 kCi) at the time of disposal.

To estimate potential individual and collective doses to the public from disposal of self-luminous wristwatches, the generic disposal methodology in Appendix A.2 was used. It has been assumed in applying this methodology that most of the GTLSs remain intact during waste collection and landfill disposal. Thus, a reduction factor of 10 was applied to the following dose-to-source ratios for inhalation and ingestion in Appendix A.2: (1) waste collectors at both landfills and incineration, (2) workers at landfills, (3) off-site members of the public exposed to airborne releases during landfill operations and releases to groundwater following disposal in landfills, and (4) future on-site residents at landfills.

For disposal at landfills, the annual individual EDE would be about less than 1×10^{-5} mSv (<0.001 mrem) to waste collectors. The annual individual doses to workers at landfills, off-site members of the public, and future on-site residents would be less. The total collective EDE was found to be about 2×10^{-3} person-Sv (0.2 person-rem), due almost entirely to exposure to off-site members of the public from groundwater releases.

For disposal by incineration, the annual EDE would be 4×10^{-5} mSv (0.004 mrem) to waste collectors. The annual individual dose is less to workers at incinerators and off-site members of the public. The total collective EDE is about 3×10^{-3} person-Sv (0.3 person-rem), due mainly to exposures to off-site members of the public from airborne releases during incinerator operations.

The above dose estimates are for exposure to multiple exempt units during waste collection and disposal. For exposure to a single exempt unit used for 10 years before disposal, the individual EDE for waste collectors at either landfills or incinerators would be less than 1×10^{-5} mSv (<0.001 mrem). The individual doses to other workers and members of the public from exposure during disposal of a single exempt unit would be less.

2.14.4.5 Accidents and Misuse

In the case of accidents, the following was considered: (1) a catastrophic release from crushing of a single watch in a repair shop, (2) an accident involving the crushing of a single watch in a home, and (3) a shipping accident in a storeroom or cargo-handling area involving the crushing of a shipment of 200 watches. The ³H contained in the watches is assumed to be 99% HT and 1% HTO. Based on these assumptions and the generic accident methodology in Appendix A.1, the potential radiation doses from the crushing of self-luminous watches containing ³H can be summarized as follows:

- For a watch repairman, the individual EDE from crushing of a single watch containing 1.9 GBq (50 mCi) of ³H could be 0.03 mSv (3 mrem) at a small repair shop or 0.02 mSv (2 mrem) at a large repair shop.
- For a person at home, the individual EDE from crushing of a single watch containing 1.9 GBq (50 mCi) of ³H could be 0.001 mSv (0.1 mrem).
- For a worker in a storeroom or cargo-handling area, the individual EDE from crushing of 200 watches containing a total of 370 GBq (10 Ci) of ³H could be 0.1 mSv (10 mrem).

In the case of misuse, this analysis considers the exposure to a 5-year-old child who plays with a self-luminous watch as a "glow-in-the-dark" toy at night while going to sleep during one year. It is assumed that (1) the watch is a 10-year-old watch containing 1.1 GBq (30 mCi) of ³H, (2) the child handles the watch 10 min/day, (3) the child absorbs 2% of the ³H released from the watch through a skin area of 10 cm² while handling the watch, and (4) the child sleeps in a closed bedroom with the watch 12 h/day. It is further assumed that (1) the bedroom has an enclosed volume of 27 m³ and a ventilation rate of 1 air change per hour (see Appendix A.1), (2) the child's breathing rate is 0.24 m³/h while sleeping (ICRP 66), (3) the dose conversion factors for inhalation and ingestion⁸ are about twice those for an adult (ICRP 67; ICRP 71), and (4) the total surface area of the child's skin is approximately 0.8 m² (ICRP 23). Based on these assumptions, the potential radiation doses to the 5-year-old child can be summarized as follows:

- The dose equivalent to the skin of the 5-year-old child due to absorption of ³H from the watch could be 0.002 mSv (0.2 mrem) over a skin area of 10 cm² in contact with the watch.
- The EDEs would be less than 1×10⁻⁵ mSv (<0.001 mrem) due to absorption of ³H through the skin in contact with the watch and 2×10⁻⁵ mSv (0.002 mrem), due to airborne releases of ³H from the watch.

2.14.4.6 Safety Criteria

Table 2.14.2 compares results of the current dose assessment for self-luminous wristwatches with safety criteria for self-luminous products set forth in 10 CFR 32.23 (see Section 2.14.1).

⁸ The dose conversion factors for effective dose equivalent due to ingestion of ³H or absorption of ³H through the skin are the same numerically.

First, the limiting 10 CFR 32.23 values for dose to the whole body are compared with the current assessment's maximum estimates of the annual individual EDE due to ³H releases from the self-luminous watches for the current assessment. Second, the limiting 10 CFR 32.23 values for dose to skin are compared with the current assessment's maximum estimates of the annual dose equivalent to skin of individuals exposed by ³H absorption through the skin in contact with the watches. It is important to note that the skin dose from normal use of a single self-luminous wristwatch could exceed the limiting skin dose set forth in 10 CFR 32.23. This result contains some rather large uncertainties for the following reasons: (1) the rate of leakage of ³H from the GTLSs in the self-luminous watches is not well known, and (2) the conversion factor for relating skin dose to absorption of ³H through the skin in contact with the wristwatch is not well established (see Section 2.14.4.2.1). However, the currently estimated annual dose equivalent of 0.4 mSv (40 mrem) for skin of a wearer in contact with the self-luminous wristwatch is much less than the annual dose limit of 50 mSv (5 rem) for skin of members of the public that is recommended in both ICRP 60 and NCRP 116.

2.14.5 Current Assessment for Gun Sights

Table 2.14.3 provides the results of our assessment of potential radiation doses from an assumed annual distribution of 100,000 handgun sights initially containing 1.9 GBq (50 mCi) of ³H each. It should be noted that the average activity of the handgun sights is significantly less than the maximum activity of 11 GBq (300 mCi) used under this exemption for rifle scopes, and the number of rifle scopes distributed annually is quite small in comparison to the number of handgun sights. Hence, individual doses resulting from exposures to a single exempt item during routine use and during accidents are assumed to be controlled by the maximum amount of activity in a rifle scope. All other individual doses resulting from exposure to multiple exempt items and, thereby, all collective doses are assumed to be controlled by the average activity per handgun sight and by the number distributed annually. However, individual doses based on both rifle scopes and handgun sights for routine exposures are provided here, because the latter were developed for use in the estimation of collective dose. It is further assumed that the useful lifetime of the handgun sights and rifle scopes is 10 years, and the leakage rate of ³H is 10 ppb/h.

2.14.5.1 Distribution and Transport

The potential radiation doses from distribution and transport of the gun sights are estimated using the generic methodology of Appendix A.3. In applying this methodology, it is assumed that local parcel-delivery drivers in large trucks pick up the self-luminous gun sights from suppliers and take them to a local terminal, where they are shipped by semi-truck to other local terminals for delivery to gunsmiths or manufacturers. Furthermore, it is assumed that each shipment passes through an average of four regional terminals before reaching its final destination. It is also assumed that the radiation dose to workers at both local terminals and regional terminals are the same as those estimated for workers in a large warehouse and the leakage rate from the gun sights is 10 ppb/h, or 100 times less than the value of 1 ppm/h used in the development of the generic methodology in Appendix A.3.

Most of the handgun sights are assumed to be shipped in lots of 1000 or less to gun manufacturers, who mount the sights as an option for buyers. These handguns are then picked up by local drivers and returned to local terminals, where they are shipped to other local

terminals for delivery to the gun buyers. It is assumed further that essentially all of the gun sights are mounted by installers in factories and the above scenario applies during shipment of these mounted sights to gun buyers. Based on these assumptions and generic methodology in Appendix A.3, the individual EDE could be 0.001 mSv (0.1 mrem) for a local parcel-delivery driver who is assumed to pick up 1000 unmounted gun sights from the same supplier each week (50 wk/yr). Individual doses to other truck drivers, terminal workers, and members of the public along truck routes would be less. The total collective EDE from distribution of a total of 100,000 unmounted gun sights could be 3×10^{-5} person-Sv (0.003 person-rem).

2.14.5.2 Gun Sight Installers

In a gun repair shop, the time taken by a gunsmith to mount and align a handgun sight is estimated to be about 30 minutes, and in a factory, about 10 to 15 minutes. A factory installer could mount and align about 5000 sights per year (100 sights per week for 50 wk/yr). A total of about 20 factory installers would be needed to mount and align 1 year's distribution of 100,000 gun sights if it is assumed that essentially all gun sights are installed at factories. It is unlikely that ³H will be released catastrophically from the gun sights during installation, but normal leakage of ³H will continue. The unlikely breakage of GTLSs in a self-luminous sight at a gun repair shop is considered in Section 2.14.5.5.

For a gun installer at a factory, the potential doses were estimated using the follow scenarios: (1) the installer was exposed for 40 h/wk (2000 h/yr) to airborne releases from 1 week's supply of 100 gun sights into a hemispherical air space with a radius of 1.5 meters, a volume of 7 m³, and a ventilation rate of 5 volume changes per hour, and (2) each sight was handled for 15 minutes while being installed on a handgun and 2% of the ³H escaping from the sight was absorbed through a skin area of about 3 cm² on the ends of the installer's fingers. Based on these assumptions, a breathing rate of 1.2 m³/h for light activity, and the installation of 5000 gun sights per year, the annual EDE to a single gun installer could be 0.003 mSv (0.3 mrem), essentially all from inhalation intake. The annual dose equivalent to the skin could be 0.3 mSv (30 mrem) over a skin area of 3 cm² in contact with the gun sights. The collective EDE to all gun installers from 1 year's distribution of 100,000 gun sights is estimated to be approximately 7×10^{-5} person-Sv (0.007 person-rem).

2.14.5.3 Routine Use

Because a handgun with self-luminous sights may be used in a variety of ways by law enforcement agents and private gun owners, the following two scenarios have been chosen to indicate potential doses from routine use.

Scenario I. A police officer spends 6 h/day (1500 h/yr) in a cruiser with another police officer (exposure to sights on two guns) and 12 h/day (4380 h/yr) at home (exposure to sights on one gun). The cruiser is assumed to have a volume of 6.2 m^3 and a ventilation rate of 5 volume changes per hour, and the residence is assumed to have a volume of 450 m³ and a ventilation rate of 1 volume change per hour (see Appendix A.1). The breathing rates are assumed to be 0.9 and 1.2 m³/h while at home and at work, respectively.

The equilibrium concentrations of ³H in the cruiser and in the home are approximately 1.11 Bq/m³ (30 pCi/m³) and 0.037 Bq/m³ (1 pCi/m³), respectively. For the police officer, the annual EDE is estimated to be 6×10^{-5} mSv (0.006 mrem) in the first year. For an assumed

average of three other family members, the initial annual EDEs are estimated to be less than 1×10^{-5} mSv (<0.001 mrem), assuming the other family members are exposed at home over the same 12-hour period as the police officer.

Scenario II. A private gun owner stores the handgun at home for protection. The equilibrium concentration is 0.037 Bq/m³ (1 pCi/m³), under the above assumptions, and the annual EDE is less than 1×10^{-5} mSv (<0.001 mrem) for the gun owner and three other family members, assuming they each spend 12 h/day at home.

To estimate the collective EDE from the routine use of the 100,000 handgun sights used for 10 years, it was assumed that 20% of the sights were purchased by private gun owners and 80% were purchased for police officers (or other law enforcement agents). The collective EDE would total about 4×10^{-3} person-Sv (4 person-rem), essentially all attributable to police officers, for 1 year's distribution of 100,000 handgun sights initially containing 1.9 GBq (50 mCi) of ³H each.

In the case of a private gun owner who stores a rifle with a self-luminous scope containing 11 GBq (300 mCi) of ³H at home, the annual individual EDE could be 3×10^{-5} mSv (0.003 mrem) the case of a police officer who patrols alone and carries such a rifle in a police cruiser, the annual individual EDE could be 3×10^{-4} mSv (0.03 mrem). These estimates are based on the same assumptions as above, except the rifle used by the police officer is assumed to be stored at a police station when not in use, rather than at home.

2.14.5.4 Disposal

To estimate the potential radiation doses due to the disposal of self-luminous gun sights containing ³H, the generic disposal methodology of Appendix A.2 has been used. It is assumed in applying this methodology that most of the GTLSs remain intact during waste collection and landfill disposal. Thus, a reduction by a factor of 10 is applied to the following dose-to-source ratios for inhalation and ingestion in Appendix A.2: (1) waste collectors at both landfills and incinerators, (2) workers at landfills, (3) off-site members of the public exposed to airborne releases during landfill operation and releases to groundwater following disposal in landfills, and (4) future on-site residents. The total activity of ³H from 1 year's distribution of 100,000 gun sights after 10 years of use is approximately 110 TBq (3,000 Ci).

For disposal at landfills, the annual individual EDE would be less than 1×10^{-5} mSv (<0.001 mrem) to waste collectors. The annual individual doses to workers at landfills, off-site members of the public, and future on-site residents would be less. The total collective EDE was found to be about 3×10^{-3} person-Sv (0.3 person-rem), due almost entirely to exposures to off-site members of the public from groundwater releases.

For disposal by incineration, the annual EDE would be 4×10^{-5} mSv (0.004 mrem) to waste collectors. The annual individual dose to workers at incinerators and off-site members of the public are less. The total collective EDE is about 4×10^{-3} person-Sv (0.4 person-rem), due mainly to exposures to off-site members of the public from airborne releases during incinerator operations.

The above dose estimates are for exposure to multiple exempt units during waste collection and disposal. For a gun sight containing 1.9 GBq (50 mCi) of ³H initially, the activity is about

1.1 GBq (30 mCi) at time of disposal, and the individual EDE to a waste collector would be less than 1×10^{-5} mSv (<0.001 mrem). For a rifle scope containing 11 GBq (300 mCi) of ³H initially, the activity is about 6.7 GBq (180 mCi) at time of disposal, and the individual EDE to a waste collector would also be less than 1×10^{-5} mSv (<0.001 mrem).

2.14.5.5 Accidents and Misuse

In the case of accidents, the following scenarios have been considered: (1) a catastrophic release from crushing of a rifle scope in a repair shop, (2) an accident involving the crushing of a rifle scope in a home, and (3) a shipping accident in a storeroom or cargo-handling area involving the crushing of a shipment of 1000 handgun sights. The ³H contained in the rifle scopes and handgun sights is assumed to be 99% HT and 1% HTO. Based on these assumptions and the generic accident methodology in Appendix A.1, the potential radiation doses from the crushing of self-luminous watches containing ³H can be summarized as follows:

- For a gunsmith at a repair shop (see data for watch repair shops in Tables A.1.2 and A.1.9 of Appendix A.1), the individual EDE from crushing of a rifle scope containing 11 GBq (300 mCi) of ³H could be 0.2 mSv (20 mrem) at a small shop or 0.1 mSv (10 mrem) at a large shop.
- For a person at home (see Tables A.1.2. and A.1.9 of Appendix A.1), the individual EDE from crushing of a rifle scope containing 11 GBq (300 mCi) of ³H could be 8×10⁻³ mSv (0.8 mrem).
- For a worker in a storeroom or cargo-handling area (see Tables A.1.2 and A.1.9 of Appendix A.1), the individual EDE from crushing of 1000 handgun sights containing a total of 1.9 TBq (50 Ci) of ³H could be 0.5 mSv (50 mrem).

In the case of misuse, the analysis considers the exposure to a 5-year-old child who plays with a self-luminous sight from a handgun as a "glow-in-the-dark" toy at night while going to sleep during one year. It is assumed that the (1) handgun sight is a 10-year-old sight containing 1.1 GBq (30 mCi) of ³H, (2) the child handles the sight 10 min/day, (3) the child absorbs 2% of the ³H released from the sight through a skin area of 3 cm² while handling the sight, and (4) the child sleeps in a closed bedroom with the sight 12 h/day. It is further assumed that (1) the bedroom has an enclosed volume of 27 m³ and a ventilation rate of 1 air change per hour (see Appendix A.1), (2) the child's breathing rate is 0.24 m³/h while sleeping (ICRP 66), (3) the dose conversion factors for inhalation and ingestion⁹ are about twice those for an adult (ICRP 67; ICRP 71), and (4) the total surface area of the child's skin is approximately 0.8 m² (ICRP 23). Based on these assumptions, the potential radiation doses to the 5-year-old child can be summarized as follows:

• The dose equivalent to the skin of the 5-year-old child due to absorption of ³H from the gun sight could be 0.008 mSv (0.8 mrem) when averaged over 3 cm² of skin area in contact with the sight.

⁹ The dose conversion factors for EDE due to ingestion of ³H or absorption of ³H through the skin are the same numerically.

The EDE would be less than 1×10⁻⁵ mSv (<0.001 mrem) due to absorption of ³H through the skin in contact with the sight and 2×10⁻⁵ mSv (0.002 mrem), due to airborne releases of ³H from the sight.

2.14.5.6 Safety Criteria

Table 2.14.4 compares the results of the current dose assessment for self-luminous gun sights with the safety criteria for self-luminous products set forth in 10 CFR 32.23 (see Section 2.14.1). First, the limiting 10 CFR 32.23 values for dose to the whole body are compared with the current assessment's maximum estimates of the annual individual EDE due to ³H releases from the self-luminous gun sights. Second, the limiting 10 CFR 32.23 values for dose to skin are compared with the current assessment's maximum estimates of the annual dose equivalent to skin of individuals exposed by ³H absorption through the skin in contact with the sights. The individual doses to the whole body (EDE) and the dose equivalents to skin do not exceed the dose limits set forth in 10 CFR 32.23 (see Table 2.14.6).

2.14.6 Summary

Tables 2.14.1 and 2.14.3 present results of the current assessments of potential radiation doses to the public from self-luminous products containing ³H. These results are based on an annual distribution of 100,000 self-luminous wristwatches and 100,000 self-luminous gun sights. The self-luminous wristwatches are assumed to contain 1.9 GBq (50 mCi) of ³H per watch, and the self-luminous gun sights are assumed to contain from 1.9 to 11 GBq (50 to 300 mCi) of ³H per sight. The watches and sights are both assumed to have an effective lifetime of 10 years.

The estimate of the total collective EDE to the public from self-luminous products is 0.2 person-Sv (20 person-rem) (see Tables 2.14.1 and 2.14.3). This estimate is smaller than the previous estimate of 77 person-Sv (7700 person-rem) in NCRP 95. Part of the difference is due to design modifications of the wristwatches and assumptions regarding the annual distribution of these consumer products (see Section 2.14.2.1). However, the difference is largely due to the conservative assumptions used in making dose estimates for landfill burial and incineration of self-luminous products in the previous study by Buckley et al. (NUREG/CR–1775) (see, also, Section 2.14.3).

Tables 2.14.2 and 2.14.4 also present comparisons of the current estimates of annual radiation doses to individuals with the safety criteria in 10 CFR 32.23 (see Section 2.14.1). It is important to note that the annual dose equivalent to skin from normal use of a self-luminous watch could exceed the skin dose limit set forth in 10 CFR 32.23 (see Table 2.14.3). This result contains some rather large uncertainties for the following reasons: (1) the rate of ³H leakage from the GTLSs in the self-luminous wristwatches is not well known, and (2) the conversion factor for relating skin dose to absorption of ³H through the skin in contact with the wristwatch is not well established (see Section 2.14.4.2.1). Therefore, studies are needed to obtain better data for both of these important dose-related parameters.

Exposure Pathway	Individual Annual Effective Dose Equivalent (mrem)ª	Collective Effective Dose Equivalent ^b (person-rem) ^a
Distribution and transport	0.1°	1.0
Routine use	0.004 ^d	10
Watch repair	0.001 ^e	0.005
<u>Disposal</u> Landfills Incinerators	<0.001 ^f 0.004 ^g	0.2 0.3
Accidents and misuse Toy for a child Crushing of watches	0.003 ^h 10 ⁱ	

Table 2.14.1 Potential Radiation Doses From Self-Luminous Wristwatches Containing ³H

^a 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

^b Collective doses are based on an assumed annual distribution of 100,000 wristwatches having an effective lifetime of 10 years and initially containing 1.9 GBq (50 mCi) of gaseous ³H per watch. Refer to text for time period of calculations.

^c Dose estimate applies to a local parcel-delivery driver exposed to multiple exempt units during distribution and transport; dose estimates are less for other truck drivers, workers at truck terminals and distribution warehouses, store clerks and shoppers, and members of public along truck routes (see Section 2.14.4.1).

^d Dose estimate applies to 16-hour-per-day wearers of wristwatches; dose estimate for annual dose equivalent to skin of such wearers is 0.4 mSv (40 mrem) over an area of 10 cm² in contact with the case of the wristwatch (see Section 2.14.4.2).

^e Dose estimate applies to watch repairmen exposed to multiple exempt units; dose estimate for annual dose equivalent to skin is 0.004 mSv (0.4 mrem) over a skin area of 3 cm² in contact with watches (see Section 2.14.4.3).

^f Dose estimate applies to waste collectors exposed to multiple exempt units during disposal at landfills; dose estimates are less for workers at landfills, off-site members of public, and future on-site residents (see Section 2.14.4.4).

⁹ Dose estimate applies to waste collectors exposed to multiple exempt units during disposal at incinerators; dose estimates are less for workers at incinerators and off-site members of public (see Section 2.14.4.4).

^h Dose estimate applies to a 5-year-old child who is allowed to use an old self-luminous watch as a toy during a year; dose estimate for dose equivalent to skin is less than 1×10^{-5} mSv (<0.001 mrem) due to absorption of ³H through the skin and 2×10^{-5} mSv (0.002 mrem) due to airborne releases of ³H from the watch (see Section 2.14.4.5).

¹ Dose estimate applies to a worker exposed during a shipping accident in a storeroom or cargo-handling bay involving the crushing of 200 self-luminous watches; dose estimate for a watch repairman following the crushing of a single exempt unit in a small watch repair shop is 0.03 mSv (3 mrem) (see Section 2.14.4.5).

Exposure Conditions ^a	Maximum Individual Dose (mrem/yr or mrem)⁵	Regulatory Limit for Organ Dose (mrem/yr or rem) ^ь			
EFFECTIVE DOSE EQUIVALENT					
Routine use and disposal of a single exempt unit	0.004 mrem/yr ^c	1 mrem/yr ^d			
Accidents involving a single exempt unit	3 mrem ^e	0.5 rem ^f 15 rem ^g			
Normal handling and storage of multiple exempt units	0.1 mrem/yr ^h	10 mrem/yr ⁱ			
Accidents involving multiple exempt units	10 mrem ^j	0.5 rem ^f 15 rem ^g			
DOSE EQUIVALENT TO SKIN					
Normal use and disposal of a single unit	40 mrem/yr ^k	15 mrem/yr			

Table 2.14.2 Comparison of Estimated Maximum Individual Doses From Self-Luminous Wristwatches Containing ³H and Limiting Organ Doses From 10 CFR 32.23

^a See Section 2.14.1 of this report.

^b 1 mrem = 0.01 mSv; 1 rem = 0.01 Sv

^c Dose estimate applies to a 16-hour-per-day wearer of a self-luminous wristwatches (see Section 2.14.4.2).

^d Limit on whole-body dose during normal use and disposal of a single exempt unit (see Section 2.14.1).

^e Dose estimate applies to a watch repairman following crushing of a single watch in a small repair shop (see Section 2.14.4.5).

¹ Limit on whole-body dose when probability of failure of safety features in product is less than one failure per year for each 10,000 exempt units distributed (see Section 2.14.1).

⁹ Limit on whole-body dose when probability of failure of safety features in product is less than one failure per year for each 1 million exempt units distributed (see Section 2.14.1).

^h Dose estimate applies to a local parcel-delivery driver (see Section 2.14.4.1).

¹ Limit on whole-body dose during normal handling and storage of multiple exempt units (see Section 2.14.1).

¹ Dose estimate applies to a worker exposed during a shipping accident involving the crushing of 200 self-luminous watches (see Section 2.14.4.5).

^k Skin dose to a 16-hour-per-day wearer of a self-luminous wristwatch (see Section 2.14.4.2).

Exposure Pathway	Individual Annual Effective Dose Equivalent (mrem)ª	Collective Effective Dose Equivalent ^ь (person-rem)ª
Distribution and transport	0.1°	0.003
Gun sight installers	0.3 ^d	0.007
Routine use	0.03 ^e	4
<u>Disposal</u> Landfills Incinerators	<0.001 ^f 0.004 ^g	0.3 0.4
<u>Accidents and misuse</u> Toy for a child Crushing of gun sights	0.002 ^h 50 ⁱ	

Table 2.14.3 Potential Radiation Doses From Self-Luminous Gun Sights Containing ³H

^a 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

^b Collective doses are based on an assumed annual distribution of 100,000 handgun sights having an effective lifetime of 10 years and initially containing 1.9 GBq (50 mCi) of gaseous ³H per sight. Refer to text for time period for collective dose calculations.

^c Dose estimate applies to a local parcel-delivery driver exposed to multiple exempt units during distribution and transport; dose estimates are less for other truck drivers, workers at truck terminals, store clerks and customers, and members of public along truck routes (see Section 2.14.5.1).

^d Dose estimate applies to a factory installer who mounts and aligns 5,000 gun sights per year; dose estimate for annual dose equivalent to skin of a factory installer is 0.3 mSv (30 mrem) when averaged over 3 cm² of skin area in contact with gun sights (see Section 2.14.5.2).

^e Dose estimate applies to police officer who patrols alone in a police cruiser and carries a rifle with a self-luminous scope containing 11 GBq (300 mCi) of ³H (Section 2.14.5.3).

[†] Dose estimate applies to waste collectors exposed to multiple exempt units during disposal at landfills; dose estimates are less for workers at landfills, off-site members of the public, and future on-site residents (see Section 2.14.5.4).

⁹ Dose estimate applies to waste collectors exposed to multiple exempt units during disposal at incinerators; dose estimates are less for workers at incinerators and off-site members of the public (see Section 2.14.5.4).

^h Dose estimate applies to a 5-year-old child who is allowed to use an old self-luminous gunsight as a toy during one year; dose estimate for dose equivalent to skin is 0.008 mSv (0.8 mrem) when averaged over a skin area of 3 cm² in contact with gun sight (see Section 2.14.5.5).

¹ Dose estimate applies to a worker exposed during a shipping accident in a storeroom or cargo-handling bay involving the crushing of 1,000 handgun sights containing 1.9 GBq (50 mCi) of ³H each; dose estimate is 0.2 mSv (20 mrem) for a gunsmith at a small gun shop following the crushing of a single rifle scope containing 11 GBq (300 mCi) of ³H (see Section 2.14.5.5).

Table 2.14.4 Comparison of Estimated Maximum Individual Doses From
Self-Luminous Gun Sights Containing ³H and Limiting Organ Doses
From 10 CFR 32.23

Exposure Conditions ^a	Maximum Individual Dose (mrem/yr or mrem) ^ь	Regulatory Limit For Organ Dose (mrem/yr or rem) ^b			
EFFECTIVE DOSE EQUIVALENT					
Routine use and disposal of a single exempt unit	0.03 mrem/yr ^c	1 mrem/yr ^d			
Accidents involving a single exempt unit	20 mrem ^e	0.5 rem ^f 15 rem ⁹			
Normal handling and storage of multiple exempt units	0.3 mrem/yr ^h	10 mrem/yr ⁱ			
Accidents involving multiple exempt units	50 mrem ⁱ	0.5 rem ^f 15 rem ^g			
DOSE EQUIVALENT TO SKIN					
Normal handling and storage of multiple units	30 mrem/yr ^k	150 mrem/yr			

^a See Section 2.14.1 of this report.

^b 1 mrem = 0.01 mSv; 1 rem = 0.01 Sv.

^c Dose estimate applies to a police officer using a self-luminous rifle scope containing 11 GBq (300 mCi) of ³H (see Section 2.14.5.3).

^d Limit on whole-body dose during normal use and disposal of a single exempt unit (see Section 2.14.1).

^e Dose estimate applies to a gunsmith in a small gun shop following crushing of a self-luminous rifle scope containing 11 GBq (300 mCi) of ³H (see Section 2.14.5.5).

^t Limit on whole-body dose when probability of failure of safety features in product is less than one failure per year for each 10,000 exempt units distributed (see Section 2.14.1).

⁹ Limit on whole-body dose when probability of failure of safety features in product is less than one failure per year for each 1 million exempt units distributed (see Section 2.14.1).

^h Dose estimate applies to an installer at a factory exposed to multiple self-luminous handgun sights containing 1.9 GBq (50 mCi) of ³H each (see Section 2.14.5.2).

¹ Limit on whole-body dose during normal handling and storage of multiple exempt units. ¹ Dose estimate applies to a worker exposed during a shipping accident involving the crushing of 1,000 handgun sights containing 1.9 GBq (50 mCi) of ³H (see Section 2.14.5.5).

^k Skin dose for an installer at a factory exposed to multiple self-luminous handgun sights containing 1.9 GBq (50 mCi) of ³H each (see Section 2.14.5.2).

2.15 Gas and Aerosol Detectors

2.15.1 Introduction

In 10 CFR 30.20, any person is exempt from licensing requirements to the extent that the person receives, possesses, uses, transfers, owns, or acquires byproduct material in gas and aerosol detectors designed to protect life or property from fire and airborne hazards. This "class exemption" was proposed on November 1, 1968 (33 FR 16089), and issued as a final rule on April 18, 1969 (34 FR 6653).

Requirements for a license to manufacture, process, produce, or initially transfer gas and aerosol detectors intended for use under 10 CFR 30.20 are contained in 10 CFR 32.26. Conditions of license are also included in 10 CFR 32.29; these include requirements for quality control, labeling, and reporting of transfers.

Section 32.26 specifies that the license applicant must demonstrate the product will meet certain safety criteria set forth in 10 CFR 32.27 and includes the specific information to be submitted with the application. The safety criteria are the primary factors in controlling the doses associated with this exemption and are described below:

- In normal use and disposal of a single exempt unit and in normal handling and storage of the quantities of exempt units that are likely to accumulate in one location during marketing, distribution, installation, and servicing of the product, it is unlikely that the annual dose from external exposure, or the annual dose commitment resulting from intake of radioactive material, to a suitable sample of the group of individuals expected to be the most highly exposed to radiation or radioactive material from the product will exceed 0.05 millisievert (mSv) (5 mrem) to the whole body, head and trunk, active blood-forming organs, gonads, or lens of the eye; 0.75 mSv (75 mrem) to the hands and forearms, feet and ankles, or localized areas of the skin averaged over areas no larger than 1 cm²; and 0.15 mSv (15 mrem) to any other organs.
- It is unlikely there will be a significant reduction in the effectiveness of the containment, shielding, or other safety features of the product from wear and abuse that is likely to occur in normal handling and use of the product during its useful life.
- In use and disposal of a single exempt unit and in handling and storage of the quantities of exempt units that are likely to accumulate in one location during marketing, distribution, installation, and servicing of the product, it is unlikely that the containment, shielding, or other safety features of the product would fail under such circumstances that an individual would receive an external dose or internal dose commitment in excess of 5 mSv (0.5 rem) to the whole body, head and trunk, active blood-forming organs, gonads, or lens of the eye; 75 mSv (7.5 rem) to the hands and forearms, feet and ankles, or localized areas of the skin averaged over areas no larger than 1 cm²; and 15 mSv (1.5 rem) to any other organs. The likelihood is negligible that an individual would receive an external dose or dose commitment in excess of 150 mSv (15 rem) to the whole body, head and trunk, active blood-forming organs, cor localized areas of the skin averaged over areas no larger than 1 cm²; and 15 mSv (1.5 rem) to any other organs. The likelihood is negligible that an individual would receive an external dose or dose commitment in excess of 150 mSv (15 rem) to the whole body, head and trunk, active blood-forming organs, gonads, or lens of the eye; 2 Sv (200 rem) to the hands and forearms, feet and ankles, or localized areas of

the skin averaged over areas no larger than 1 $\rm cm^2$; and 0.5 Sv (50 rem) to any other organs.

These safety criteria have the effect of establishing an upper limit on the amount of radioactive material in an exempt unit and of establishing a need to provide some degree of containment for the material even under accident conditions. Furthermore, the probabilities associated with the dose limits for accidental exposures ensure that the risk to individuals would be very small.

In the last set of safety criteria for accident exposures, the guidance for quantifying low and negligible probabilities of failure of safety features in products for purposes of demonstrating compliance with the criteria is the same as that previously described for self-luminous products in Section 2.14.1 (i.e., a probability of failure is "low" if there is not more than one failure per year for each 10,000 exempt units distributed, and "negligible" if there is not more than one failure per year for each 1 million exempt units distributed).

The "class exemption" for self-luminous products (see Section 2.14.1) set out the concept of safety criteria (dose limits for individuals) that consumer products containing tritium (³H), ⁸⁵Kr, or ¹⁴⁷Pm must meet under conditions of handling, storage, use, and disposal. The "class exemption" for gas and aerosol detectors carried that concept one step further by not specifying the byproduct material that may be used in the product. Instead, it charged the manufacturer or importer (license applicant) with exercising care in the selection of the type and quantity of the radionuclide and in the design and construction of containment and shielding features used to meet the safety criteria. Both "class exemptions" provide for consideration of the probability and consequences of failure of containment, shielding, or other safety features of the product. The safety criteria for gas and aerosol detectors are similar to the safety criteria for self-luminous products containing ³H, ⁸⁵Kr, and ²⁴¹Am, except only a single set of criteria with somewhat different dose limits is specified for normal handling, storage, use, and disposal of gas and aerosol detectors.

Therefore, the safety criteria for gas and aerosol detectors provide: (1) radiation dose limits for individuals from normal handling, storage, use, and disposal of these products, and (2) radiation dose limits for individuals in conjunction with approximate associated probabilities of occurrence for accidents. The safety criteria do not include any limits on the collective radiation dose to members of the public.

2.15.2 Description of Items

The most commonly used products that have been distributed for use under this "class exemption" are smoke detectors containing ²⁴¹Am and chemical detectors containing either ²⁴¹Am or ⁶³Ni. The radionuclide ⁶³Ni has also been used in smoke detectors to some degree. The major decay mode of ²⁴¹Am is alpha-particle emission, and the major decay mode of ⁶³Ni is beta-particle emission, thereby limiting external exposure during routine use of these devices. The photons emitted during the decay of ²⁴¹Am have energies of only 69 keV or less, and only very low-energy photons from bremsstrahlung are produced during the decay of ⁶³Ni.

2.15.2.1 Smoke Detectors

For more than a decade, residential deaths from fire in the United States have declined by an average of 4% per year (Smoke Detectors, 1994). This can be attributed to the increased use of smoke detectors in U.S. homes during the late 1970s and early 1980s. The two most common types of smoke detectors are optical smoke detectors (OSDs), also referred to as photoelectric smoke detectors, and ionization chamber smoke detectors (ICSDs). OSDs work best with smoke from smoldering fires that contain large particles, while ICSDs work best with smoke from fast-burning fires that contain smaller particles with diameters of about 0.01 to 1 μ m (Sacks, 1986). Some fire prevention experts recommend installation of both OSDs and ICSDs in the home.

An ICSD consists of an ionization chamber, electronic circuitry, a power source that is usually a battery, an alarm mechanism, and an outer case. The ionization chamber is the main component. It consists of a source of ionizing radiation, usually ²⁴¹Am, positioned between two oppositely charged electrodes. Alpha particles emitted during radioactive decay of the ²⁴¹Am interact with neutral air molecules flowing through the chamber and convert them to positive ions by removal of electrons. The removed electrons then form negative ions by attachment to other neutral molecules. The resulting positive and negative ions are attracted toward the electrodes, causing a small, reasonably steady current between the electrodes. The electronic circuitry monitors this current and, if the current drops below a preset level, triggers an audible alarm.

Under normal conditions, there is an equilibrium between ion production in the chamber and ion removal from the chamber by either striking an electrode, ion recombination, or being carried out of the chamber by air flow. However, if the air entering the chamber contains combustion products or smoke particles that are more massive than the air molecules, this equilibrium will be disturbed. The smoke particles will also become charged by alpha-particle interactions, electron capture, and combination with existing ions. The charged smoke particles move toward the electrodes more slowly because they are more massive than the ionized air molecules. This allows an excess of charged smoke particles to be swept out of the chamber by air flow before reaching the electrodes. The net effect is a reduction in the current flowing between the electrodes. When the current decreases below a preset level, the alarm is activated to signal the very early stages of a fire.

Between 1971 and 1986, Nuclear Regulatory Commission (NRC) records (NRC, Unpublished Reports, 1989) indicate that 124,000 ICSDs containing a total of 44 gigabecquerel (GBq) (1.2 curie (Ci)) of ⁶³Ni and 92 million ICSDs containing a total of 12 TBq (320 Ci) of ²⁴¹Am were sold in the United States. In 1973, only 250,000 ICSDs were sold, with most of the units going into public and commercial buildings (NCRP 95). These units contained a total of 480 GBq (13 Ci) of ²⁴¹Am, with an average of approximately 1.9 MBq (50 μ Ci) per unit. From 1973 to 1978, the sales of ICSDs increased dramatically (O'Donnell et al., 1981), and the number of homes with ICSDs increased from 10 to 77% (Hall, 1989). In 1978, approximately 14 million ICSDs were sold, with most of the units going into homes. These units contained a total of 1.7 TBq (46 Ci) of ²⁴¹Am, with an average of 0.11 MBq (3 μ Ci) per unit (NUREG/CR–1156).

It is currently estimated that more than 80% of U.S., homes have 1 or more ICSDs and the total number of ICSDs in residential use is approximately 100 million. An ICSD should work for about 10 years, after which it should be discarded and replaced by a new smoke detector

(Smoke Detectors, 1994). Thus, the annual distribution of ICSDs in the United States currently totals about 10 million units. The average ²⁴¹Am activity per unit is approximately 37 kBq (1 μ Ci), and the total number of curies of ²⁴¹Am distributed annually in this product is approximately 0.37 TBq (10 Ci).

The specific design features of the smoke detectors vary among manufacturers, but the following features are common to all designs (O'Donnell et al., 1981). First, the source is normally cut or punched from a composite material that consists of gold and ²⁴¹Am that is hot-forged onto a 0.2-mm-thick silver backing and covered by a 0.001- to 0.002-mm-thick gold foil to form a sealed source, typically 3 to 5 mm in diameter. Next, the sealed source is either pressed and crimped into a recess or electron beam welded within the ionization chamber. Finally, the ionization chamber is also sealed, except for screen-covered openings to allow gas and smoke particulates to flow into the ICSD. The detector designs require deliberate tampering in order to remove the ²⁴¹Am source.

2.15.2.2 Chemical Detectors

Chemical detectors are used to monitor for harmful or toxic gases and a variety of vapors. They may be used by industry to monitor for leakage of gases such as sulfur hexafluoride, refrigerants, and products of combustion; by the military to monitor for chemical warfare agents such as nerve and blister gases; by hospitals to monitor for waste anesthetic gases; and by airports to monitor for vapors from explosives or narcotic substances. The chemical detectors may be designed for use as either a fixed monitor or a portable instrument for field measurements. Typical chemical detectors use sources containing a total of either 0.37 GBq (10 mCi) of ⁶³Ni per detector or 5.9 MBq (0.16 mCi) of ²⁴¹Am per detector.

A chemical detector consists typically of a detector cell, electronic circuitry, a power source, an air pump, a heater, and an outside case. The detector cells contain a radioactive source which is usually coiled into a cylindrical shape with the radioactive side inward. The technology used in making the ²⁴¹Am sources is similar to that used in making the ²⁴¹Am sources for smoke detectors (see Section 2.15.2.1). The ⁶³Ni sources are made by electroplating the nickel onto a metallic foil, which can then be formed into a cylindrical source. Tamper-resistant screws are used to restrict human access to the radioactive sources in the chemical detectors.

Air is drawn into the detector through air flow tubes via the internal pump, heated as it is drawn past the detector cell, and expelled as exhaust. The air flow tubes are too small in diameter to permit human access, and bends in them block any direct radiation path from the sources. As the heated air passes over the radioactive sources, electrons and positive ions are formed, which cause a current in the detector cell. The free electrons are readily captured by many gases and vapors, causing a reduction of the current in the detection cell. If the current drops below a present level, a visual and audible alarm is given to alert the operator or others in the vicinity of the chemical detector.

Chemical detectors are relatively new products whose total distribution is not accurately known at present and whose distribution will probably increase quite rapidly in the future. Thus, it is assumed here that 10,000 chemical detectors containing a total of 59 GBq (1.6 Ci) of ²⁴¹Am and 10,000 chemical detectors containing a total of 3.7 TBq (100 Ci) of ⁶³Ni are distributed annually, and the effective lifetime of the detectors is approximately 10 years.

2.15.3 Summary of Previous Analyses and Assessments

Assessments of the potential radiological impacts on the public from smoke detectors containing ²⁴¹Am sources have been performed by Belanger et al. (NUREG/CR–1156) and O'Donnell et al. (1981), and the results of these assessments have been summarized in NCRP 95. Belanger et al. also evaluated the potential radiological impacts on the public from smoke detectors containing ⁶³Ni and ²²⁶Ra sources. However, the manufacture of smoke detectors containing ²²⁶Ra sources ceased following the development of ²⁴¹Am sources for smoke detectors, and only a relatively small number of smoke detectors containing ⁶³Ni sources have been distributed since the 1970s.

For smoke detectors containing ²⁴¹Am sources, NCRP 95 estimated a total collective dose of 7.8 person-Sv (780 person-rem) based on the results of Belanger et al. (NUREG/CR–1156). This estimate assumes (1) an annual distribution and disposal of 10 million smoke detectors containing 0.11 MBq (3 μ Ci) of ²⁴¹Am each and (2) an effective lifetime of 10 years for the smoke detectors during routine home usage. For these conditions, the collective dose was distributed as follows: manufacture and distribution—0.7 person-Sv (70 person-rem), use and maintenance—7 person-Sv (700 person-rem), and disposal—0.05 person-Sv (5 person-rem). The total collective dose estimated by O'Donnell et al. (1981) was 2.3 person-Sv (230 person-rem), or a factor of about 3 less than the value of 7.8 person-Sv (780 person-rem) adopted by the NCRP (1987) based on the results of Belanger et al. (1979). This difference resulted mostly from assumptions used in estimating individual and collective doses from direct external exposures to the smoke detectors during routine home usage. The individual and collective doses based on the external dose-rate calculations of O'Donnell et al. (1981) were judged to be the more reliable for the following reasons.

Belanger et al. assumed that the dose rates at 1 meter from a bare ²⁴¹Am foil source were 3.1 picocoulomb (pC)/kg-h (12 nanoroentgen (nR)/h), or 0.11 nSv/h (11 nrem/h), per 37 kBq (1 μ Ci) of ²⁴¹Am and the dose rates at 1 meter from an ²⁴¹Am foil source inside an assembled smoke detector were 74 pSv/h (7.4 nrem/h), per 37 kBq (1 μ Ci) of ²⁴¹Am. The latter value, used in their assessment of individual and collective doses from routine use of smoke detectors in homes, appears to have been based on the maximum measured value reported by manufacturers. It seems to provide an overly conservative estimate of the external dose from a smoke detector when compared with the calculated value used by O'Donnell et al. (1981). However, the value assumed for a bare ²⁴¹Am foil source is in good agreement with a value recently reported by Schmitt-Hannig et al. (1995). They reported that measurements on a bare ²⁴¹Am foil source with a 4 μ m gold window indicate a dose rate at 1 meter of 0.09 nSv/h (9 nrem/h) per 37 kBq (1 μ C) of ²⁴¹Am.

Schmitt-Hannig et al. also noted that calculated values that do not account for air or self-absorption give overly conservative dose rates from a bare ²⁴¹Am foil source. The ²⁴¹Am calculations of Unger and Trubey (1981, also see Table 2.1.2 of Section 2.1), for example, give unreasonably high dose rates because the calculations ignore the self-absorption of the very low-energy photons from ²⁴¹Am in a bare foil source (and an assembled smoke detector). Photons with energies of only 14, 26, 33, 59, and 69 keV are emitted in the radioactive decay of ²⁴¹Am with intensities of 42.7, 2.4, 0.11, 35.9, and 0.18%, respectively (Kocher, 1981).

O'Donnell et al. (1981) carefully modeled the foil source in an assembled smoke detector as a 5-mm-diameter by 0.002-mm-thick cylinder of a gold plus ²⁴¹Am matrix covered by a 0.0015-mm-thick gold window. The matrix contained 0.11 MBq (3 μ Ci) of ²⁴¹Am. The foils were also assumed to be enclosed in a 0.254-cm-thick iron housing within the assembled smoke detector. The dose equivalent rate at 1 meter from the assembled smoke detector was calculated to be 39 pSv/h (3.9 nrem/h), or 13 pSv/h (1.3 nrem/h) per 37 kBq (1 μ Ci) of ²⁴¹Am. This value was within the range of dose equivalent rates of 7 to 58 pSv/h (0.7 to 5.8 nrem/h) per 37 kBq (1 μ Ci) of ²⁴¹Am measured by manufacturers (based on information from the NRC's licensing files (NRC, Unpublished Reports, 1989)) and was a factor of 5.7 lower than the value of 74 pSv/h (7.4 nrem/h) per 37 kBq (1 μ Ci) of ²⁴¹Am assumed by Belanger et al. O'Donnell et al. also modeled arrays of detectors (cartons and pallets) during shipment as a composite material having an effective density of 1.4 g/cm³. This composite was a homogeneous mixture of ²⁴¹Am, plastic, cellulose, iron, carbon, and lead chosen to approximate the exposure rates from arrays of smoke detectors in cartons and on pallets during distribution and transport.

For chemical detectors containing ²⁴¹Am and ⁶³Ni, there are no known previously published analyses of the potential radiological impacts on members of the public. However, licensees would have supplied analyses to the NRC in their applications as required by 10 CFR 32.26.

2.15.4 Present Exemption Analysis for Smoke Detectors

Table 2.15.1 presents the results of potential radiation doses to the public from an assumed annual distribution of 10 million smoke detectors containing 37 kBq (1 μ Ci) of ²⁴¹Am each. The smoke detectors are assumed to have an effective lifetime of 10 years during routine home use. If external exposure is the only significant mode of exposure, this analysis will simply use a third of the estimates of effective dose equivalent (EDE) made by O'Donnell et al. (1981). This scales the results of their analysis for smoke detector activities of 0.11 MBq (3 μ Ci) to smoke detector activities of 37 kBq (1 μ Ci) used in the current analysis. If internal exposure is a significant mode of exposure, then 50-year committed EDEs are calculated here using dose conversion factors from Table 2.1.2 in Section 2.1 of this report. Internal exposure by inhalation or ingestion of ²⁴¹Am from smoke detectors may occur following accidents or misuse and disposal at landfill or incinerator sites.

2.15.4.1 Distribution and Transport

Based on the work of O'Donnell et al., the individual EDE could be 2×10^{-5} mSv (0.002 mrem) to stockhandlers who handle large numbers of cartons of packaged smoke detectors in wholesale warehouses and who work near the stored cartons. Individual doses would be less for other warehouse and truck terminal workers, store clerks and shoppers, truck drivers, and members of the public along truck routes. The total collective EDE from 1 year's distribution of 10 million smoke detectors containing 37 kBq (1 μ Ci) of ²⁴¹Am each could be 0.04 person-Sv (4 person-rem), due almost entirely to exposure to store customers.

2.15.4.2 Routine Use

Based on the work of O'Donnell et al., it is assumed that two ICSDs were installed in each home, 10% in bedrooms and 90% in hallways. Because the ICSDs may be installed in a home in a variety of different ways, the following representative exposure scenario was chosen to

indicate potential radiation doses from routine use of smoke detectors: one smoke detector installed in a hallway in the sleeping area of the home and the other installed in the master bedroom.

Table 2.15.2 lists the exposure conditions and radiation doses from use of two 37-kBq (1- μ Ci) ICSDs, one in the hallway in the sleeping area of the home and the other in the master bedroom. If the ICSDs have a 10-year lifetime, this set of detectors could deliver the estimated EDEs for 10 years. Purchase, installation, removal, and disposal occur only once during the 10-year lifetime of an ICSD, and the doses from these exposures were found to be insignificant compared to those from routine use (O'Donnell et al., 1981).

A resident who purchases, installs, and maintains two smoke detectors in the home, sleeps 8 h/day, and spends 4 h/day at other activities in the home could receive an annual EDE of 1×10^{-5} mSv (0.001 mrem). Essentially all of the dose is from the single detector located in the master bedroom. Other family members could receive annual EDEs of less than 1×10^{-5} mSv (<0.001 mrem).

The collective EDE from routine home use of 10 million ICSDs has been estimated based on the above doses for one detector in the hallway and one in the master bedroom. The collective EDE would total approximately 1 person-Sv (100 person-rem) for the 5 million homes (see Table 2.15.3).

Major commercial users of smoke detectors are motels and hotels (Bill, 1990). If a hotel maid is exposed to a smoke detector for 2000 h/yr at an average distance of 3 meters from a smoke detector unit, not necessarily the same unit, containing 37 kBq (1 μ Ci) of ²⁴¹Am, then the maximum individual annual EDE would be less than 1×10⁻⁵ mSv (<0.001 mrem).

2.15.4.3 Disposal

The potential individual and collective doses from disposal in landfills and by incineration are estimated using the generic methodology in Appendix A.2 of this report. In applying this methodology, it is assumed that the ²⁴¹Am sources remain intact during waste collection and burial at landfills. Thus, the pathway for inhalation is not operative in the case of off-site members of the public near landfills, and the pathways for ingestion or inhalation are not operative in the case of workers at landfills or waste collectors at both landfills and incinerators. The average activity of ²⁴¹Am in each detector is assumed to be 37 kBq (1 μ Ci) and the total activity of ²⁴¹Am in the 10 million smoke detectors disposed of annually is assumed to be 0.37 TBq (10 Ci).

For disposal at landfills, the annual EDE equivalent would be about 0.002 mSv (0.2 mrem) to waste collectors. The annual individual doses to workers at landfills, off-site members of the public near the landfills, and future on-site residents at landfills would be substantially less. The total collective dose to the public would be about 0.3 person-Sv (30 person-rem), due almost entirely to exposure to future on-site residents at landfills for 1000 years after loss of institutional controls over the sites. If exposure to future on-site residents is not considered, the collective EDE would be 0.008 person-Sv (0.8 person-rem), due mainly to exposure to waste collectors at landfills.

For disposal at incinerators, the annual EDE would be 0.01 mSv (1 mrem) to waste collectors. The individual doses to workers at incinerators and off-site members of the public are substantially less. The total collective dose to the public would be 0.005 person-Sv (0.5 person-rem), due mainly to exposure to off-site members of the public to airborne releases from incinerators.

2.15.4.4 Accidents and Misuse

In the case of accidents, the following is considered: (1) a residential fire involving 2 smoke detectors, (2) a transportation fire involving a typical shipment of 7,200 smoke detectors (O'Donnell et al., 1981), and (3) a manufacturer's warehouse fire involving 36,000 smoke detectors (i.e., five typical shipments). Also, a release factor of 0.01% is assumed for the ²⁴¹Am source foils inside the smoke detectors (NUREG/CR–0403; Hall and Hunt, NUREG/CP–0001; Niemeyer, 1969). Based on these assumptions and the generic accident methodology of Appendix A.1, the individual doses can be summarized as follows:

- For a person escaping from a residential fire or a neighborhood hero trying to rescue a person from a residential fire, the individual EDE from two smoke detectors containing a total of 74 kBq (2 μCi) of ²⁴¹Am could be 4×10⁻⁵ mSv (0.004 mrem).
- For a firefighter wearing a respirator at a residential fire, the individual EDE from two smoke detectors containing a total of 74 kBq (2 μCi) of ²⁴¹Am would be less than 1×10⁻⁵ mSv (<0.001 mrem)
- For a firefighter wearing a respirator at a transportation fire, the individual EDE from 7,200 smoke detectors containing a total of 266 MBq (7.2 mCi) could be 0.003 mSv (0.3 mrem).
- For a firefighter wearing a respirator at a warehouse fire, the individual EDE from 36,000 smoke detectors containing a total of 1.3 GBq (36 mCi) could be 0.003 mSv (0.3 mrem).

For the cleanup following a residential fire, the individual EDE could be about 5×10^{-5} mSv (0.005 mrem) for an individual who works for 8 hours and does not wear a respirator, and for fire inspectors who do not wear respirators and spend about 25% of their time inspecting mostly residential fires (62.5 working days), the annual individual EDE could be about 0.003 mSv (0.3 mrem) or approximately the same radiation dose as that to a firefighter who fights a single transportation or warehouse fire.

In the case of misuse, the following scenarios are considered: (1) exposure to a teacher who removes an ²⁴¹Am source from a smoke detector for use in classroom demonstrations about radioactivity, and (2) exposure to a person who removes an ²⁴¹Am source from a smoke detector and subsequently swallows it by accident. To estimate the potential radiation dose to the teacher, it is assumed that the teacher stores the ²⁴¹Am source in a convenient location in a classroom and is exposed at an average distance of about 1 meter from the source for 1000 h/yr. The EDE rate at 1 meter from a 37-kBq (1- μ Ci) source of ²⁴¹Am without regard for shielding by other materials is about 0.09 nSv/h (9 nrem/h) (see Section 2.15.3) and the annual EDE to the teacher could be 9×10⁻⁵ mSv (0.009 mrem). Also, the teacher handles the ²⁴¹Am source for 10 h/yr during classroom demonstrations at a nominal 1 cm distance (due to the size of the disk upon which the source is mounted) then the dose equivalent to a small area of skin

on the hand is about 0.3 mSv (30 mrem), based on a calculation using photon fluence and tissue (skin) energy absorption. Assuming a 10 cm² exposed skin area out of a total skin area of 1.8×10^4 cm² (ICRP 26) and a skin weighting factor of 0.01 (ICRP 60), the calculated EDE would be less than 1×10^{-5} mSv (< 0.001 mrem).

The inadvertent ingestion of an ²⁴¹Am smoke detector source by a consumer is not considered a realistic scenario, due to the source being contained within a protective housing and mounted on a larger metal or plastic disk. However, ingestion of an ²⁴¹Am smoke detector source has occurred during the manufacturing process. For this situation, the potential radiation to a person who swallows an ²⁴¹Am source is based on data from an actual case history reported by Rundo et al. (1977). From these data, it was determined that an ²⁴¹Am source foil lost 15.6% of its original activity in the GI tract. If the 15.6% is applied as the quantity available for uptake and the f₁ factor of 1×10⁻³ assumed by Eckerman et al. (EPA–520/1–88–020) is used, then the calculated individual EDE is about 6 mSv (600 mrem) to a person who swallows an ²⁴¹Am source foils in smoke detectors are inaccessible under normal conditions of distribution, use, and disposal, and a person would have to deliberately destroy a smoke detector to remove the source foil for other purposes. Thus, the removal of an ²⁴¹Am source from a smoke detector should be considered as a potential accident having a "negligible" probability of failure of the safety features (i.e., less than one failure per year per 1 million units distributed) (see Section 2.15.1).

2.15.4.5 Safety Criteria

Table 2.15.4 compares the results of the current dose assessment for smoke detectors containing ²⁴¹Am sources with the safety criteria for gas and aerosol detectors set forth in 10 CFR 32.27 (see Section 2.15.1). First, the limiting 10 CFR 32.27 values for dose to the whole body are compared with the maximum estimates of individual EDE, and second, the limiting 10 CFR 32.27 values for dose to other organs are compared with the maximum estimates of dose to red marrow (i.e., an active blood-forming organ) and bone surfaces (i.e., the endosteal cells on the bone surfaces). The dose equivalent to the red marrow of the skeleton and the endosteal cells on the bone surfaces of the skeleton are approximately 1.5 and 20 times greater, respectively, than the EDE to the whole body (EPA-520/1-88-020). The endosteal cells provide an important function as the bone-forming and bone-resorbing cells of the skeleton (Matthews, 1980). It is important to note that the maximum individual doses to the whole body (or EDE) and to the red marrow or bone surfaces (or organ doses) are less than the organ dose limits set forth in the safety criteria of 10 CFR 32.27 if, and only if, the swallowing of an²⁴¹Am source from a smoke detector is considered as a potential accident having a "negligible" incidence of failure of the safety features (i.e., less than 1 failure per year per 1 million exempt units distributed) (see Section 2.15.1).

2.15.5 Present Exemption Analysis for Chemical Detectors

Table 2.15.5 presents results of the current assessment of potential radiation dose to the public from chemical detectors containing either 0.37 GBq (10 mCi) of ⁶³Ni each or 5.9 MBq (0.16 mCi) of ²⁴¹Am each. As noted previously, chemical detectors are relatively new products whose total distribution is not accurately known at present and whose distribution will undoubtedly increase very rapidly in the future. Thus, it is assumed here that 10,000 sources containing a total of 59 GBq (1.6 Ci) of ²⁴¹Am and 10,000 sources containing a total of 3.7 TBg

(100 Ci) of ⁶³Ni could be distributed annually in these detectors, which have an effective lifetime of about 10 years.

In the case of ²⁴¹Am, the dose equivalent rate at 10 cm from an unshielded 5.9-MBq (0.16-Ci) source is about 0.15 μ Sv/h (150 μ rem/h) (see Section 2.15.2), and the dose equivalent rate at 10 cm from a partially shielded 5.9-MBq (0.16-mCi) source is about 300 nSv/h, (30 μ rem/h) according to the chemical director manufacturer. The manufacturer also determined that the dose equivalent rates at 30 cm and 100 cm from the partially shielded source in a chemical detector were about 40 nSv/h (4 μ rem/h) and 4 nSv/h (0.4 μ rem/h), respectively. In the case of ⁶³Ni, the dose equivalent rate from an unshielded 0.37-GBq (10-mCi) source due to bremsstrahlung is about 5 nSv/h (0.5 μ rem/h) at 30 cm and 0.5 nSv/h (0.05 μ rem/h) at 100 cm, using CONDOS II (Computer Codes, O'Donnell et al., 1981) and correcting for over estimation as discussed in Appendix A.4. For the current analysis it is assumed that partial shielding of the ⁶³Ni source by its protective housing, the external case of the detector, and various other detector components will reduce the dose equivalent rates due to the very low-energy bremsstrahlung from the ⁶³Ni source to essentially zero (0) for routine use and handling, distribution and transport, and disposal.

During normal use and handling of the chemical detectors, it is assumed that inhalation or ingestion of the ⁶³Ni and ²⁴¹Am does not occur, and the only significant mode of exposure is external irradiation of the body. Ingestion and inhalation may occur, however, during disposal of the chemical detectors or during accidents and misuse.

2.15.5.1 Distribution and Transport

To estimate the potential radiation doses to the public from the distribution and transport of chemical detectors containing sources of ²⁴¹Am, the generic methodology developed in Appendix A.3 is used. In applying this methodology, it is assumed that the chemical detectors are shipped primarily by a parcel-delivery service, and that a local driver in a large delivery truck picks up the chemical detectors and takes them to a local terminal for shipment to other local terminals for delivery to customers. It is assumed further that (1) each chemical detector passes through an average of four regional terminals before reaching its final destination, (2) the radiation exposure to workers in both local and regional terminals are the same as those estimated for workers in a large warehouse, and (3) the local parcel-delivery driver could pick up an average of 40 chemical detectors per day (250 day/yr) from the same manufacturer.

Based on these assumptions and the generic methodology in Appendix A.3 for the ²⁴¹Am sources, the annual individual EDE to the local parcel-delivery driver could be 0.02 mSv (2 mrem). Individual doses to other truck drivers, workers in truck terminals, and members of the public along truck routes would be significantly less. The collective EDEs could be 3×10^{-4} person-Sv (0.03 person-rem). These dose estimates are very conservative because the generic methodology does not account for the partial shielding of the ²⁴¹Am sources within the chemical detectors.

If partial shielding of the sources is considered (see Section 2.15.5), then the above individual and collective doses from 1 year's distribution of 10,000 chemical detectors containing 59 GBq (1.6 Ci) of ²⁴¹Am each would be a factor of 5 less, or about 0.004 mSv (0.4 mrem) and 6×10^{-5} person-Sv (0.006 person-rem), respectively.

2.15.5.2 Routine Use

To estimate potential radiation doses during routine use of the chemical detectors, it is assumed that 50% of the instruments are portable devices and the other 50% are fixed devices. In the case of the portable devices, it is assumed that an operator carries the instrument for 500 h/yr at an average distance of 30 cm from the body and that the operator is also exposed for 500 h/yr at an average distance of 1 meter while using and transporting the instrument. In case of the fixed devices, it is assumed that two individuals are exposed for 1000 h/yr at an average distance of the tax two individuals are exposed for 1000 h/yr at an average distance of 1 meter from each of the instruments.

For the portable instruments containing ²⁴¹Am sources (see Section 2.15.5), it is estimated that the annual individual EDE to the operator could be (4 μ rem/h) × (500 h/yr) + (0.4 μ rem/h) × (500 h/yr), or 0.02 mSv (2 mrem). The collective EDE to all operators over the 10-year effective lifetime of the 5000 portable instruments could be 1 person-Sv (100 person-rem). For the fixed instruments containing ²⁴¹Am sources, it is estimated that the annual individual EDE could be (0.4 μ rem/h) × (1000 h/yr), or 0.004 mSv (0.4 mrem). The collective EDE from exposure to the 5000 fixed instruments over the 10-year effective lifetime of the instruments could be 0.4 person-Sv (40 person-rem).

For both portable and fixed chemical detectors containing ²⁴¹Am sources, the estimated total collective EDE during routine use over the 10-year effective lifetime of the instruments is 1 person-Sv (100 person-rem).

2.15.5.3 Disposal

To estimate the potential radiation doses from landfill disposal and incineration of the ²⁴¹Am sources in chemical detectors, the analysis uses the generic disposal methodology in Appendix A.2 and the same assumptions as used for smoke detectors in Section 2.15.4.3. Sources are assumed to remain intact during waste collection (i.e., no ingestion or inhalation pathway).

For the disposal of 1 year's distribution of 10,000 chemical detectors containing a total of 59 GBq (1.6 Ci) of ²⁴¹Am, we can summarize the individual and collective doses as follows:

- At landfills, the annual EDE to a waste collector could be 3×10⁻⁴ mSv (0.03 mrem). Individual doses are less to workers at landfills and off-site members of the public. The total collective dose to all waste collectors, workers at landfills, and members of the public could be 0.05 person-Sv (5 person-rem), due almost entirely to exposure to future on-site residents for 1000 years after loss of institutional controls over the sites. If exposure to future on-site residents is not considered, the collective EDE would be 0.001 person-Sv (0.1 person-rem), due mainly to exposure to waste collectors at landfills.
- At incinerators, the annual EDE to a waste collector could be 0.002 mSv (0.2 mrem). Individual doses are significantly less to workers at incinerators and off-site members of the public. The collective EDE to all waste collectors, workers at incinerators, and members of the public could be 8×10⁻⁴ person-Sv (0.08 person-rem), due mainly to exposure to off-site members of the public to airborne releases from the incinerators.

For the disposal of chemical detectors containing ⁶³Ni, doses are essentially zero (0), since there is no photon component for ⁶³Ni decay and the sources are assumed to remain intact (i.e., no ingestion or inhalation pathway).

2.15.5.4 Accidents and Misuse

In the case of accidents, the following is considered: (1) a transportation fire involving a typical shipment of 40 chemical detectors (see Section 2.15.5.1), and (2) a manufacturer's warehouse fire involving 200 chemical detectors (i.e., five typical shipments). Also, a release factor of 0.01% is assumed for the ⁶³Ni and ²⁴¹Am sources inside the chemical detectors. Based on these assumptions and the generic accident methodology of Appendix A.1, the individual doses can be summarized as follows:

- For a firefighter wearing a respirator at a transportation fire, the individual EDE from 40 chemical detectors containing a total of 15 GBq (0.4 Ci) of ⁶³Ni would be less than 1×10⁻⁵ mSv (<0.001 mrem) and the individual EDE from 40 chemical detectors containing a total of 0.24 GBq (6.4 mCi) of ²⁴¹Am could be 0.003 mSv (0.3 mrem).
- For a worker who is involved in the cleanup following a transportation fire and does not wear a respirator, the individual EDE from the 40 chemical detectors containing ⁶³Ni could be 1×10⁻⁵ mSv (0.001 mrem), and the individual EDE from the 40 chemical detectors containing ²⁴¹Am could be 0.3 mSv (30 mrem).
- For a firefighter wearing a respirator at a warehouse fire, the individual EDE from 200 chemical detectors containing a total of 74 GBq (2 Ci) of ⁶³Ni would be less than 1×10⁻⁵ mSv (<0.001 mrem), and the individual EDE from 200 chemical detectors containing a total of 1.2 GBq (32 mCi) of ²⁴¹Am could be 0.002 mSv (0.2 mrem).
- For a worker who is involved in the cleanup following a warehouse fire and does not wear a respirator, the individual EDE from the 200 chemical detectors containing ⁶³Ni could be 6×10⁻⁵ mSv (0.006 mrem), and the individual EDE from the 200 chemical detectors containing ²⁴¹Am could be 0.2 mSv (20 mrem).

In the case of misuse, the exposure to a person who finds and vandalizes a chemical detector containing a 5.9-MBq (0.16-mCi) source of ²⁴¹Am is considered. It is assumed that the person deliberately destroys the chemical detector out of curiosity and handles the ²⁴¹Am source for 30 minutes before discarding both the source and other detector parts. Assuming that the hand exposure is at a nominal 1 cm distance (due to the size of the disk upon which the source is mounted) then the dose equivalent to a small area of skin on the hand is about 2 mSv (200 mrem), based on a calculation using photon fluence and tissue (skin) energy absorption. Assuming a 10 cm² exposed skin area out of a total skin area of 1.8×10⁴ cm² (ICRP 26) and a skin weighting factor of 0.01 (ICRP 60), the calculated EDE would be 1×10⁻⁵ mSv (0.001 mrem). For potential ingestion of material due to the handling of a 0.37-GBq (10-mCi) ⁶³Ni source or a 5.9-MBq (0.16-mCi) ²⁴¹Am source, the generic accident methodology developed in Appendix A.2 is applied for spills of radioactive material in the form of a powder. It is assumed, first, that 10% of the material on the source is deposited on the body and, second, that 0.1% of this deposited material is ingested before it is removed from the body by washing. Thus, the EDE could be 0.006 mSv (0.6 mrem) from the ingestion of ⁶³Ni and 0.6 mSv (60 mrem) from ingestion of ²⁴¹Am.

2.15.5.5 Safety Criteria

Tables 2.15.6 and 2.15.7 compare results of our present dose assessment for chemical detectors containing ⁶³Ni and ²⁴¹Am sources with the safety criteria for gas and aerosol detectors set forth in 10 CFR 32.27 (see Sect. 2.15.1). The comparisons for chemical detectors containing ²⁴¹Am sources are similar to those for smoke detectors in Table 2.15.4. The comparisons for chemical detectors containing ⁶³Ni sources are selected to reflect the different metabolic behavior of ⁶³Ni in the body. In the case of ⁶³Ni, the dose equivalents to all organs of the body from inhalation are essentially the same as the EDE, but the dose equivalent to the colon (i.e., large intestines) from ingestion is about 4 times greater than either the dose equivalents to all other organs of the body or EDE (Private communication with Keith F. Eckerman, Oak Ridge National Laboratory, Oak Ridge, TN, July 1997). It is important to note from comparisons in Tables 2.15.6 and 2.15.7 that the maximum estimates of individual dose to the whole body (or EDE) and dose equivalent to various organs of the body are less than the dose limits set forth in 10 CFR 32.27. In the case of ²⁴¹Am, however, the estimate of dose equivalent to bone surfaces of a person who vandalizes a chemical detector and ingests radioactive material due to contamination of the hands is very close to the dose limit for a potential accident having a "low" incidence of failure of the safety features (i.e., less than one failure per year per 10,000 exempt units distributed) (see Section 2.15.1).

2.15.6 Summary

The results of the current assessment of potential radiation doses to the public from smoke detectors containing ²⁴¹Am are shown in Table 2.15.1. These results are based on a 10-year effective lifetime for the smoke detectors and an annual distribution of 10 million smoke detectors containing 37 kBq (1 μ Ci) of ²⁴¹Am each. In Table 2.15.4 the results from the current assessment for smoke detectors are compared further with the safety criteria for gas and aerosol detectors. These comparisons show that the maximum estimates of individual doses for the current analysis are within the applicable dose limits set forth in the safety criteria of 10 CFR 32.27.

An annual collective dose of 2 person-Sv (200 person-rem) from smoke detectors containing ²⁴¹Am was estimated previously by O'Donnell et al. (1981). NCRP 95 estimated a collective dose from smoke detectors containing an ²⁴¹Am source of 7.8 person-Sv (780 person-rem) based on the results of Belanger et al. (NUREG/CR–1156). These differences can be attributed mostly to the different assumptions used in estimating both individual and collective doses from exposure to smoke detectors during routine use in homes (see Section 2.15.3).

The collective dose estimated in this study for smoke detectors containing ²⁴¹Am sources is 1 person-Sv (100 person-rem), or about half that estimated in the previous study by O'Donnell et al. (1981). The difference can be attributed mostly to a decrease in the average ²⁴¹Am content of a smoke detector from about 0.11 MBq (3 μ Ci) in the early 1980s to about 37 kBq (1 μ Ci) at present (see Section 2.15.2.1).

Also presented are the results from a completely new assessment of the potential radiation doses to the public for an assumed annual distribution of 10,000 chemical detectors each containing 0.37 GBq (10 mCi) of ⁶³Ni each and 10,000 chemical detectors each containing 5.9 MBq (0.16 mCi) of ²⁴¹Am each (see Table 2.15.5). The results of this assessment, which

are based on an effective lifetime of 10 years for the chemical detector units, suggest a total collective dose of about 2 person-Sv (200 person-rem).

The results of the completely new assessment of potential radiation dose to the public from chemical detectors containing ²⁴¹Am or ⁶³Ni sources are also compared with the safety criteria for gas and aerosol detectors in Tables 2.15.6 and 2.15.7. These comparisons show that the maximum estimates (for the new assessment) of individual dose to members of the public from chemical detectors containing ⁶³Ni and ²⁴¹Am sources are less than the applicable dose limits set forth in the safety criteria of 10 CFR 30.27.
Exposure Pathway	Individual Annual Effective Dose Equivalent (mrem) ^a	Collective Effective Dose Equivalent ^b (person-rem) ^a
Distribution and transport	0.002°	4
Routine use	0.002 ^d	100
Disposal in ordinary trash Landfills Incinerators	0.2° 1 [†]	30 0.5
<u>Accidents and misuse</u> Classroom use of source Fire Swallowing of source	0.009 ⁹ 0.3 ^h	

Table 2.15.1 Potential Radiation Doses From Smoke Detectors Containing ²⁴¹Am

^a 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

^b Collective doses are based on an assumed annual distribution of 10 million smoke detectors containing 37 kBq (1 μ Ci) of ²⁴¹Am each and having a 10-year effective lifetime.

^c Dose estimate applies to stockhandlers exposed to multiple exempt units in wholesale warehouses during distribution and transport; dose estimates are less for other warehouse and truck terminal workers, truck drivers, store clerks and customers, and members of the public along truck routes (see Section 2.15.4.1).

^d Dose estimate applies to residents who are exposed to two smoke detectors during routine home use for 12-24 hours per day; dose estimates are less for family members who are exposed for less than 12 hours per day (see Section 2.15.4.2).

^e Dose estimate applies to waste collectors exposed to multiple smoke detectors during disposal at landfills; dose estimates are significantly less for workers at landfills, off-site members of the public, and future on-site residents are significantly less (see Section 2.15.4.3).
 ^f Dose estimate applies to waste collectors exposed to multiple smoke detectors during disposal at incinerators; dose estimates are significantly less for other workers at incinerators and off-site members of public (see Section 2.15.4.3).

⁹ Dose estimate applies to a teacher who removes the ²⁴¹Am source from a smoke detector unit and stores the unshielded source in a classroom (see Section 2.15.4.4).

^h Dose estimate applies to a firefighter who is exposed to multiple smoke detector units during a transportation or warehouse fire (see Section 2.15.4.4).

ⁱ Dose estimate applies to a person who removes a source from a smoke detector and accidentally swallows it (see Section 2.15.4.4).

Exposed Person	Exposure Activity	Sources	Duration of Exposure (h/yr)	Distance From Source (m)	Individual Dose Over First Year (mrem) ^b
Resident	Purchase	2 detectors	0.5	0.3	<0.001
	Transport	2 detectors	0.5	1	<0.001
	Install	2 detectors	0.5	1	<0.001
	Maintain	2 detectors	2	1	<0.001
	Sleep	1 detector	2,920	2	0.001
		1 detector	2,920	6	<0.001
	Other	2 detectors	1,460	6	<0.001
	Total:				0.001
Other family	Sleep	2 detectors	2,920	6	<0.001
members	Other	2 detectors	1,460	6	<0.001
// -=	Total:		····	****	<0.001

Table 2.15.2 Exposure Conditions and Radiation Doses From Routine Home Useof Two Smoke Detectors Containing 0.037 MBq (1 μ Ci) of ²⁴¹Am Each ^a

^a Adapted from O'Donnell et al. (1981).
^b 1 mrem = 0.01 mSv.

Table 2.15.3 Potential Radiation Doses From Routine Home Use of 10 MillionSmoke Detectors Containing 0.037 MBq (1 μ Ci) Each of ²⁴¹Am ^a

Exposed Person	Individual Dose Over First Year (mrem) ⁶	Individual Dose Over 10 Years (mrem) ^b	Number of Exposed Persons	Collective Dose Over 10 Years (person-rem) ^b
Resident	0.001	0.01	10,000,000	100

^a Scenario is for one detector in a hallway in the sleeping area of a home and one detector in the master bedroom.

^b 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

.

Table 2.15.4 Comparison of Estimated Maximum Individual Doses From SmokeDetectors Containing 37 kBq (1 μCi) of 241 Am and Limiting Organ DosesFrom 10 CFR 32.27

Exposure Conditions ^a	Maximum Individual Dose ^b (mrem, rem, or mrem/yr) ^c	Regulatory Limit for Organ Dose (mrem, rem, or mrem/yr)°
Effective dose equivalent		
Routine use and disposal of a single exempt unit	0.001mrem ^d	5 mrem/yr
Accidents involving a single exempt unit	600 mrem ^e	0.5 rem ^f 15 mrem ^g
Normal handling and storage of multiple exempt units	0.002 mrem/yr ^h	5 mrem/yr
Accidents involving multiple exempt units	0.3 mrem ⁱ	0.5 rem ^f 15 mrem ^g
Dose equivalent to red marrow		
Accidents involving a single exempt unit	800 mrem ⁱ	0.5 rem ^f 15 rem ^g
Accidents involving multiple exempt units	0.5 mrem ^k	0.5 mrem ^t 15 rem ^g
Dose equivalent to bone surfaces		
Accidents involving a single exempt unit	10 rem ⁱ	1.5 rem ^l 50 rem ^m
Accidents involving multiple exempt units	6 mrem ^k	1.5 rem ⁱ 50 rem ^m

See the following page for footnotes.

Footnotes to Table 2.15.4

^a See Section 2.15.4 of this report.

^b Maximum individual dose based on the typical quantity used (i.e., 37 kBg (1 μ Ci) of ²⁴¹Am). However, the activity of the source could be increased until the regulatory limit for organ dose is reached.

[°] 1 mrem = 0.01 mSv; 1rem = 0.01 Sv.

^d Dose estimate applies to spouse of a resident who is exposed to a single smoke detector during routine home use (see Section 2.15.4.2).

^e Dose estimate applies to a person who removes an ²⁴¹Am source from a smoke detector and subsequently swallows it (see Section 2.15.4.4).

^f Limit on whole-body dose and dose to active blood-forming organs when probability of failure of safety features in product is less than one failure per year for each 10,000 exempt units distributed (see Section 2.15.1).

⁹ Limit on whole-body dose and dose to active blood-forming organs when probability of failure of safety features in product is less than one failure per year for each 1 million exempt units distributed (see Section 2.15.1).

^h Dose estimate applies to stockhandlers exposed to multiple smoke detectors at wholesale warehouses (see Section 2.15.4.1).

ⁱ Dose estimate applies to a firefighter who is exposed to radioactive material from multiple smoke detector units at a transportation or warehouse fire (see Section 2.15.4.4).

^j Bone surface dose or red marrow dose to a person who removes an ²⁴¹Am source from a smoke detector and subsequently swallows it (see Section 2.15.4.4).

^k Bone surface dose or red marrow dose to a firefighter exposed to radioactive material from multiple smoke detector units at a transportation or warehouse fire (see Section 2.15.4.5).

¹ Limit on "other organ" doses when probability of failure of safety features in product is less than one failure per year for each 10,000 units distributed (see Section 2.15.1).

^m Limit on "other organ" doses when probability of failure of safety features in product is less than one failure per year for each 1 million exempt units distributed (see Section 2.15.1).

Exposure Pathway	Individual Annual Effective Dose Equivalent (mrem)ª	Collective Effective Dose Equivalent ^b (person-rem) ^a
Distribution and transport	0.4 ^c	0.006
Routine use	2 ^d	100
<u>Disposal</u> Landfills Incinerators	0.03 ^e 0.2 ^f	5 0.08
Accidents and misuse Fire Vandalism of detector unit	30 ⁹ 60 ^h	

Table 2.15.5 Potential Radiation Doses From Chemical Detectors Containing ²⁴¹Am

^a 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

^b Collective doses are based on (1) an assumed annual distribution of 10,000 chemical detectors containing 5.9 MBq (0.16 mCi) of ²⁴¹Am each and (2) an assumed 10-year effective lifetime for the detectors.

^c Dose estimate applies to local parcel-delivery drivers exposed to multiple exempt units in wholesale warehouses during distribution and transport; dose estimates are significantly less for other truck drivers, workers in truck terminals, and members of public along truck routes (see Section 2.15.5.1).

^d Dose estimate applies to a user of a portable chemical detector containing an ²⁴¹Am source; dose estimates are less for a user of a fixed chemical detector containing an ²⁴¹Am source (see Section 2.15.5.2).

^e Dose estimate applies to waste collectors exposed to multiple chemical detectors during disposal at landfills; dose estimates are significantly less for workers at landfills, off-site members of the public, and future on-site residents (see Section 2.15.5.3).

^f Dose estimate applies to waste collectors exposed to multiple chemical detectors during disposal at incinerators; dose estimates are significantly less for other workers at incinerators and off-site members of the public (see Section 2.15.5.3).

⁹ Dose estimate applies to a worker who are involved in the cleanup following the transportation fire involving multiple chemical detectors with ²⁴¹Am sources (see Section 2.15.5.4).

^h Dose estimate applies to ingestion of ²⁴¹Am by a person who finds and destroys a chemical detector containing an ²⁴¹Am source, (see Section 2.15.4.4).

Table 2.15.6 Comparison of Estimated Maximum Individual DosesFrom Chemical Detectors Containing 5.9 MBq (0.16 mCi) of 241 Amand Limiting Organ Doses From 10 CFR 32.27

Exposure Conditions ^a	Maximum Individual Dose ^b (mrem, rem, or mrem/yr) ^c	Regulatory Limit for Organ Dose (mrem, rem, or mrem/yr) ^c
Effective dose equivalent		
Routine use and disposal of a single exempt unit	2 mrem/yr ^d	5 mrem/yr ^e
Accidents involving a single exempt unit	60 mrem ^f	0.5 rem ^g 15 rem ^h
Normal handling and storage of multiple exempt units	0.4 mrem/yr ⁱ	5 mrem/yr ⁱ
Accidents involving multiple exempt units	30 mrem ^k	0.5 rem ⁹ 15 rem ^h
Dose equivalent to red marrow		
Accidents involving a single exempt unit	90 mrem ⁱ	0.5 rem ^g 15 rem ^h
Accidents involving multiple exempt units	40 mrem ^m	0.5 mrem ⁹ 15 rem ^h
Dose equivalent to bone surfaces		
Accidents involving a single exempt unit	1 rem ⁱ	1.5 rem ⁿ 50 rem ^o
Accidents involving multiple exempt units	500 mrem ^m	1.5 rem ⁿ 50 remº

See following page for footnotes.

...

Footnotes to Table 2.15.6

^a See Section 2.15.4 of this report.

^b Maximum individual dose based on the typical quantity used (i.e., 5.9 mBq (160 mCi) of ²⁴¹Am). However, the activity of the source could be increased until the regulatory limit for organ dose is reached.

^c 1 mrem = 0.01 mSv; 1 rem = 0.01 Sv.

^d Dose estimate applies to an operator who is exposed to a single chemical detector unit during routine use (see Section 2.15.5.2).

^e Limit on whole-body dose and active blood-forming organs during normal use and handling of a single exempt unit (see Section 2.15.1).

^f Dose estimate applies to a person who vandalizes a chemical detector unit and ingests radioactive material due to contamination of hands (see Section 2.15.5.4).

⁹ Limit on whole-body dose and dose to active blood-forming organs when probability of failure of safety features in product is less than one failure per year for each 10,000 exempt units distributed (see Section 2.15.1).

^h Limit on whole-body dose and dose to active blood-forming organs when probability of failure of safety features in product is less than one failure per year for each 1 million exempt units distributed (see Section 2.15.1).

¹ Dose estimate applies to local parcel-delivery driver who is exposed to multiple chemical detector units during distribution and transport (see Section 2.15.5.1).

¹ Limit on whole-body dose and dose to active blood-forming organs during normal handling and storage of multiple exempt units (see Section 2.15.1).

^k Dose estimate applies to a worker who inhales radioactive material from multiple chemical detector units during cleanup after a transportation fire (see Section 2.15.5.4).

¹ Bone surface dose or red marrow dose to a person who vandalizes a chemical detector unit and ingests radioactive material due to contamination of hands (see Section 2.15.5.4)

^m Bone surface dose or red marrow dose to worker who inhales radioactive material from multiple chemical detector units during cleanup after a transportation fire (see Section 2.15.5.4). ⁿ Limit on "other organ" doses when probability of failure of safety features of the product is less than one failure per year for each 10,000 exempted units distributed (see Section 2.15.1).

^o Limit on "other organ" doses when probability of failure of safety features of the product is less than one failure per year for each 1 million exempt units distributed (see Section 2.15.1).

Table 2.15.7 Comparison of Estimated Maximum Individual DosesFrom Chemical Detectors Containing 0.37 GBq (10 mCi) of 63Niand Limiting Organ Doses From 10 CFR 32.27

Exposure Conditions ^a	Maximum Individual Dose ^b (mrem or μ rem) ^c	Regulatory Limit for Organ Dose (mrem/yr or rem) ^c
Effective dose equivalent		
Routine use and disposal of a single exempt unit	Oq	5 mrem/yr ^e
Accidents involving a single exempt unit	0.6 mrem ^f	0.5 rem ⁹ 15 rem ^h
Normal handling and storage of multiple exempt units	Oʻ	5 mrem/yr ⁱ
Accidents involving multiple exempt units	0.01 mrem ^k	0.5 rem ^f 15 rem ^g
Dose equivalent to critical organ		
Accidents involving a single exempt unit	3 mrem ^m	1.5 rem ⁿ 50 rem ^o
Accidents involving multiple exempt units	14 µrem⁰	1.5 rem ⁿ 50 rem°

See following page for footnotes.

.

Footnotes to Table 2.15.7

^a See Section 2.15.5 of this report.

^b Maximum individual dose based on the typical quantity used (i.e., 0.37 GBq (10 mCi) of ⁶³Ni). However, the activity of the source could be increased until the regulatory limit for organ dose is reached.

^c 1 mrem = 0.01 mSv; 1 rem = 0.01 Sv.

^d Dose estimate applies to an operator who is exposed to a single chemical detector unit during routine use. As discussed in Section 2.15.4.2, EDE during normal use and handling is essentially zero (0).

^e Limit on whole-body dose during normal use and disposal of a single exempt unit (see Section 2.15.1).

^f Dose estimate applies to a person who vandalizes a chemical detector unit and ingests source material due to contamination of hands (see Section 2.15.5.4).

⁹ Limit on whole-body dose when probability of failure of safety features in product is less than one failure per year for each 10,000 exempt units distributed (see Section 2.15.1).

^h Limit on whole-body dose when probability of failure of safety features in product is less than one failure per year for each 1 million exempt units distributed (see Section 2.15.1).

¹ Dose estimate applies to a local parcel-delivery driver exposed to multiple chemical detector units during distribution and transport. As discussed in Section 2.15.5.1, dose during routine use and handling is essentially zero (0).

Limit on whole-body dose during normal handling and storage of multiple exempt units (see Section 2.15.1).

^k Dose estimate applies to a worker who inhales radioactive material from multiple chemical detector units during cleanup after a transportation fire (see Section 2.15.5.4).

'The colon is the critical organ for accident involving a single unit and the lung is the critical organ for an accident involving multiple units.

^m Colon dose to a person who vandalizes a chemical detector and ingests radioactive material due to contamination of hands (see Section 2.15.5.4).

ⁿ Limit on "other organ" doses when probability of failure of safety features in product is less than one failure per year for each 10,000 exempt units distributed (see Section 2.15.1).

^o Limit on "other organ" doses when probability of failure of safety features in product is less than one failure per year for each one million exempt units distributed (see Section 2.15.1). ^p Colon dose to a worker who inhales radioactive material from multiple exempt units during cleanup after a transportation fire (see Section 2.15.5.4).

3 EXEMPTIONS FOR SOURCE MATERIAL

3.1 Properties of Source Material

3.1.1 Introduction

This subsection provides an introduction to the following subsections of Section 3, that present the results of assessments of radiological impacts on the public from products or materials containing exempted amounts of source material, i.e., uranium or thorium. Information presented in this section includes the definition of source material (see Section 3.1.2), radioactive decay data for the naturally occurring isotopes of uranium and thorium and their radioactive decay products (see Section 3.1.3), and dosimetry data used in estimating dose from external and internal exposure for the radionuclides of concern in source material (see Section 3.1.4).

3.1.2 Definition of Source Material

As defined in 10 CFR 40.4 and used in this report, the term "source material" means:

(1) Uranium or thorium, or any combination thereof, in any physical or chemical form or
(2) ores which contain by weight one-twentieth of one percent (0.05%) or more of:
(i) Uranium, (ii) thorium, or (iii) any combination thereof. Source material does not include special nuclear material.

The term "special nuclear material" is defined in Section 2.1.2 of this report. In essence, source material is the raw material from which nuclear fuel is made, and it includes uranium or thorium in their natural isotopic abundances.

In most of the exempted products or materials containing source material discussed in the following subsections of Section 3, the uranium or thorium is initially in a chemically separated form. However, both uranium and thorium decay to short-lived radionuclides that are important in estimating dose at times shortly after chemical separation of these elements. The short-lived decay products are included in all dose assessments for chemically separated materials.

In addition, in a few cases (e.g., the exemption for unrefined and unprocessed ore containing source material discussed in Section 3.3), the exempted materials contain uranium or thorium that has not been chemically separated. In these cases, all decay products would be present, including the long-lived decay products of uranium when this is the source material of concern. However, the regulatory jurisdiction over the decay products can vary depending on the particular situation. From the definition of source material given above, any ore containing 0.05% or more by weight of uranium and thorium is source material in its entirety, whereas only the uranium and thorium are source material in other materials containing uranium or thorium.

Other naturally occurring radioactive materials that would be present, including the decay products of uranium and thorium, normally fall under the jurisdiction of the Environmental Protection Agency and the States. However, particularly when the decay products are present in connection with an activity involving source material that is licensed by the Nuclear Regulatory Commission (NRC). Therefore, in the assessments for materials containing less

nonchemically separated source material other than ore, doses resulting from exposure to the source material and its radiologically significant decay products are identified separately. Information on the total dose from exposure to the source material and its decay products also is included.

3.1.3 Decay Data for Source Materials

This section presents radioactive decay data that are important in estimating dose from exposure to uranium or thorium and their radioactive decay products.

First, Table 3.1.1 lists the mass and activity abundances of chemically separated uranium and thorium. For uranium, these data are given for naturally occurring and depleted uranium, because both types of uranium are used in exempted products or materials. For naturally occurring uranium, the activity abundances of the different isotopes are based on their known mass abundances and half-lives and the assumption that ²³⁸U and ²³⁴U occur in equal activity abundances. As defined in 10 CFR 40.4, depleted uranium is any source material uranium in which the mass abundance of ²³⁵U is less than the value for naturally occurring uranium. Depleted uranium also has a lower mass abundance of ²³⁴U. The abundances of the different isotopes in depleted uranium are somewhat variable. The values in Table 3.1.1 are intended to be representative of byproduct residues from uranium enrichment, i.e., the production of material from natural uranium with a mass abundance of ²³⁵U of about 3% for use as special nuclear material, because this has been the most important source of depleted uranium used in exempted products or materials.

Thorium at the time of chemical separation is assumed to consist of equal activity abundances of ²³²Th and ²²⁸Th, and the ²²⁸Th has a negligible mass abundance due to its short half-life. Justification for not considering other isotopes of thorium, principally ²³⁰Th, that also may be present in chemically separated thorium is discussed in Section 3.1.5.

Then, radioactive decay data for the naturally occurring isotopes of uranium and thorium and their radioactive decay products are given in Tables 3.1.2 and 3.1.3. These data include:

- The half-life of each isotope of uranium or thorium.
- The specific activity of each isotope of uranium or thorium, which is defined as the activity per unit mass.
- All decay products of the isotopes of uranium or thorium that are produced in at least 0.1% of the decays of the parent radionuclide, their half-lives, and the branching fraction in the decay of the parent radionuclide.
- Identification of the principal decay modes for each radionuclide (i.e., alpha, beta, or beta and gamma).

Table 3.1.2 also identifies the short-lived radioactive decay products of the uranium isotopes that would be present in significant quantities within a few days or months after chemical separation. All other decay products would be present in significant quantities in chemically separated uranium only after very long times (see Table 3.1.4). For thorium, all radioactive

decay products listed in Table 3.1.3 would be present in significant quantities within a few days or years after chemical separation (see Table 3.1.5). When uranium and thorium have not been chemically separated, all decay products presumably would be present and in activity equilibrium.

Finally, data on the activity of the longer lived radioactive decay products of uranium and thorium relative to the initial activity of the parent radionuclide as a function of time after chemical separation are given in Tables 3.1.4 and 3.1.5. Data for the decay chains of uranium in Table 3.1.4 are shown out to 1 million years only to indicate the very long times required for the decay products to reach activity equilibrium with the parent radionuclides in chemically separated natural and depleted uranium. The long-lived decay products of uranium generally are unimportant in chemically separated uranium at times less than 100 years. The shorter lived decay products listed in Table 3.1.2 or 3.1.3 that are not shown in Table 3.1.4 or 3.1.5 would always be in activity equilibrium with their immediate longer lived precursors. Decay chains of the isotopes of uranium and thorium are described in the following paragraphs.

For chemically separated uranium, each naturally occurring isotope has a long-lived decay product that occurs relatively early in the decay chain (²³⁴U in the decay of ²³⁸U, ²³¹Pa in the decay of ²³⁵U, and ²³⁰Th in the decay of ²³⁴U). Therefore, only the activities of the immediate short-lived decay products identified in Table 3.1.2 (²³⁴Th, ^{234m}Pa, and ²³⁴Pa in the decay of ²³⁸U and ²³¹Th in the decay of ²³⁵U) would be important during normal distribution, use, and disposal of exempted products or materials containing chemically separated uranium. Because of the very long time periods required for significant buildup of activity for the long-lived decay products of the naturally occurring isotopes of uranium, as shown in Table 3.1.4, these decay products and their short-lived decay products generally can be neglected in dose assessments for exempted products or materials containing chemically separated uranium. The long-lived decay products of uranium and their short-lived decay products may need to be considered in dose assessments for exempted products or materials containing chemically separated uranium that has not been chemically separated, because all decay products would be in activity equilibrium with the parent isotopes of uranium. However, as indicated in Section 3.1.2, the regulatory jurisdiction over the decay products can vary depending on the particular situation.

The decay chains of ²³²Th and ²²⁸Th, which are assumed to occur in equal activities in chemically separated uranium, include only relatively short-lived radionuclides. Therefore, in contrast to the situation for uranium described above, the contributions from all decay products must be taken into account in dose assessments for exempted products or materials containing thorium, regardless of whether the thorium has been chemically separated. For chemically separated thorium that initially contains equal activities of ²³²Th and ²²⁸Th, the data in Table 3.1.5 show that the activity of ²²⁸Th and its short-lived decay products decreases with time for the first few years, due to the decay of the initial activity of ²²⁸Th. However, the activity of ²²⁸Th then increases with time after a few years, due to its buildup in the decay of ²³²Th, until activity equilibrium in the decay chain is achieved within 50 years.

Dose assessments presented in this report for distribution and use of exempted products or materials containing thorium presented in this report usually assume that the thorium has been aged for 20 years after chemical separation. At this time, the data in Table 3.1.5 show that the activity of all decay products of thorium has achieved about 90% of activity equilibrium. The assumption that the thorium has been aged for 20 years provides conservative estimates of dose for times less than 20 years, and it does not significantly underestimate doses for times

beyond 20 years. For thorium that has not been chemically separated, all decay products would be in activity equilibrium with the ²³²Th. Dose assessments for disposal operations also assume that the thorium has been aged for 20 years, but the assessments of dose after disposal in landfills assume that the decay products are in activity equilibrium with the ²³²Th.

3.1.4 Dosimetry Data for Source Materials

Data that can be used to estimate external and internal doses from exposure to uranium or thorium and their decay products are listed in Tables 3.1.6 and 3.1.7. These data include:

- The specific gamma-ray dose constant, which is defined as the dose-equivalent rate per unit activity at a distance of 1 meter from an unshielded point source in air and which is a reasonable approximation to the effective dose equivalent (EDE) rate per unit activity for radionuclides that emit high-energy photons.
- The external dose coefficient for submersion in an atmospheric cloud, which is defined as the external EDE rate per unit concentration in a uniformly contaminated, semiinfinite volume of air. The skin dose component has been included using a 0.01 weighting factor.
- The internal dose coefficient for ingestion, which is defined as the 50-year committed EDE per unit activity intake by ingestion.
- The internal dose coefficient for inhalation, which is defined as the 50-year committed EDE per unit activity intake by inhalation.

The specific gamma-ray dose constant and external dose coefficient for air submersion are listed in Tables 3.1.6 and 3.1.7 only if a radionuclide emits photons of sufficient energy and intensity that external exposure could be of concern in assessing dose from exposure to uranium or thorium and its decay products. Thus, these data are not listed for radionuclides that are not photon emitters or that emit only very low-energy photons. Similarly, the dose coefficients for ingestion or inhalation are listed for a decay product only if the dose from internal exposure would be significant compared with the dose from other radionuclides that would be present in chemically separated or unseparated materials. As noted in Table 3.1.2, only the ²³⁴Th, ^{234m}Pa, and ²³⁴Pa decay products of ²³⁸U and the ²³¹Th decay product of ²³⁵U would be present in chemically separated uranium. The dosimetry data for each decay product take into account the branching fraction in the decay of the parent radionuclide given in Table 3.1.2 or 3.1.3.

For some radionuclides, dose coefficients for ingestion are listed for more than one value of the gastrointestinal-tract absorption fraction. Dose coefficients for inhalation are listed for more than one lung clearance class. Absent specific information on the chemical form of uranium or thorium and their decay products in a particular product or material, the largest dose coefficient for ingestion or inhalation generally is used in assessing dose. If the chemical form of a radionuclide in a particular product or material is known, the appropriate dose coefficients for ingestion and inhalation can be selected based on the assignments given in Table 3.1.8.

For ²²²Rn and ²²⁰Rn, the dose coefficient for inhalation is the EDE rate from inhalation of radon and its short-lived decay products per unit concentration of radon in air. In each case, values are given for both indoor and outdoor exposures. The ²¹⁹Rn decay product of ²³⁵U is sufficiently short-lived that inhalation exposure generally is not of concern for this radionuclide. The inhalation dose coefficients for ²²²Rn and ²²⁰Rn assume that their short-lived decay products are in activity equilibrium in air. Thus, these values generally are conservative for both indoor and outdoor exposures, because they do not consider the effects of ventilation and deposition. Recommended corrections to the dose coefficients for radon to account for the extent of activity equilibrium of the decay products in air also are discussed in Tables 3.1.6 and 3.1.7.

The dosimetry data in Tables 3.1.6 and 3.1.7 are used, when appropriate, in the dose assessments for source material in the remainder of Section 3. The dose coefficients for ingestion and inhalation are used in all assessments of internal exposure. Because the dose coefficients for ingestion or inhalation are nearly the same for ²³⁸U, ²³⁵U, and ²³⁴U, the dose from internal exposure to chemically separated uranium does not depend significantly on whether the uranium is naturally occurring or depleted or on the particular mass abundances of the different isotopes in depleted uranium. The external dose coefficient for air submersion also is used whenever this exposure pathway is considered. The external dose coefficients provide conservative estimates of dose for submersion in a finite atmospheric cloud.

However, the specific gamma-ray dose constant, which provides an indication of the potential importance of external exposure, is used to estimate external dose only when exposure to an unshielded point source is an appropriate assumption, i.e., when the dimensions of the source are small compared with the distance between the source and receptor locations, and the radionuclides of concern emit photons with energies above about 100 keV. For finite sources that cannot be represented as a point and for radionuclides that emit only lower energy photons (e.g., ²³⁸U, ²³⁴U, and ²³²Th), external dose rates normally are calculated using the CONDOS-II (Computer Codes, O'Donnell et al., 1981) or MicroShield (Computer Codes, Grove Engineering, 1996) computer codes for the appropriate source geometry and amount of shielding between the source and receptor locations.

In addition, even for small sources that can be represented as a point, the external dose from ²³⁸U, ²³⁴U, and ²³²Th generally can be neglected because, for these low-energy photon emitters (Kocher, 1981), the self-shielding provided by the uranium or thorium in the source would reduce the dose rate by a large factor compared with the values given in Tables 3.1.6 and 3.1.7 (Unger and Trubey, 1981). Therefore, external exposure to ²³⁸U, ²³⁴U, and ²³²Th is well approximated by assuming that the dose is due only to any photon-emitting decay products that would be present.

3.1.5 Isotopic Composition of Chemically Separated Thorium

As discussed in Section 3.1.3 and summarized in Table 3.1.1, all assessments in this report assume that thorium consists of equal activity abundances of ²³²Th and ²²⁸Th at the time of chemical separation and that no other isotopes of thorium are present. However, thorium-bearing ores also contain varying amounts of uranium (Albert, 1966), and the thorium decay products of the naturally occurring isotopes of uranium would be present in chemically separated thorium. The particular isotopes of thorium that would be present, in addition to ²³²Th

and ²²⁸Th, include ²³⁴Th and ²³⁰Th produced in the decay of ²³⁸U, and ²³¹Th and ²²⁷Th produced in the decay of ²³⁵U (see Table 3.1.2).

The amounts of the thorium isotopes other than ²³²Th and ²²⁸Th that would be present in chemically separated thorium depend on the particular ore. However, based on the following considerations, the ²³⁴Th decay product of ²³⁸U and the ²³¹Th and ²²⁷Th decay products of ²³⁵U would not be present in significant activities in any thorium that had been aged for a few months after chemical separation, regardless of their initial activities relative to the activities of ²³²Th and ²²⁸Th. First, the activities of these decay products decrease rapidly with time, because of their short half-lives (see Table 3.1.2). Second, the external dose from these decay products, including the contributions from any of their short-lived decay products, is less than the contribution from ²³²Th and ²²⁸Th with their decay products present. Finally, the internal dose from ingestion or inhalation of these decay products is much less than the contributions from ²³²Th and ²²⁸Th with their decay products present. Finally, the internal dose from ingestion or inhalation of these decay products is much less than the contributions from ²³²Th and ²²⁸Th and ²²⁸Th and ²²⁸Th (see Tables 3.1.6 and 3.1.7).

Therefore, ²³⁰Th is the only decay product of the naturally occurring isotopes of uranium of potential concern in chemically separated thorium. Indeed, there was a report that ²³⁰Th accounted for most of the alpha activity in welding rods containing thorium (see Section 3.6), although the presence of significant quantities of ²³⁰Th in chemically separated thorium has not been noted in any other studies.

Available information on the most commonly used sources of thorium indicates that ²³⁰Th should not be radiologically significant in chemically separated thorium compared with ²³²Th and ²²⁸Th. Information supporting this conclusion is summarized below.

Most of the thorium used commercially has been extracted from the mineral monazite (Hedrick, 1991; NCRP 118). Other ores containing greater amounts of uranium have not been significant sources of thorium. The amount of uranium and, thus, ²³⁰Th in monazite is somewhat variable (NCRP 118), but Albert (1966) reported that monazite typically contains an activity of ²³⁰Th equal to 11% of the activity of ²³²Th and ²²⁸Th. Based on this activity, the contribution to dose from ²³⁰Th relative to the contribution from ²³²Th and ²²⁸Th and ²²⁸Th and their decay products can be estimated as follows.

Because ²³⁰Th emits only low intensities of low-energy photons (Kocher, 1981), this isotope would not contribute significantly to external dose for chemically separated thorium, especially when the rapid buildup of the photon-emitting decay products of ²²⁸Th is considered (see Tables 3.1.5 and 3.1.7). For ingestion, the dose from ²³⁰Th would be only about 2% of the total dose from all thorium isotopes in freshly separated materials, based on the dose coefficients given in Tables 3.1.6 and 3.1.7. The contribution would decrease to about 1.4% in aged materials in which the decay products of ²³²Th and ²²⁸Th are in activity equilibrium. For inhalation of either Class W or Class Y materials, the dose from ²³⁰Th also would be only about 2% of the total dose from ²³⁰Th in monazite would be insignificant.

The isotope ²³⁰Th decays to ²²⁶Ra and its short-lived decay products, and these decay products are radiologically more significant than the parent isotope (see Table 3.1.6). However, the decay products of ²³⁰Th do not occur in significant quantities in chemically separated thorium for hundreds of years or more (see Table 3.1.4). For example, the contribution to internal dose from ²³⁰Th would approach 10%, which is still a relatively insignificant amount, for those times

after 1000 years following chemical separation. Therefore, the buildup of ²²⁶Ra and its decay products would not be of concern during the normal life cycle of exempted products or materials containing thorium extracted from monazite, or for long times after disposal.

...

Based on the analysis described above, the possible presence of ²³⁰Th in chemically separated thorium is not considered in the assessments of exempted products or materials containing source material in the following subsections of Section 3.

Table 3.1.1 Mass and Activity Abundances of Naturally Occurring Isotopes in Chemically Separated Uranium and Thorium

Element	Isotope	Half-life ^a (yr)	Mass Abundance	Activity Abundance
Uranium (natural)	²³⁸ U	4.468×10 ⁹	99.2745% [⊳]	48.83%°
	²³⁵ U	7.038×10 ⁸	0.720%	2.34%
	²³⁴ U	2.445×10⁵	0.0055%	48.83%
Uranium (depleted)	238U	4.468×10 ⁹	99.75% ^d	90.1% ^e
	²³⁵ U	7.038×10 ⁸	0.25%	1.5%
	²³⁴ U	2.445×10⁵	0.0005%	8.4%
Thorium	²³² Th	1.405×10 ¹⁰	100%	50% ^f
	²²⁸ Th	1.9132		50%

^a Values from Kocher (1981).

^b Mass abundances for isotopes in natural uranium from Parrington et al. (1996).

^c Activity abundances for isotopes in natural uranium are based on known mass abundances and half-lives and assumption that ²³⁸U and ²³⁴U occur in equal activity abundances.

^d Mass abundances for isotopes in depleted uranium obtained as byproduct residues from uranium enrichment reported by Rich et al. (1988).

^e Activity abundances for isotopes in depleted uranium are based on assumed mass abundances and known half-lives.

[†] Activity abundances for isotopes in natural thorium are based on assumption of equal activity abundances of ²³²Th and ²²⁸Th at time of chemical separation and assumption that no other isotopes of thorium produced in decay of naturally occurring isotopes of uranium are present (see Section 3.1.5).

Isotope	Decay Product	Half-Life ^a	Specific Activity ^ь (curie(Ci)/g) ^c	Branching Fraction ^d	Principal Decay Modes
²³⁸ U		4.468×10 ⁹ yr	3.35×10⁻ ⁷		Alpha
	²³⁴ Th ^e	24.10 day		1.0	Beta/gamma
	^{234m} Pa ^e	1.17 min		1.0	Beta/gamma
	²³⁴ Pa ^e	6.70 h		0.0016	Beta/gamma
	²³⁴ U	2.445×10⁵ yr		1.0	Alpha
	²³⁰ Th	7.7×10⁴ yr		1.0	Alpha
	²²⁶ Ra	1,600 yr		1.0	Alpha
	²²² Rn	3.8235 day		1.0	Alpha
	²¹⁸ Po	3.05 min		1.0	Alpha
	²¹⁴ Pb	26.8 min		0.9998	Beta/gamma
	²¹⁴ Bi	19.9 min		1.0	Beta/gamma
	²¹⁴ Po	163.7 μs		0.99979	Alpha
	²¹⁰ Pb	22.26 yr		1.0	Beta
	²¹⁰ Bi	5.013 day		1.0	Beta
	²¹⁰ Po	138.378 day		1.0	Alpha

Table 3.1.2 Decay Data for Naturally Occurring Isotopes of Uranium and Their Radioactive Decay Products

See end of table for footnotes.

....

Isotope	Decay Product	Half-Life ^a	Specific Activity ^b (Ci/g) ^c	Branching Fraction ^d	Principal Decay Modes
²³⁵ U		7.038×10 ⁸ yr	2.16×10⁻⁵		Alpha/gamma
	²³¹ Th ^e	25.52 h		1.0	Beta/gamma
	²³¹ Pa	3.276×10⁴ yr		1.0	Alpha/gamma
	²²⁷ Ac	21.773 yr		1.0	Beta/alpha
	²²⁷ Th	18.718 day		0.9862	Alpha/gamma
	²²³ Fr	21.8 min		0.0138	Beta/gamma
	²²³ Ra	11.434 day		1.0	Alpha/gamma
	²¹⁹ Rn	3.96 s		1.0	Alpha/gamma
	²¹⁵ Po	1.778 ms		1.0	Alpha
	²¹¹ Pb	36.1 min		1.0	Beta/gamma
	²¹¹ Bi	2.13 min		1.0	Alpha/gamma
	²¹¹ Po	0.516 s		0.00273	Alpha/gamma
	²⁰⁷ Tl	4.77 min		0.99727	Beta/gamma
²³⁴ U ^f	g	2.445×10⁵ yr	6.24×10 ⁻³		Alpha

Table 3.1.2 Decay Data for Naturally Occurring Isotopes of Uranium and Their Radioactive Decay Products (continued)

^a Values from Kocher (1981).

^b Specific activity of isotope, defined as activity per unit mass.

^c 1 Ci/g = 0.037 TBq/g.

^d Number of atoms of decay product per decay of parent radionuclide.

^e Short-lived radioactive decay product that would be present in significant quantities within short time after chemical separation of uranium.

[†] Entries for radioactive decay products are given following entry for ²³⁴U decay product of ²³⁸U. ⁹ No decay products would be present in significant quantities for many years after chemical separation of uranium.

lsotope	Decay Product	Half-Life ^a	Specific Activity ^b (Ci/g)°	Branching Fraction ^d	Principal Decay Modes
²³² Th	e	1.405×10 ¹⁰ yr	1.09×10 ⁻⁷		Alpha
	²²⁸ Ra	5.75 yr		1.0	Beta
	²²⁸ Ac	6.13 h		1.0	Beta/gamma
	²²⁸ Th	1.9132 yr		1.0	Alpha
	²²⁴ Ra	3.62 day		1.0	Alpha/gamma
	²²⁰ Rn	55.61 s		1.0	Alpha
	²¹⁶ Po	0.146 s		1.0	Alpha
	²¹² Pb	10.643 h		1.0	Beta/gamma
	²¹² Bi	60.55 min		1.0	Beta/gamma
	²¹² Po	0.298 μs		0.6407	Alpha
	²⁰⁸ TI	3.053 min		0.3593	Beta/gamma
²²⁸ Th ¹	е	1.9132 yr	8.19×10 ²		Alpha

Table 3.1.3 Decay Data for Naturally Occurring Isotopes of Thorium and Their Radioactive Decay Products

^a Values from Kocher (1981).

···.

^b Specific activity of isotope, defined as activity per unit mass.
^c 1 Ci/g = 0.037 TBq/g.
^d Number of atoms of decay product per decay of parent radionuclide.
^e All radioactive decay products would be present in significant quantities within short time after chemical separation of thorium.

^f Entries for radioactive decay products are given following entry for ²²⁸Th decay product of ²³²Th.

	Time After Chemical Separation of Uranium (yr)				
Radionuclide ^b	10 ²	10 ³	10⁴	10 ⁵	10 ⁶
²³⁸ U chain				······	
²³⁸ U	1.0	1.0	1.0	1.0	1.0
²³⁴ U	2.8×10⁻⁴	2.8×10⁻³	2.8×10 ⁻²	0.25	0.94
²³⁰ Th	1.3×10⁻ ⁷	1.3×10⁻⁵	1.2×10⁻³	8.8×10⁻²	0.91
²²⁶ Ra	1.8×10⁻⁰	1.7×10⁻ ⁶	7.9×10⁻⁴	8.4×10 ⁻²	0.91
²¹⁰ Pb	8.5×10⁻¹⁰	1.5×10⁻ ⁶	7.9×10 ⁻⁴	8.4×10 ⁻²	0.91
²³⁵ U chain					
²³⁵ U	1.0	1.0	1.0	1.0	1.0
²³¹ Pa	2.1×10⁻³	2.1×10 ⁻²	0.19	0.88	1.0
²²⁷ Ac	1.5×10 ⁻³	2.0×10⁻²	0.19	0.88	1.0
²³⁴ U chain					
²³⁴ U	1.0	1.0	0.97	0.75	5.9×10⁻²
²³⁰ Th	9.0×10⁻⁴	9.0×10 ⁻³	8.5×10⁻²	0.51	8.6×10 ⁻²
²²⁶ Ra	1.9×10⁻⁵	1.7×10⁻³	6.6×10⁻²	0.50	8.6×10 ⁻²
²¹⁰ Pb	1.1×10⁻⁵	1.6×10 ⁻³	6.6×10 ⁻²	0.50	8.6×10 ⁻²
²³⁸ U + ²³⁴ U ^c	1.0	1.0	1.0	1.0	1.0
²³⁸ U	0.50	0.50	0.50	0.50	0.50
²³⁴ U	0.50	0.50	0.50	0.50	0.50
²³⁰ Th	4.5×10⁻⁴	4.5×10⁻³	4.3×10⁻²	0.30	0.50
²²⁶ Ra	9.6×10⁻ ⁶	8.5×10⁻⁴	3.3×10 ⁻²	0.29	0.50
²¹⁰ Pb	5.4×10 ⁻⁶	8.0×10⁻⁴	3.3×10⁻²	0.29	0.50

Table 3.1.4 Activity of Radioactive Decay Products of Uranium vs. Time After Chemical Separation of Uranium ^a

^a Activities of each radionuclide are normalized to initial activity of unity for parent radionuclide or isotopes of uranium at time of chemical separation.

^b At any time after chemical separation of uranium, all other shorter lived decay products listed in Table 3.1.2 would be in activity equilibrium with their immediate longer lived precursors, taking into account the decay branching fraction.

^c Activities of the two uranium isotopes are assumed to be equal at time of chemical separation, as would be the case for naturally occurring uranium.

-	Time After Chemical Separation of Thorium (yr)				
Radionuclide	0.5	11	5	10	20
²³² Th chain					
²³² Th	1.0	1.0	1.0	1.0	1.0
²²⁸ Ra	5.8×10 ⁻²	0.11	0.45	0.70	0.91
²²⁸ Th	5.0×10⁻³	1.9×10 ⁻²	0.26	0.56	0.87
²²⁴ Ra	4.8×10⁻³	1.8×10 ⁻²	0.26	0.56	0.87
²²⁸ Th chain					
²²⁸ Th	0.83	0.70	0.16	2.7×10 ⁻²	7.1×10⁻⁴
²²⁴ Ra	0.84	0.70	0.16	2.7×10⁻²	7.1×10⁻⁴
²³² Th + ²²⁸ Th ^c	0.92	0.86	0.71	0.80	0.93
²³² Th	0.50	0.50	0.50	0.50	0.50
²²⁸ Ra	2.9×10⁻²	5.7×10 ⁻²	0.23	0.35	0.46
²²⁸ Th	0.42	0.36	0.21	0.30	0.43
²²⁴ Ra	0.42	0.36	0.21	0.30	0.43

Table 3.1.5 Activity of Radioactive Decay Products of Thorium vs. Time After Chemical Separation of Thorium ^a

^a Activities of each radionuclide are normalized to initial activity of unity for parent radionuclide or isotopes of thorium at time of chemical separation.

^b At any time after chemical separation of thorium, all other shorter lived decay products listed in Table 3.1.3 would be in activity equilibrium with their immediate longer lived precursors, taking into account decay branching fraction.

^c Activities of the two thorium isotopes are assumed to be equal at time of chemical separation, as would be the case for naturally occurring thorium.

		Dose Coefficient		
Radionuclide ^a	Γ ^ь (rem/h per μCi)	Air Submersion ^c (rem-m³/µCi-yr)	Ingestion ^{d, e} (rem/µCi)	Inhalation ^{d, f} (rem/µCi)
²³⁸ U	6.5×10⁻ ⁸	4.3×10⁻⁴	2.6×10 ⁻¹ (0.05) 2.4×10 ⁻² (0.002)	2.5 D 7.0 W 1.2×10² Y
²³⁴ Th ^g	7.5×10⁻ ⁸	4.0×10 ⁻²	1.4×10⁻²	
^{234m} Pa ^g	1.0×10 ⁻⁸	1.5×10⁻¹		
²³⁴ Pa ⁹	3.2×10 ⁻⁹	1.8×10 ⁻²		
²³⁴ U			2.8×10 ⁻¹ (0.05) 2.6×10 ⁻¹ (0.002)	2.7 D 7.9 W 1.3×10² Y
²³⁰ Th			5.5×10 ⁻¹	3.3×10² W 2.6×10² Y
²²⁶ Ra	1.2×10 ⁻⁸	3.7×10 ⁻²	1.3	8.6 W
²²² Rn				3.2×10 ^{2 h}
²¹⁴ Pb	3.2×10 ⁻⁷	1.4		
²¹⁴ Bi	8.4×10 ⁻⁷	9.1		
²¹⁰ Pb			5.4	1.4×10 ¹ D
²¹⁰ Po			1.9	9.4 D 8.6 W

Table 3.1.6 Dosimetry Data for Naturally Occurring Isotopes of Uranium and Their Radioactive Decay Products

See end of table for footnotes.

		Dose Coefficient		
Radionuclide ^a	۲ ^ь (rem/h per µCi)	Air Submersion ^c (rem-m³/µCi-yr)	Ingestion ^{d, e} (rem/µCi)	Inhalation ^{d, f} (rem/µCi)
²³⁵ U	3.4×10 ⁻⁷	8.5×10 ⁻¹	2.7×10 ⁻¹ (0.05) 2.7×10 ⁻² (0.002)	2.5 D 7.3 W 1.2×10 ² Y
²³¹ Th ^g	5.5×10⁻ ⁷	6.4×10 ⁻²		
²³¹ Pa	3.7×10⁻ ⁷	2.0×10 ⁻¹	1.1×10 ¹	1.3×10³ W 8.6×10² Y
²²⁷ Ac			1.4×10 ¹	6.7×10 ³ D 1.7×10 ³ W 1.3×10 ³ Y
²²⁷ Th	4.2×10 ⁻⁷	5.7×10⁻¹		
²²³ Ra	3.3×10⁻ ⁷	7.2×10 ⁻¹	6.6×10⁻¹	
²¹⁹ Rn	5.3×10 ⁻⁸	3.2×10⁻¹		
²¹¹ Pb	3.6×10 ⁻⁸	3.3×10⁻¹		
²¹¹ Bi	4.7×10 ⁻⁸	2.6×10⁻¹		
²⁰⁷ TI	1.3×10 ⁻⁹	5.5×10 ⁻²		
²³⁴ ∪i	7.8×10 ⁻⁸	9.4×10 ⁻⁴	2.8×10 ⁻¹ (0.05) 2.6×10 ⁻² (0.002)	2.7 D 7.9 W <u>1.3×10² Y</u>

Table 3.1.6 Dosimetry Data for Naturally Occurring Isotopes of Uranium and Their Radioactive Decay Products (continued)

.

See following page for footnotes.

Footnotes to Table 3.1.6

^a Indented entries are radioactive decay products of parent uranium isotopes. Dosimetry data for decay products are listed only if they are significant compared with data for parent uranium isotope or other precursor radionuclides for uranium that is not chemically separated. Dosimetry data for decay products take into account branching fraction in decay of parent radionuclide given in Table 3.1.2.

^b Specific gamma-ray dose constant obtained from Unger and Trubey (1981) gives doseequivalent rate per unit activity at a distance of 1 meter from an unshielded point source in air. 1 rem/h per μ Ci = 270 millisieverts (mSv)/h per MBq.

^c Values for external exposure from submersion in uniformly contaminated, semi-infinite atmospheric cloud obtained from EPA 402--R-93-081. The skin dose component has been included using a 0.01 weighting factor. 1 rem-m³/ μ Ci-yr = 8.57×10⁻¹⁵ Sv-m³/Bq-s.

^d Values for internal exposure obtained from EPA-520/1-88-020. 1 rem/ μ Ci = 270 mSv/MBq. ^e If more than one value is given, entry in parentheses is corresponding gastrointestinal-tract absorption fraction. Assigned absorption fraction for different chemical forms of element is given in Table 3.1.8.

¹ Assumed lung clearance class is denoted by D for days, W for weeks, or Y for years. Assigned clearance class for different chemical forms of element is given in Table 3.1.8. ⁹ Short-lived decay product that would be present in chemically separated uranium; all other decay products would be present only in uranium that is not chemically separated.

^h Value obtained from ICRP 50 (see references) gives effective dose-equivalent rate from inhalation of ²²²Rn and its short-lived decay products per unit concentration of ²²²Rn in indoor air, in units of rem-m³/ μ Ci-yr, assuming activity equilibrium in air between radon and its decay products. For ²²²Rn in outdoor air, value is increased by factor of 1.4. Representative mean values of activity equilibrium factor for ²²²Rn decay products in indoor and outdoor air are 0.45 and 0.7, respectively (ICRP 50). 1 rem-m³/ μ Ci-yr = 8.57×10⁻¹⁵ Sv-m³/Bq-s.

¹ Entries for radioactive decay products are given following entry for ²³⁴U decay product of ²³⁸U.

		Dose Coefficient			
Radionuclide ^a	Γ ^ь (rem/h per μCi)	Air Submersion ^c (rem-m³/µCi-yr)	Ingestion ^d (rem/µCi)	Inhalation ^{d, e} (rem/µCi)	
²³² Th	6.8×10 ⁻⁸	1.1×10⁻³	2.7	1.6×10 ³ W 1.2×10 ³ Y	
²²⁸ Ra			1.4		
²²⁸ Ac	8.4×10 ⁻⁷	5.7			
²²⁸ Th	7.9×10⁻ ⁸	1.1×10 ⁻²	4.0×10⁻¹	2.5×10² W 3.4×10² Y	
²²⁴ Ra	1.1×10 ⁻⁸	5.6×10 ⁻²	3.7×10⁻¹		
²²⁰ Rn				1.6×10 ³¹	
²¹² Pb	2.7×10 ⁻⁷	8.2×10 ⁻¹			
²¹² Bi	1.9×10 ⁻⁷	1.1			
²⁰⁸ TI	6.1×10 ⁻⁷	7.6			
²²⁸ Th ⁹	7.9×10 ⁻⁸	1.1×10 ⁻²	4.0×10 ⁻¹	2.5×10 ² W 3.4×10 ² Y	

Table 3.1.7 Dosimetry Data for Naturally Occurring Isotopes of Thorium and Their Radioactive Decay Products

^a Only ²³²Th and ²²⁸Th are assumed to be present in significant amounts in naturally occurring thorium (see Section 3.1.5). Indented entries are decay radioactive products of parent thorium isotopes. All decay products would be present in chemically separated thorium within a short time after separation (see Table 3.1.5), but dosimetry data for decay products are listed only if they are significant compared with data for parent thorium isotope. Dosimetry data for decay products take into account branching fraction in decay of parent radionuclides given in Table 3.1.2.

^b Specific gamma-ray dose constant obtained from Unger and Trubey (1981) gives doseequivalent rate per unit activity at a distance of 1 meter from an unshielded point source in air. 1 rem/h per μ Ci = 270 mSv/h per MBq.

^c Values for external exposure from submersion in uniformly contaminated, semi-infinite atmospheric cloud obtained from EPA 402–R–93–081. The skin dose component has been included using a 0.01 weighting factor. 1 rem-m³/ μ Ci-yr = 8.57×10⁻¹⁵ Sv-m³/Bq-s.

^d Values for internal exposure obtained from EPA-520/1-88-020. 1 rem/ μ Ci = 270 mSv/MBq. ^e Assumed lung clearance class is denoted by D for days, W for weeks, or Y for years. Assigned clearance class for different chemical forms of element is given in Table 3.1.8.

^f Value obtained from ICRP 50 (see references) gives effective dose-equivalent rate from inhalation of ²²⁰Rn and its short-lived decay products per unit concentration of ²²⁰Rn in indoor or outdoor air, in units of rem-m³/ μ Ci-yr, assuming activity equilibrium in air between radon and its decay products. Theoretical estimates of activity equilibrium factor for ²²⁰Rn decay products in indoor air are 0.02 to 0.1 (ICRP 50). 1 rem-m³/ μ Ci-yr = 8.57×10⁻¹⁵ Sv-m³/Bq-s. ⁹ Entries for radioactive decay products are given following entry for ²²⁸Th decay product of ²³²Th

	Ingestion		Inhalation		
Element	Compound	f	Compound	f ₁ /Class	
U (Uranium)	Hexavalent Insoluble forms	0.05 0.002	UF_6 , UO_2F_2 , and $UO_2(NO_3)_2$	0.05 D	
			UO_3 , UF_{4} and UCl_4	0.05 W	
			UO_2 and U_3O_8	0.002 Y	
Th (Thorium)	All forms	2×10 ⁻⁴	Oxides and hydroxides	2×10⁻⁴ Y	
			All others	2×10⁻⁴ W	
Pa (Protactinium)	All forms	1×10 ⁻³	Oxides and hydroxides	1×10⁻³ Y	
			All others	1×10⁻³ W	
Ac (Actinium)	All forms	1×10 ⁻³	Oxides and hydroxides	1×10 ⁻³ Y	
			Halides and nitrates	1×10 ⁻³ W	
			All others	1×10⁻³ D	
Au (Gold)	All forms	0.1	Oxides and hydroxides	0.1 Y	
			Halides and nitrates	0.1 W	
			All others	0.1 D	
Po (Polonium)	All forms	0.1	Oxides, hydroxides and nitrates	0.1 W	
·			All others	0.1 D	

Table 3.1.8 Gastrointestinal-Tract Absorption Fractions (f1) and Lung Clearance Classesfor Chemical Compounds of Uranium, Thorium, and Their Decay Products a

^a Assignments of gastrointestinal-tract absorption fractions and lung clearance classes obtained from EPA–520/1–88–020. For elements not listed in this table, dose coefficients for ingestion and inhalation in Tables 3.1.6 and 3.1.7 apply to all chemical forms.

3.2 Chemical Mixture, Compound, Solution, Alloy Containing <0.05 Percent by Weight of Source Material

3.2.1 Introduction

In 10 CFR 40.13(a), any person is exempt from the requirements for a license, to the extent the such person receives, possesses, uses, transfers, or delivers source material in any chemical mixture, compound, solution, or alloy in which the source material is by weight less than 0.05% of the mixture, compound, solution, or alloy. This exemption was proposed on September 7, 1960 (25 FR 8619), and issued as a final rule on January 14, 1961 (26 FR 284).

However, previously the definition of source material in 10 CFR Part 40 excluded all uranium and thorium in materials less than 0.05% by weight of uranium and thorium. In the original 10 CFR Part 40, published in 1947, source material was defined as "any material, except fissionable material, which contains by weight one-twentieth of 1% (0.05%) or more of (1) uranium, (2) thorium, or (3) any combination thereof." The rulemaking addressed in the aforementioned Federal Register notices constituted an overall revision of 10 CFR Part 40. In that rulemaking, the definition of source material was changed to be consistent with the definition in the Atomic Energy Act of 1954 to its current wording: "Source material means (1) Uranium or thorium, or any combination thereof, in any physical or chemical form or (2) ores which contain by weight one-twentieth of 1% (0.05%) or more of (i) uranium, (ii) thorium or (iii) any combination thereof. Source material does not include special nuclear material." Because of the change in the definition of source material, the exemption for materials other than ore that are less than 0.05% of source material was added, with the net effect of no change to the licensing program. The record does not reflect a full consideration of the health and safety significance of this concentration of source material. Applying the same concentration limit, as used for ore in the definition of source material, to source material in other forms was administratively convenient.

In the Atomic Energy Act of 1946, source material was defined as "uranium, thorium, or any other material which is determined by the Atomic Energy Commission (AEC), with the approval of the President, to be peculiarly essential to the production of fissionable materials; but includes ores only if they contain one or more of the foregoing materials in such concentrations as the AEC may by regulation determine from time to time." In the Atomic Energy Act of 1954, the definition was modified somewhat, but not substantively. Also in the 1946 Act, Section 5(b)(2), it was stated that licenses shall not be required for quantities of source materials that, in the opinion of the AEC, are unimportant. Section 62 of the Atomic Energy Act of 1954 contains the same statement. It appears that the Commission selected the 0.05% value primarily on the basis of that concentration of source material that was considered strategically important for the production of special nuclear material.

The exemption refers only to the weight percent of uranium and thorium, 0.05%, which equates to 500 ppm. To provide a comparison of weight percent to activity concentrations for uranium and thorium, two major types of materials are considered--natural and processed. Natural uranium consists of ²³⁸U (99.27% by mass), ²³⁵U (0.72% by mass), and ²³⁴U (0.0054% by mass). Natural and processed thorium consists of ²³²Th (nearly 100% by mass and 50% by activity) and ²²⁸Th (50% by activity). Depleted uranium is an example of chemically processed uranium; it differs from natural uranium by containing less ²³⁵U (0.2 to 0.25% by mass) and ²³⁴U

0.05% weight of uranium and thorium. Refer to Section 3.1 for a more detailed discussion on the properties of source material.

3.2.2 Description of Materials

Since uranium and thorium are ubiquitous in the environment, there is a vast range of products and materials that may contain low concentrations of source material and thus be covered by this exemption. Only a few of these involve purposeful introduction of the source material. Uranium and thorium exist naturally in soil, and trace quantities appear in many products and materials, including the human body. Based on a literature review and discussions with knowledgeable individuals, it appears that some of the specific materials and products covered by this exemption that may have significant concentrations of source material are:

- dental products containing uranium;
- ophthalmic glass;
- mineral-derived products from minerals such as zirconium, hafnium, and titanium and large-volume metals, such as copper and iron, coal, and phosphates; and
- waste-derived products from materials such as coal ash, phosphate slag, and water treatment sludge.

The following subsections contain brief descriptions of individual products, such as dental products containing uranium and ophthalmic glass, and categories of products, such as mineral-derived and waste-derived products, which are covered by this exemption.

3.2.2.1 Dental Products

The practice of adding uranium to dental ceramics in the United States began during the early 1900s and continued until the early 1980s. It was discovered that a small amount of uranium contributed a natural color and fluorescence to dentures. Restoration of natural appearance is one of the major reasons for using prostheses. Other substances have been found to imitate these characteristics over a broad range of daylight and artificial lighting conditions. American manufacturers have used depleted uranium in their porcelain dental products, whereas, at least in certain Japanese products, natural uranium has been used (Phone call, D. L. Thompson, Center for Devices and Radiological Health, U.S. Food and Drug Administration, May 1995).

In 1979, the American Dental Association (ADA) developed Specification No. 52, which provided standards for the uranium content in dental porcelain and porcelain teeth (ANSI/ADA Spec. No. 52). This standard established a 40% reduction in the permissible amount of uranium that could be used in porcelain teeth (0.03% by weight). Assuming depleted uranium is used, 0.03% by weight is equivalent to about 3.7 becquerels (Bq)/g (100 picocuries (pCi)/g). However, the fluorescent behavior of dental ceramics was still not identical to natural teeth in all cases and the use of uranium in ceramics was considered to be undesirable. Dental porcelain manufacturers, while adhering to this standard, were also looking for alternative materials to replace uranium. In the early 1980s, the last of the U.S. major manufacturers had phased out uranium in dental ceramics (Phone call, D. L. Thompson, Center for Devices and Radiological

Health, U.S. Food and Drug Administration, May 1995). According to ANSI/ADA Specification No. 69, which became effective May 1992, the manufacturer must submit a certification of assurance verifying that fluorescing agents that will increase the radioactivity of the ceramic have not been added (ANSI/ADA, Spec. No. 69). In addition, Specification No. 52 had been withdrawn (ANSI/ADA, Spec. No. 69).

Full dentures have an average life of only 5 to 11 years since the oral cavity may change over time (Mazurat, 1992). The predicted service life for removable partial dentures is 8 to 10 years (Mazurat, 1992). For denture wearers 65 years and over, the age of dentures worn is typically 10 years or more. In one study it was found that 20% of denture wearers had worn dentures for 20 years (Mazurat, 1992). Typically, old dentures are either thrown into the trash or retained in the home (Phone call, A. Kayes, President of the Prosthodontics Society, May 1995).

Crowns and inlays are considered permanent dentifrices since they are replaced infrequently. The predicted service life for fixed restorations is 8 to 11 years, though there are instances in which these restorations may be left in place for greater periods of time. Therefore, though uranium has not been used in dental products since the early 1980s, some individuals may still have these crowns.

3.2.2.2 Ophthalmic Lenses

The manufacture of ophthalmic glass, which is glass used in eyeglasses or sunglasses, frequently used mixtures of rare earths and zirconium oxides, which contain natural uranium and thorium. These lenses should not be confused with intentionally thoriated glass lenses used in special instruments. Refer to Section 3.19 for information on thorium in finished optical lenses.

In 1975, the Optical Manufacturers Association (OMA) voluntarily issued an ophthalmic glass radiological standard (OMA, 1975). The purpose of this standard was to establish a uniform maximum level for radioactive emissions from ophthalmic glass. Use of the standard is voluntary. However, it appears that U.S. ophthalmic glass manufacturers comply with this standard (Phone call, W. Price, Product Assurance, Corning, Inc., Corning, NY, May 1995).

The use of glass lenses in eyeglasses has decreased considerably over the years as plastic lenses have become more popular. It is estimated that glass has only 5-8% of the U.S. eyeglass market (Optical Advisor (OA), 1999). Based on this estimate, there may be about 12 million glass-containing eyeglass wearers (Vision Council of America (VCA), 1999). The average lifetime of a pair of eyeglasses ranges from 2 to 4 years (Phone call, T. Loomis, Manager, Product Assurance, Optical Products Department, Corning Inc., Corning, NY, May 1995; phone call, W. Price, Product Assurance, Corning, Inc., Corning, NY, May 1995). Typically, eyeglasses are stored in the home or are disposed in a sanitary landfill and could go to municipal incinerators. Eyeglasses, both frames and lenses, have been redistributed to individuals in the United States and other parts of the world by service organizations such as the Lions Club (Phone call, W. Price, Product Assurance, Corning, Inc., Corning, NY, May 1995).

3.2.2.3 Mineral-Derived Products

Mineral-derived products can encompass a wide range of minerals that contain naturally occurring radioactive material, including uranium and thorium. Table 3.2.2 summarizes the uranium and thorium content found in selected mineral resources, products, and process wastes. Table 3.2.3 lists a few common building materials and associated uranium and thorium concentrations. As one may note, most of these products typically contain less than 0.05% by weight of uranium or thorium.

Products that are derived from minerals are numerous. Mineral-derived products range from mineral concentrates, which may only be physically processed, to products that have been highly refined, e.g., chemically processed and purified. These mineral-derived products may then be used in other consumer products. Examples of mineral-derived intermediate and final products are shown in Table 3.2.4.

Mineral extraction may include a number of processes, as illustrated in Table 3.2.5. Some of the processes may carry uranium and thorium into the product, or it may concentrate within the product or the waste. How radionuclides are incorporated into products and wastes are determined by a combination of factors, including uranium and thorium levels in ore, process chemistry, process temperature, and process and collection efficiency (Hendricks, 1987). The heavy mineral extraction industry provides an excellent example of how thorium and uranium can concentrate in wastes based on processing method, as shown in Figure 3.2.1.

Five major categories of minerals from which mineral-derived products result are: (1) rare earths; (2) the special application metals, such as zirconium (Zr), hafnium (Hf), and titanium (Ti); (3) the large volume-metals, such as copper and iron; (4) coal; and (5) phosphates. The rare earth industry is not described in this section since it is covered by a separate exemption in 10 CFR 40.13(c)(1)(vi); an assessment of that exemption is contained in Section 3.9. Heavy minerals, such as zirconium and titanium, and phosphates appear to typically exhibit elevated concentrations as compared to the other mineral resources. Therefore, the radiological impacts associated with these mineral resources are highlighted in this assessment.

3.2.2.4 Waste-Derived Products

3.2.2.4.1 Coal Ash

Coal contains naturally occurring uranium and thorium as well as their radioactive decay products. The radioactivity of coal is known to vary more than two orders of magnitude depending upon the type of coal and the region from which it has been mined. Utility and industrial boilers are estimated to generate 61 million metric tons of coal ash per year (Environmental Protection Agency (EPA) RAE–9232/1–2). Coal ash consists of about 74% fly ash, 20% bottom ash, and about 6% boiler slag. Uranium and its progeny tend to stay with the ash when the coal is burned. Concentrations of uranium in coal and ash vary widely, but some indications of mean values are given in Table 3.2.6.

Of the total amount of coal ash generated, nearly 20 million metric tons are used in a variety of applications instead of being sent to disposal facilities. Coal ash is primarily being used as an additive in concrete, as a structural fill, and for road construction. Fly ash has been used to replace cement in concrete in the United States since 1910. Typically, fly ash is substituted for

10 to 30% of the cement (EPA, RAE–9232/1–2). Since the early 1970s, all three types of coal ash have been used in construction projects, including industrial parks, housing developments, roadbed embankments, and soil stabilization (EPA, RAE–9232/1–2).

3.2.2.4.2 Phosphate Slag

Uranium in phosphate ores in the United States ranges in concentration from 20 to 200 ppm, while thorium occurs at ambient background concentrations of between 1 to 5 ppm (EPA 402–R–92–002). Phosphate slag is the principal waste byproduct generated from the production of elemental phosphorus. Phosphogypsum is the principal waste byproduct generated during the phosphoric acid production process. Some of the impurities contained in the phosphate slag and phosphogypsum include uranium and thorium and their radioactive decay products, which become concentrated in the waste. Table 3.2.7 summarizes typical radionuclide concentrations found in phosphate ore, phosphate slag, and phosphogypsum. As noted, uranium and thorium concentrations are higher in phosphate slag as compared to phosphogypsum. Because of the high temperatures involved in the thermal process, slag is a vitrified material that binds nonvolatile radionuclides.

Phosphate slag wastes have been used in a number of different applications (EPA, RAE–9232/1–2). Phosphate slag is used as an aggregate in asphalt manufacturing. Phosphate slag has been used extensively in highway construction for many years in Idaho, Montana, and Tennessee. The material is used as crushed base and crushed aggregate for asphalt and in casting material for highway structures. It has also been used as an aggregate for Portland cement and concrete in making construction blocks, driveways, patios, and drainage ditches. Phosphate slag is used as railroad ballast and as stabilization material for stockyards. In Florida, where use of phosphate slag in habitable structures has not been prohibited, slag has been used on roofing shingles, in septic tank fields, and in manufacture of rockwool insulation.

3.2.2.4.3 Water Treatment Sludge

Water supply systems generate an estimated yearly total of 3.1 million metric tons of waste, including sludge and other waste forms (EPA, RAE–9232/1–2). Of this, it is estimated that 700 water utilities generate 300,000 metric tons of naturally occurring radioactive material waste yearly, including sludge, spent resin, and charcoal beds. Most of this waste is disposed in landfills and lagoons or applied to agricultural fields (EPA, RAE–9232/1–2).

Federal and State regulations allow radioactive material to be discharged into sanitary sewage systems with specified concentrations, providing the material is readily soluble or dispersible biological material. In many cases, naturally occurring radioactive materials are not readily soluble and may accumulate in the sewer lines. Likely contributors of radionuclides that may reconcentrate in the sanitary sewage systems are the users of zircon sand, zircon flour, and thorium oxide (Conference of Radiation Control Program Directors (CRCPD), 1994). These users include foundries, casting facilities, producers of zirconium or refractories, and ceramic and welding rod manufacturers (CRCPD, 1994). In addition, naturally occurring radioactive material, including uranium and thorium, has been found to reconcentrate in sewage sludge. Therefore, it poses potential problems with the use and distribution of sludge and its byproducts, such as fertilizer and compost.

3.2.2.4.4 Oil and Gas Production and Process Waste

The presence of naturally occurring radioactive materials in oil and gas production and processing wastes has been recognized since the 1930s (Smith et al., 1995). The primary radionuclides of concern are ²²⁶Ra and ²²⁸Ra. The uranium and thorium radionuclides, present in the subsurface formation, are largely immobile and remain in the subsurface (Smith et al., 1995). The more soluble radium can become mobilized in the formation water and be transported to the surface in the produced water stream. Since uranium and thorium are not the radionuclides of concern, the radiological impacts associated with oil and gas production and process waste are not evaluated in this assessment.

3.2.3 Summary of Previous Analyses and Assessments

When this exemption was proposed and finalized, there was no accompanying information on analyses of radiological impacts on members of the public from the use of the materials. The proposed rule states only that possession and use of these materials "can be conducted without any unreasonable hazard to life," but the meaning of "unreasonable hazard" is not quantified.

In previous analyses for this exemption, dose assessments were performed only for consumer products distributed for use under this exemption: specifically, dental products (e.g., dentures and crowns) containing uranium and for ophthalmic lenses (e.g., eyeglasses) containing natural thorium. For the current assessment, information was also compiled on mineral-derived products, with a particular emphasis on zircon-derived products, and on waste-derived products, with an emphasis on coal ash, phosphate slag, and sewer sludge. A general description of the aforementioned products follows.

3.2.3.1 Dental Products

. .

Summarized in Table 3.2.8 are the results from a number of studies that evaluated the uranium concentrations in dental products and potential doses to denture and crown wearers. Buckley et al. (NUREG/CR–1775) conducted a comprehensive dose assessment associated with the manufacture, distribution, and use of dental products containing uranium. A brief description of the dental product dose assessment done by Buckley et al. (NUREG/CR–1775) is provided below.

Doses from routine use of dental products containing 0.05% by weight of uranium were estimated by Buckley et al. (NUREG/CR–1775). The routine exposure scenarios for exempt uses of these products included exposures to distributors and workers in dental laboratories and dental offices and exposures to wearers of dentures and crowns. The exposure scenarios involving accidents or misuse of dental products include exposures to firefighters from uranium released in a warehouse fire, and exposures to dental technicians from uranium dust generated during tooth construction. Doses were not estimated for disposal of dental products, since few false teeth would be placed in landfills and there should be little release of uranium from the porcelain material.

For wearers of dental restorations, the gums, inner cheeks, tongue, and lips may be exposed to alpha, beta, and gamma radiations. Dose estimates made by Buckley et al.

(NUREG/CR-1775), from alpha and beta particles from teeth containing 0.05% by weight of uranium are summarized below:

- Annual dose equivalents from alpha particles at the surface of the teeth range from 1 to about 4 sieverts (Sv) (100 to 400 rem). However, since the maximum range of the alpha particles in tissue is 30 μ m, essentially the entire dose to tissues of the mouth would be received by superficial cells on the surface and the dose to radiosensitive cells in the basal layer of the tissues would be zero (0).
- The annual dose equivalent to the basal layer of tissues from beta particles would be about 14 mSv (1.4 rem).

The dose estimates for distribution workers and denture wearers from gamma irradiation are summarized as follows:

- The annual whole-body dose equivalent to workers in distribution facilities from photon irradiation is about 3×10⁻⁵ mSv (0.003 mrem).
- For wearers of full dentures, the annual dose equivalent to the whole body from external exposure was 0.003 μSv (0.0003 mrem). For wearers of a few crowns, the estimated annual dose equivalent from external exposure was 0.0005 μSv (0.05 μrem). For all wearers (65 million), the annual collective dose equivalent was 0.1 person-Sv (10 person-rem).

The dose estimates obtained by Buckley et al. (NUREG/CR-1775) for the accident and misuse scenarios are summarized as follows:

- For a fire in a warehouse, the maximum dose commitment was 5 mSv (0.5 rem) to the lungs, and the whole-body dose to firefighters could be as high as 0.06 mSv (6 mrem). Conservative assumptions were made about the number of teeth in the warehouse and the amount of uranium that would be released in a fire. No respiratory protection equipment was assumed.
- For a dental technician working in a room in which uranium powder is lost during tooth construction, and using conservative assumptions for the concentration of uranium in dust in air and for exposure time, the whole-body dose was estimated to be $0.02 \ \mu$ Sv (2 μ rem).

Buckley et al. (NUREG/CR–1775) stated that the doses from alpha and beta particles to dental wearers were conservative, but not greatly overestimated. Thompson (1976) stated that it was possible that the absorption by saliva, pellicle, air and/or calculus may be sufficient in a portion of the population to prevent any alpha from ever reaching the soft tissue. As with alpha dose rates, the dose rate due to beta particles may be overestimated because of the assumption of intimate and continuous contact between teeth and tissues. However, particle-attenuating factors would not yield as significant a reduction for high-energy betas as for alpha particles (Thompson, 1976).

3.2.3.2 Ophthalmic Lenses

Ophthalmic glass is used to manufacture lenses for eyeglasses and eyepieces. Buckley et al. (NUREG/CR-1775) also analyzed radiological impacts on the members of the public from eyeglasses containing thorium. The routine exposure scenarios evaluated were (1) exposure during distribution by wholesale optical labs and dispensers, (2) exposure to wearers of eyeglasses, (3) exposure to workers during collection of solid waste containing discarded eyeglasses, and (4) exposure to the public following disposal of eyeglasses in a landfill or incineration. However, only doses to eyeglass wearers were estimated. In the other scenarios, doses were not estimated, because of the complexities in defining the exposure scenarios or to the presumption that doses would be very low based on similar scenarios for other consumer products, or because the dose estimates were unreasonably conservative, particularly for disposal of eyeglasses in landfills.

For an individual wearing eyeglasses containing 0.05% by weight of thorium, dose analyses were done for exposure to the eye, primarily from irradiation by alpha particles, and for wholebody exposures from photons. The dose estimates reported by Buckley et al. (NUREG/CR-1775) and in NCRP 95 assume that an individual wears eyeglasses for 16 h/day and that 55% of the population wears glasses. These dose estimates are summarized as follows:

- The annual dose equivalent to the cornea from alpha particles is about 0.04 Sv (4 rem). Assuming a tissue weighting factor of ≤10⁻⁴, the annual individual dose was estimated to be 0.004 mSv (0.4 mrem).
- The annual dose equivalent to the whole body of the individual from photons is about 2×10⁻⁴ mSv (0.02 mrem).

In NCRP 95, it was estimated that 50 million people in the United States wear eyeglasses with glass lenses. Assuming an annual individual effective dose equivalent (EDE) of 0.004 mSv (0.4 mrem) from alpha particles and a whole-body dose equivalent of 2×10^{-4} mSv (0.02 mrem), the annual collective EDE to the U.S. population would be about 200 person-Sv (20,000 person-rem).

NCRP 95 also evaluated doses to the cornea of the eye from irradiation by alpha particles during use of an instrument eyepiece containing 0.05% by weight of thorium. By assuming that the instrument was used by an individual for 20 h/wk, the estimated annual dose equivalent to the cornea from alpha particles is 0.03 Sv (3 rem).

The Yaniv study (AEC, 1974) concluded that radiation dose rates from ophthalmic glass could be reduced significantly with better quality control of the rare earth and zirconium oxides. Another problem revealed in this study was that the observed radiation is not directly related to the source material content of the glass, but due to the widely varying parent-progeny equilibrium conditions. The radiation emissions are mainly due to the short-lived decay products of ²³²Th and ²³⁸U, which can be present in glass even after the parent radionuclides are removed. Therefore, control of source material was not sufficient to eliminate radioactive material from glass (NCRP 95).
3.2.3.3 Mineral-Derived Products

This subsection is further divided into three sections. The first section describes studies that estimated radionuclide air concentrations and doses due to industrial uses of mineral sand products. The second section provides direct irradiation measurement results from the use of a cellophane tape dispenser that contains zircon sand as ballast. The final section describes studies that estimated doses and risks attributed to the disposal of mineral sand wastes, with an emphasis on zircon sand wastes.

3.2.3.3.1 Industrial Worker Exposures

Four articles evaluated exposures associated with the handling of zircon and ilmenite (FeTiO₂) products. They are from Shimko (1996), Lischinsky et al. (1991), Wallace and Leach (1987), and Boothe et al. (1980). In Shimko (1996), air sampling was performed at two plants that process zircon sands. The first plant (Plant A) blends zircon flour, which is ground zircon sand, with sodium carbonates, and pelletizes the mixture to produce a zirconium carbonate. At the second plant, Plant B, zircon sand is ground in a ball mill to produce zircon flour. The zircon flour is bagged and resold, primarily as a refractory and to make welding rods. Both plants adhere to Occupational Safety and Health Administration (OSHA) regulations for ionizing radiation (29 CFR 1910.96). Neither plant has an Nuclear Regulatory Commission (NRC) license. The uranium and thorium concentrations in zircon sands are given as 286 to 343 ppm and 116 to 157 ppm, respectively (Shimko, 1996). Measured air concentration data from plant B is provided in Table 3.2.9.

The EDEs due to inhalation were estimated. At Plant A, where the worker continuously performs the same tasks, the annual EDE was estimated to be 3 mSv (300 mrem), without any respiratory protection. At Plant B, the worker spends less than 10 h/wk processing zircon. Therefore, the worker's weekly exposure was less than 8% of the derived air concentration (DAC). However, if this worker processed material for 2000 h/yr, the estimated annual EDE could be 17 mSv (1700 mrem), without respiratory equipment. At both plants the workers wore National Institute for Occupational Safety and Health (NIOSH)-approved respirators for zirconium compounds.

During decommissioning activities of a former NRC-licensed ceramic manufacturing facility, Lischinsky and Vigliani (1991) found concentrations of 0.053% uranium and thorium in zirconium oxide that was marginally over the 0.05% weight criterion. The unlicenced zirconium oxide powders were used in the manufacture of an electronic component.

Several screening measurements of other zirconium sands from different vendors showed levels of radioactivity from very low trace quantities to about the same levels found by Lischinsky and Vigliani (1991). The worst case air concentration of 0.03 Bq/m³ (0.8 pCi/m³) was estimated using the OSHA zirconium compound air concentration limit of 5 mg/m³ multiplied by the uranium concentration of 5.4 Bq/g (150 pCi/g) in the zirconium sand. The worst case air concentration was considered to be about 15% of the unrestricted maximum permissible concentration (MPC). Using the same approach for ²³² Th, the estimated air concentration was 2.2×10^{-3} Bq/m³ (0.06 pCi/m³) in air, or roughly 6% of the unrestricted MPC (Lischinsky and Vigliani, 1991).

Surface contamination measurements of the zircon sand were in the range of 250 to $333 \text{ Bq}/100 \text{ cm}^2$ (15,000 to 20,000 dpm/100 cm²), which is three times greater than the average fixed beta-gamma limit stipulated in the NRC's surface contamination guides (NRC, Memoranda, Guidelines, 1982; Lischinsky and Vigliani, 1991). Furthermore, for dispersed radioactivity, a uranium activity concentration of approximately 5.4 Bq/g (147 pCi/g) was 15 times greater than the unrestricted release guideline of 0.37 Bq/g (10 pCi/g) for natural uranium. It was noted that the alpha contamination of approximately 7 to 8 Bq/100 cm² (400 to 500 dpm/100 cm²) was considered to be below the NRC average fixed surface contamination limits (Lischinsky and Vigliani, 1991).

In Wallace and Leach (1987), air sampling was performed to evaluate the potential exposures during sand-blasting operations. Ilmenite is used extensively in the sand-blasting industry. The ilmenite product used in Australia normally contains other minerals such as monazite and zircon. In Australia, thorium concentrations in ilmenite range from 50 to 500 ppm and uranium concentrations range from <10 to 30 ppm (Koperski, 1993). Particle sizing of the high-volume dust samples indicated that 90% of the radioactivity was in the 0.45 to 5 micron particle size ranges. Using the Y solubility and 1 micron activity median aerodynamic diameter, the estimated annual EDE from inhalation to a sand operator, without protective respiratory equipment, was 3 mSv (300 mrem).

In Boothe et al. (1980), external radiation and radon levels were measured from zircon sand. At contact the measured external exposure rate was 41 nanocoulomb (nC)/kg-h (158 microroentgen (μ R)/h). Above the zircon sand at 0.9 meter, measured exposure rates decreased to 18% of the on pile-readings. External measurements on three other comparable piles of zircon used by foundries were within ± 20% of the survey reading of 44 nC/kg-h (170 μ R/h).

The radon emanation rate measured above a stockpile of zircon sand was 0.074 Bq/m²-s (2 pCi/m²-s). According to Boothe et al. (1980), the radon emanation rate from uranium mill tailings is about 70 times higher than that from zircon sands. As explained by Boothe et al. (1980), this low emanation rate is because radium in zircon occurs interstitially in the $ZrSiO_4$ crystal, and radon is trapped within the crystalline lattice.

Early findings of the Oregon Radiation Control Section indicated that zircon sands used in Oregon foundries originated from Australian placer mines and were stockpiled in large quantities, in both bulk and bagged form (Boothe et al., 1980). Measurements made of a single 40-kg bag of zircon sand were 44 nC/kg-h (170 μ R/h) at the surface and 3.91 nC/kg-h (15 μ R/h) at 0.9 meter above the bag. Measurements made of 40-kg bags of sand stacked upon a pallet the size of 1.2 m × 1.2 m × 1.2 m gave exposure rates of 77 nC/kg-h (300 μ R/h) at the surface and 21 nC/kg-h (80 μ R/h) at 0.9 m. Further, measurements made over 2×10⁶ kg of bulk sand gave exposure rates of 52 nC/kg-h (200 μ R/h) at a height of 0.9 m above the surface. All of the above readings were uncorrected for background, which ranged from 2 to 2.6 nC/kg-h (8 to 10 μ R/h).

Another area evaluated by Boothe et al. (1980) was the possible exposure through inhalation during the use of zircon sand as flour (200-mesh or 74-micron size). The size commonly used is 200-mesh. The mesh number indicates only the screen size through which all particles will pass, not the particle size spread. Sand of this nature is used in the mulling or mixing of the mold formula by workers directly handling the mold by the use of the scoop or dump method.

Air monitoring in a high-use area within one large carbon steel casting plant and at the breathing zone level of five mullers at the plant showed no significant zircon flour dust concentrations, with one exception. However, in Boothe et al. (1980), no air concentrations were provided. Based on the limited information gathered, it could not be said that an inhalation problem did not exist during the mold-making process. Further study was suggested.

An article written by Hipkin and Paynter (1991) summarized typical activity concentrations of many materials that have a natural radioactivity content and estimated annual individual EDEs due to external radiation and dust inhalation, as shown in Tables 3.2.10 and 3.2.11, respectively. Though there was no detailed information in the article that indicated how the activity concentrations were derived or how the EDEs were estimated, it was stated that realistic assumptions were made when estimating doses and exposure durations.

3.2.3.3.2 Public Exposure

In a survey of common cellophane tape dispensers used in Oregon, it was found that the majority of tape dispensers used zircon or monazite sand for weight (Boothe et al., 1980). The distinction was made due to the gamma-ray spectrum exhibited; zircon exhibits a predominantly ²²⁶Ra spectrum, as compared to a thorium-decay series spectrum given off by monazite. For the zircon sand weight, the maximum exposure rate at the surface of the tape dispenser was between 5 and 10 nC/kg-h (20 and 40 μ R/h). The exposure rate at 0.6 m was about 77 pC/kg-h (0.3 μ R/h) above background (Boothe et al., 1980; CRCPD, 1981).

3.2.3.3.3 Waste Disposal

In Oregon, there are 14 foundries using zircon as basic sand, flour, slurry, or paint (Boothe et al., 1980). As of 1980, these foundries used and disposed of about 4170 metric tons of zircon each year. There is some recycling of zircon, and prior to disposal zircon is diluted with silica sand and other materials. It was estimated there were 54,400 metric tons of foundry zircon in landfills in the metropolitan area of Portland, Oregon. The radiological impact of zircon sand is greatly limited by the low radon emanation of the sand. The maximum external radiation level in landfills containing zircon sands was measured at 36 nC/kg-h (140 μ R/h). The average level at such landfills appeared to be about 5 nC/kg-h (20 μ R/h). The Oregon Radiation Control Section found no cases where houses or other structures were constructed over zircon sand wastes. As noted in Boothe et al. (1980), this may not be true in more industrialized parts of the country where there are more and larger foundries.

The largest volume of radioactive waste generated is sludge from the zirconium-hafnium separation process (Boothe et al., 1980). The sludge is considered to be a good soil conditioner. Approximately 10,900 metric tons of sludge were used on two agricultural fields near Albany, Oregon, in 1975 and 1978. The gross external irradiation exposure rates over a 60-acre field ranged from 3.6 to 23 nC/kg-h (14 to 90 μ R/h). The average exposure rate was 8 nC/kg-h (30 μ R/h). Background exposure rates ranged from 2 to 2.6 nC/kg-h (8 to 10 μ R/h). The maximum exposure rate of the 24-acre field was 5 nC/kg-h (20 μ R/h). The radon emanation rate over the 60-acre field was measured as 0.44 Bq/m²-s (12 pCi/m²-s).

Special application metal waste is the category used by the EPA (RAE-9232/1-2) to describe tailings and residues associated with zircon, hafnium, titanium, and tin processing. The

uranium and thorium concentrations used in the dose assessment for special application wastes are given in Table 3.2.12. The special application metal processing waste site is assumed to consist of commingled waste rock, overburden, and a tailing pile of 10 acres. The pile was 200 meters by 200 meters with a height of 20 meters, and has a 1-meter-thick soil cover. The pile contains 1.6 million metric tons of tailings, overburden, and wastes. The generic site is assumed to be located in a rural area in Texas and underlain with an aquifer. The population density is assumed to be 65 persons per square mile (EPA, RAE–9232/1–2). Utilization and disposal of waste materials from these special application metals are varied. It is assumed that the waste materials have negligible reuse.

Summarized in Table 3.2.13 are the estimated annual EDEs for workers at the Texas disposal site. For workers at the disposal site, the primary exposure pathways are direct irradiation and dust inhalation. Only for the on-site office worker were indoor radon risks evaluated. It was assumed the office was located on top of the closed waste pile. Summarized in Table 3.2.14 are the estimated annual individual EDEs to members of the public from storage and disposal of special application metal processing wastes. Collective EDEs from storage or disposal of these wastes are shown in Table 3.2.15.

3.2.3.4 Waste-Derived Products

The use of coal ash and phosphate slag in building and roadway construction materials and the disposal of these wastes, as well as water treatment sludge, are described in the following subsections.

3.2.3.4.1 Coal Ash

3.2.3.4.1.1 Product Use

Potential doses due to use of fly ash as an aggregate in concrete or cement were estimated for the public by EPA (RAE–9232/1–2). It was assumed that 2% of the 6 million metric tons of fly ash is used in various cement and concrete applications. EPA (RAE–9232/1–2) assumed the generic dwelling area was 143 m² and that about 34 m³ of concrete were used in the dwelling. The thickness of the concrete was assumed to be 0.13 meter. The estimated annual EDE due to direct irradiation from building materials was estimated to be 0.17 mSv (17 mrem). If 7,500 dwellings for 30,000 persons are constructed from 1 year's reuse of coal ash, the resulting annual collective EDE would be about 5.1 person-Sv (510 person-rem).

3.2.3.4.1.2 Coal Ash Disposal

The EPA (Cont. No. 68–02–4375) conducted a study to estimate the potential doses and risks associated with environmental releases from coal and coal ash piles at utility and industrial facilities with coal-fired boilers. The purpose of the study was to provide background information to consider exempting coal and coal ash piles from the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA) reportable quantity notification requirements. For direct irradiation and airborne exposures, the potential doses and risks were analyzed for both the on-site worker and nearby resident. The ²³⁸U and ²³²Th concentrations used by EPA (Cont. No. 68–02–4375) were 0.16 Bq/g (4.3 pCi/g) and 0.13 Bq/g (3.5 pCi/g), respectively. The worker was assumed to be exposed only to an active disposal cell with an effective surface area of 1000 m². The external radiation exposure rates were

estimated to be 2 nC/kg-h (8 μ R/h) and 5 nC/kg-h (20 μ R/h) for standing next to and standing on the pile, respectively. In addition, potential doses from exposure to contaminated groundwater and to surface water runoff were evaluated for the nearby resident. The estimated annual EDEs for the coal ash pile worker and on-site resident are summarized in Table 3.2.16.

In 1993 the EPA conducted a dose assessment (EPA, RAE–9232/1–2) that evaluated the potential doses and risks associated with ash disposal. The ²³⁸U, ²³⁵U, and ²³⁴U concentrations in the coal ash were assumed to be 0.12 Bq/g (3.3 pCi/g), 0.007 Bq/g (0.2 pCi/g), and 0.12 Bq/g (3.3 pCi/g), respectively (see Table 3.2.12). The generic ash impoundment was assumed to be 25,000 m² with a depth of 5 meters. The disposal site was located in the Northeast and contained a total of 1.3 million metric tons of ash materials. The annual individual EDE to workers and to on-site and off-site public receptors are summarized in Tables 3.2.13 and 3.2.14, respectively. Annual collective EDEs are summarized in Table 3.2.15.

3.2.3.4.2 Phosphate Slag

3.2.3.4.2.1 Product Use

During 1986 and 1987, the EPA (EPA/520/6-90/008) conducted a study to evaluate direct irradiation exposures and attendant risks to the populations of Pocatello and Soda Springs, Idaho, from the use of phosphate slag in the construction of roads and buildings. Gamma measurements were made using both aerial surveys and ground surveys. These communities each have phosphate ore processing facilities that have operated or are operating. Radioactive slags from these plants have been used as an additive in materials for paving streets and constructing building foundations. In Pocatello, the annual average and maximum individual EDEs are about 0.14 mSv (14 mrem) and 1.5 mSv (150 mrem), respectively. In Soda Springs, the annual average and maximum individual EDEs are about 0.5 mSv (50 mrem) and 2 mSv (200 mrem), respectively (EPA/520/6-90/008). As shown in Table 3.2.17, the annual collective dose equivalents are 8 person-Sv (800 person-rem) and 2 person-Sv (200 person-rem) for Pocatello and Soda Springs, respectively (EPA/520/6-90/008).

3.2.3.4.2.2 Phosphate Slag Disposal

The EPA performed an assessment (RAE–9232/1–2) in which ²³⁸U, ²³⁵U, and ²³⁴U concentrations in phosphate slag were assumed to be 0.9 Bq/g (25 pCi/g), 0.05 Bq/g (1.3 pCi/g), and 0.9 Bq/g (24 pCi/g), respectively (Table 3.2.12). The generic disposal pile is assumed to contain 24 million metric tons of slag. This size is slightly larger than the largest pile at an active facility. Assuming a square surface and a height of 20 meters for the slag pile, the disposal site has a surface area of 630,000 m² (156 acres). The annual individual EDE to workers and to on-site and off-site members of the public are summarized in Tables 3.2.13 and 3.2.14, respectively. Annual collective EDEs are summarized in Table 3.2.15.

3.2.3.4.3 Water Treatment Sludge

The EPA assumed that water treatment sludge is disposed in a sanitary landfill (RAE-9232/1-2). The volume of sludge sent to the landfill over more than 20 years has a volume of 7,500 m³ when diluted by the other landfill wastes. The ²³⁸U, ²³⁵U, and ²³⁴U concentrations are assumed to be 0.004 Bq/g (0.1 pCi/g), 3×10^{-5} Bq/g (7×10^{-4} pCi/g), and

0.004 Bq/g (0.1 pCi/g), respectively (Table 3.2.12). The size of the facility is assumed to be 400 meters by 400 meters and there is a 0.3-meter layer of topsoil placed over the fill area. The annual individual EDEs to workers and to on-site and off-site public receptors are summarized in Tables 3.2.13 and 3.2.14, respectively. Annual collective EDEs to the public are summarized in Table 3.2.15.

3.2.4 Present Exemption Analysis

In the present exemption analyses, annual individual and collective EDEs are estimated for truck drivers transporting zircon flour. To estimate EDEs for routine worker and public exposures from uranium in dental products, the results from previous analyses are used in the current assessment. Annual EDEs are calculated for industrial workers handling bulk zircon flour and involved in sand-blasting operations. Mineral-derived products are of particular interest for this exemption since, as is the case for zircon-derived products, the uranium and thorium concentrations can be close to the 0.05% weight criterion. In the current assessment, doses to other members of the public from the use of ophthalmic glass containing zirconium oxides, tape dispensers containing zircon sand as ballast, and phosphate slag used to pave streets and in building construction are based on previous analyses. Individual and collective doses for the disposal of zircon sand are estimated in the current assessment and the results are compared to previous evaluations. The accident scenarios evaluated include a warehouse fire that contains zircon flour and a fire that occurs during zircon flour transport.

3.2.4.1 Transport

In the transportation component of this current assessment, it is assumed that a single truck driver is transporting 48 pallets of 0.05% by weight of uranium and thorium in zircon flour (8.5 Bq/g (230 pCi/g) natural uranium in equilibrium with short-lived progeny and 0.65 Bq/g (17.5 pCi/g)²³²Th in equilibrium with its progeny). The primary exposure pathway to the truck driver is direct irradiation. It is assumed there are fifty 41-kg bags per pallet. The dimensions of the product on a pallet are $1.2 \text{ m} \times 1.2 \text{ m} \times 0.6 \text{ m}$ (height). The pallets are assumed to be stacked four high, six deep, and two wide. The distance between the truck driver and the load is 0.9 meter and the thickness of the truck body is assumed to be 0.6 cm of steel. Using MicroShield (Computer Codes, Grove Engineering, 1996), the estimated EDE rate is $1 \times 10^{-4} \text{ mSv/h}$ (0.01 mrem/h). If the driver takes 24 hours to transport a load, the EDE per trip is 0.002 mSv (0.2 mrem). If this driver makes 25 trips per year with the same type of load, the annual EDE is estimated to be 0.06 mSv (6 mrem). Further, if 1000 truck drivers make 25 trips annually, the annual collective EDE is estimated to be 0.06 person-Sv (6 person-rem).

3.2.4.2 Routine Exposures

In this subsection, the EDEs from routine worker and public exposures to uranium in dental products, zircon and ilmenite products, and phosphate slags are summarized. For dental products and phosphate slags, the results from previous analyses, as described in Section 3.2.3, are used in the current assessment. As mentioned earlier, mineral-derived products are of particular interest for this exemption since, as is the case for zircon-derived products, the uranium and thorium concentrations can be close to the 0.05% weight criterion.

3.2.4.2.1 Dental Products

NCRP 95 estimated that the annual dose equivalent to the basal mucosa of a dental prostheses wearer from beta particles was 5 mSv (0.5 rem) at 0.02% by weight of uranium. On the basis of a weighting factor of 0.01 for the human skin, and assuming that irradiation of the basal mucosa is equivalent to irradiation of 1% of the skin, the EDE was estimated to be 5×10^{-4} mSv (0.05 mrem) (NCRP 95). NCRP 95 assumed 45 million people were wearing dental prostheses containing uranium. As the average life of the prosthetic device is not expected to exceed 11 years and 14 years have passed since that report, it is reasonable to assume that only 1 million people are still wearing some type of dental prostheses with uranium. The resulting annual collective dose equivalent to the U.S. population from beta particles would be conservatively 0.5 person-Sv (50 person-rem).

Buckley et al. (NUREG/CR–1775) assumed that the annual EDE to an individual due to external irradiation is 0.03 μ Sv (0.003 mrem). Assuming that the same number of porcelain dental prostheses containing uranium (approximately 1 million) are currently worn in the United States, the conservative collective EDE from gamma irradiation is about 0.03 person-Sv (3 person-rem). This is only 6% of the skin EDE.

Therefore, the annual individual EDE from wearing a porcelain dental prosthesis containing the average weight % of uranium is 5×10^{-4} mSv (0.05 mrem). The annual collective EDE due to beta particles and external irradiation is conservatively estimated to be 0.5 person-Sv (50 person-rem). This dose is expected to decrease significantly over time as porcelain without uranium and other types of materials used for dental prostheses replace the old porcelain containing uranium. At the maximum allowed weight %, the annual EDE would be two and one-half times the dose for the dentures of average weight %.

3.2.4.2.2 Mineral-Derived Products

In this assessment, exposures to zircon-derived products are used to represent the upper bound of potential doses from the broad category of mineral-derived products. Zircon-derived products were selected since the uranium and thorium concentrations can be at or close to the 0.05% weight exemption criterion. For the worker exposure scenario, ilmenite as a sand abrasive product is also evaluated.

Two major routes of exposure appear to dominate: dust inhalation and external irradiation. In most situations, dust inhalation is the dominant exposure pathway, especially if the particulate mass loading in air is high. The radioactive content of the dust is predominantly due to the monazite or bastnasite content, which was not completely removed from the other minerals during the separation process (Koperski, 1993). Other sources of radioactivity in the minerals are due to adsorption of radionuclides onto minerals and incorporation of thorium and uranium into the mineral lattice during the crystallization process (Koperski, 1993).

The radon emanation rate measured above a stockpile of zircon sand is about 0.074 Bq/m²-s (2 pCi/m²-s) (Boothe et al., 1980). As explained by Boothe et al., this low emanation rate is because radium in zircon occurs interstitially in the $ZrSiO_4$ crystal, and radon is trapped within the crystalline lattice. Radon emanation rate measurements over dry land have been reported to range from 2×10^{-4} to 0.05 Bq/m²-s (5×10⁻³ to 1.4 pCi/m²-s) (NCRP 103). The average radon emanation rate in background soils is about 0.02 Bq/m²-s (0.5 pCi/m²-s) (NCRP 103).

Therefore, the radon emanation rate associated with zircon sand is marginally higher than expected in normal background soil.

3.2.4.2.2.1 Worker Exposures

Typically, mineral-derived products that have not been highly processed, such as zircon flour and ilmenite used in sand-blasting operations, are more likely to contain elevated concentrations of uranium and thorium as compared to higher purity products (Phone call, J. B. Hedrick, Thorium and Rare Earth Specialist, Bureau of Mines, U.S. Department of the Interior, 1995). As mentioned earlier, the two primary exposure pathways appear to be external irradiation and dust inhalation.

MicroShield (Computer Codes, Grove Engineering, 1996) was used to estimate the EDE rate at 1 meter from 20 pallets of zircon flour containing natural uranium (8.5 Bq/g (230 pCi/g)) in activity equilibrium with its short-lived progeny and ²³²Th (0.65 Bq/g (17.5 pCi/g)) in equilibrium with its progeny. Assuming no shielding, the calculated EDE rate was about 2×10^{-4} mSv/h (0.02 mrem/h). If the exposure duration was 2000 h/yr, at this distance, an unlikely event, the annual EDE from direct irradiation is about 0.4 mSv (40 mrem). The direct gamma-exposure rate measured by Shimko (1996) at about 0.9 meter from a stack of bags was 10 nC/kg-h (0.04 mR/h). In Booth et al. (1980), the direct irradiation exposure rate, correcting for background, measured at a distance of 0.9 meter from a pallet of zircon flour was about 18 nC/kg-h (0.07 mR/h). The differences in the measured exposure rates may be from variations in actual thorium and uranium concentrations in the zircon product and the possible contribution of uranium long-lived decay products.

Shimko (1996) evaluated the worker exposure from the processing of zircon sands in Plant A and Plant B. The respirable dust concentration, 1.4 mg/m³, was measured at Plant A, which blends zircon flour with sodium carbonate to produce zircon carbonate. For this assessment, it is assumed that the product concentration is the same as the airborne zircon flour dust, i.e., natural uranium (8.5 Bq/g (230 pCi/g)) in equilibrium with its short-lived progeny and ²³²Th (0.65 Bq/g (17.5 pCi/g)) in equilibrium with its progeny. With an exposure duration of 2000 hours, the annual EDE is estimated to be 2 mSv (200 mrem). This estimate is potentially low by a factor of about 2 if consideration is given to the presence of the other uranium decay products. Radionuclide air concentrations were measured in Plant B where zircon sand is ground to zircon flour, bagged, and resold. Using measured area (near bagger) airborne radionuclide concentrations (as shown in Table 3.2.9) and assuming an exposure duration of 2000 hours, the estimated annual EDE was about 7 mSv (700 mrem). Using personal sampler data (bagger operator) and assuming an exposure duration of 2000 hours, the estimated annual EDE was about 40 mSv (4 rem). In Shimko (1996), the estimated annual EDE associated with Plant A and Plant B (bagger operator) activities was estimated to be about 3 mSv (300 mrem) and 17 mSv (1700 mrem), respectively, based on calculations using DAC values from 10 CFR 20, including consideration of a 10 micron particle size for uranium. As mentioned in Shimko (1996), dust respirators were worn by the workers in both Plant A and Plant B and the actual exposure duration for the bagger operator (Plant B) was less than that used to estimate the annual EDE.

In Wallace and Leach (1987), the mean dust concentration for sand-blasting operations was 10.4 mg/m³. Radioactivity measured from the high-volume dust samples indicated that 90% of the particles were in the size range of 0.45 to 5 microns, the same particle size ranges collected

by personal air samplers. The maximum measured area dust concentration during sand-blasting operations was 40.3 mg/m³. However, sand blasting operators usually work from an upwind location so the dust is blown away from the work station.

In ilmenite, an abrasive product used in sand blasting, the average ²³⁸U concentration is 0.092 Bq/g (2.5 pCi/g) and the ²³²Th concentration is 0.2 Bq/g (5.5 pCi/g) (Wallace and Leach, 1987). Assuming the average radionuclide concentrations (including ²³⁸ U, ²³⁴U, and ²³⁵U in equilibrium with short-lived progeny and ²³²Th in equilibrium with its progeny) and mean respirable dust concentration (10.4 mg/m³), the annual EDE is estimated to be 3 mSv (300 mrem). This estimated annual EDE is very similar to that determined by Wallace and Leach (1987). Depending on the type of dust respirator used regularly, the annual EDE could be reduced by a factor of 5 to 10 (NIOSH, 1990). It should not be assumed that respirators are always worn by workers handling zircon flour or similar mineral sand products.

The annual EDE from both external irradiation and dust inhalation to an industrial worker handling bulk zircon product is estimated to range from about 3 mSv (300 mrem) to 7 mSv (700 mrem) assuming no respiratory protection. For the bagger operator, the annual EDE was estimated to be about 35 mSv (3500 mrem); however, the actual exposure duration may be less than that used to estimate the EDE (Shimko, 1996). The annual EDE due to the use of ilmenite in sand blasting is estimated to be about 3 mSv (300 mrem). Ilmenite is not the only product that can be used as an abrasive; zircon has been used in the past and low silicate-containing minerals may also be used. As shown, the dominant exposure pathway is dust inhalation when a dispersible product is used.

The handling of bulk mineral-derived products full-time should be considered an upper bound to the potential doses that may occur during the use of products that contain 0.05% by weight of source material. It is unknown how many workers may be handling bulk zircon or other mineral-derived products with uranium and thorium concentrations nearing (or exceeding) the 0.05% weight exemption limit. However, if it is assumed that 1000 workers handle similar types of bulk zircon products annually, the annual collective EDE could range from 3 person-Sv (300 person-rem) to 7 person-Sv (700 person-rem).

3.2.4.2.2.2 Public Exposure

Public exposures to zircon-derived products include the use of ophthalmic glass and exposure to zircon sand ballast for tape dispensers. Due to the potential use and distribution of such products in the public and the elevated concentrations of thorium and uranium, these products are highlighted. Undoubtably many other mineral-derived products are disseminated.

Ophthalmic Glass. It is estimated that the number of people wearing eyeglasses in the United States is 145 million (VCA, 1999). Currently, it has been estimated that only 8%, or about 12 million people, wear eyeglasses that have glass lenses (OA, 1999). Assuming the same annual individual EDE of 0.004 mSv (0.4 mrem) from alpha particles and the annual whole-body dose equivalent of 2×10^{-4} mSv (0.02 mrem), as described by Buckley et al. (NUREG/CR–1775) and in NCRP 95, the annual collective EDE to the U.S. population that currently wears glass-lensed eyewear would be about 50 person-Sv (5,000 person-rem).

Tape Dispenser Ballast. In Boothe et al. (1980) and CRCPD (1981), the measured exposure rate at 0.6 meter from tape dispensers was about 77 pC/kg-h (0.3 μ R/h) above background.

Assuming an individual spent 80% of a 40-hour work week seated within 0.6 meter of a tape dispenser, the annual EDE is estimated to be 0.005 mSv (0.5 mrem). Based on available information, only one to two known manufacturers used zirconium sands as tape dispenser ballast (CRCPD, 1981). The manufacturer of the model tested no longer uses radioactive sands for ballast (CRCPD, 1981). Assuming 10,000 people use such tape dispensers (limited distribution), the estimated collective EDE is estimated to be 0.05 person-Sv (5 person-rem).

3.2.4.2.3 Phosphate Slag

The results from the EPA (EPA/520/6-90/008) study that evaluated gamma exposures and attendant risks to the populations of Pocatello and Soda Springs, Idaho, from the use of phosphate slag are used in the current analysis. The average annual individual and collective EDEs attributed to paving streets and constructing building foundations in the communities of Soda Springs and Pocatello, Idaho, are summarized in Table 3.2.17. The collective EDE is based on a 10-year useful lifetime. Idaho and Montana have both prohibited the use of phosphate slag in structures but continue to permit its use in highway and other construction (CRCPD, 1981).

3.2.4.3 Disposal

The disposal of zircon sand in municipal waste landfills is the primary focus of the current assessment. The generic disposal methodology, as described in Appendix A.2, is used to estimate annual individual and collective EDEs to landfill workers and other members of the public from the disposal of zircon sand. For the waste collectors, it was assumed that the waste zircon sand was transported in an off-loader with 15.3 m³ capacity to the landfill and the source-to-receptor distance was 1 meter. The trailer dimensions are 6.7 m \times 2.4 m \times 1.4 m and the weight capacity is 10 metric tons (Phone call, Vicki Esponge, Customer Service, Waste Management Incorporated, Knoxville, TN, May 1997).

In Oregon, foundries disposed of about 4180 metric tons of zircon each year (Boothe et al., 1980). For this analysis, it is assumed this amount of zircon is disposed in each State per year, resulting in a total amount of 2.1×10⁵ metric tons of zircon disposed annually in the United States. It is also assumed the zircon sand is uniformly distributed among 3,500 operating U.S. municipal landfills, which results in about 60 metric tons of zircon sand disposed of annually in a single landfill. The zircon sand contains 0.05% by weight of source material (natural uranium 8.5 Bq/g (230 pCi/g) in equilibrium with its short-lived progeny and ²³²Th 0.65 Bq/g (17.5 pCi/g) in equilibrium with its progeny). The estimated annual individual and collective EDEs for collectors, landfill operators, off-site receptors, and future on-site receptors from disposal of zircon sand are summarized in Table 3.2.18.

There are major differences between the assumptions and methods used in the present assessment as compared to those used in EPA RAE–9232/1–2). In the current assessment, ²³²Th and its decay products and only ²³⁸U, ²³⁴U, and ²³⁵U and their short-lived decay products are taken into account, whereas in EPA (RAE–9232/1–2), ²²⁶Ra and its decay products are also considered. The present assessment evaluates the disposal of zircon sand product disposed in municipal landfills, whereas EPA (RAE–9232/1–2) evaluates the disposal of the tailings and residues associated with the processing of zircon sands. Other differences include radionuclide concentrations, amount of commingled waste, waste site characteristics, impacted populations, and certain exposure assessment parameter values.

3.2.4.4 Accidents

In this evaluation, the warehouse and transportation accident scenarios were selected to represent upper bound doses associated with a product that contains 0.05% by weight of uranium and thorium. It is difficult to image a fire causing a significant fraction of the zircon flour to become airborne; however, firefighting activities could disperse the material and cleanup workers/investigators could encounter contamination. It is assumed 1000 zircon flour 41-kg bags are stored in a warehouse or are being transported. The uranium concentration is about 8.5 Bq/g (30 pCi/g) and the thorium concentration is about 0.65 Bq/g (17.5 pCi/g). Using the generic accident methodology described in Appendix A.1, the EDE to a firefighter combating a warehouse fire is 0.004 mSv (0.4 mrem) and the EDE to a firefighter combating a transportation fire is estimated to be 0.03 mSv (3 mrem). EDEs for workers involved in cleanup from the warehouse and transportation accidents were estimated to be 0.03 mSv (3 mrem) and 0.3 mSv (30 mrem), respectively. Dose estimates obtained by Buckley et al. (NUREG/CR–1775) for the accident and misuse scenarios of dental products are used in the present analyses. These dose estimates are summarized in Section 3.2.3.1.

3.2.5 Summary

This exemption limits the involvement of the NRC in the regulation of source material. It exempts from licensing many naturally occurring and technologically enhanced sources of uranium and thorium considered nonnuclear. This exemption potentially can cover a wide range of products and materials. In this analysis, the types of product evaluated were divided primarily into two major categories: mineral-derived products and waste-derived products. Uranium in dental products did not fit into either category. Therefore, it was evaluated separately. Zircon products were used to represent the upper bound of doses that could be attributed to mineral-derived products. Phosphate slag in building and road construction materials were used to evaluate the doses from waste-derived products. Evaluation of doses from landfill disposal focused on disposal of zircon sands in municipal waste landfills.

In this analysis, estimates of individual and collective doses to members of the public from routine use of consumer products were based primarily on available published information. Doses from transport, routine uses, disposal, and accidents involving zircon flour and sand were estimated for this analysis. The estimated doses are intended to provide reasonable upper bounds for current practices. The results of this current assessment are summarized in Table 3.2.19.

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

- Uranium is no longer used in U.S.-produced porcelain dental products. It is considered very unlikely that uranium will be used for this application in the future.
- The primary exposure pathways for the public appear to be due to dust inhalation and direct irradiation. Radon emanation from zircon products and byproducts appears to be very low. As explained by Boothe et al. (1980) and Koperski (1993), the low emanation rate is because radium in zircon occurs interstitially and the radon is trapped within the

crystalline lattice. Chemical processing of mineral sands can alter this configuration and may result in radon release.

- Doses to industrial workers involved in routine bulk product (e.g., zircon flour) handling operations are greater than those from transport or disposal. It is unknown how many U.S. industrial workers may be involved with the handling of bulk mineral-derived products and whether respiratory protection is routinely used. Industrial workers not handling bulk (dispersible) products routinely are expected to have lower doses than those working in dusty environments.
- The use of ophthalmic glass in eyeglasses is declining as plastic lenses are replacing glass. Tape dispensers that used zircon sand as ballast are no longer manufactured. It is unknown whether there are other products that may result in higher doses to the public.



Figure 3.2.1 Heavy Mineral Concentrate Process Schematic ¹⁰

...

¹⁰ Hewson and Upton, 1996

	Specific Activity —	Radionuclide (500	Concentration ppm)
Radionuclide	(Ci/g)°	(pCi/g)	(Bq/g)
Natural uranium (²³⁸ U+ ²³⁵ U+ ²³⁴ U)	6.8×10 ⁻⁷	340	13
Depleted uranium (²³⁸ U+ ²³⁵ U+ ²³⁴ U)	3.6×10 ⁻⁷	180	7
²³⁸ U	3.3×10⁻ ⁷	165	6
Natural/processed thorium (²³² Th + ²²⁸ Th)	2.2×10 ⁻⁷	110	4
²³² Th	1.1×10 ⁻⁷	55	2

Table 3.2.1 Source Material Concentrations Associated With 0.05 Percent Weight of Source ^{a,b}

^a Conversions: Natural uranium at 250 Bq/g (6800 pCi/g) per % weight to 0.025 Bq/g (0.68 pCi/g) per ppm. Depleted uranium at 130 Bq/g (3600 pCi/g) per % weight to 0.013 Bq/g (0.36 pCi/g) per ppm. Natural and processed thorium at 40 Bq/g (1100 pCi/g) per % weight to 0.004 Bq/g (0.11 pCi/g) per ppm.

^b Conversion of weight % to radionuclide concentration does not include decay products; however, decay products are included in the dose calculations.

° 1 Ci/g = 0.037 TBq/g.

Mineral	Mineral or Waste Radioactivity
AT OR ABOVE	THE 0.05 PERCENT WEIGHT CRITERION [®]
Monazite Niobium (columbium)-Tantalum	 Thorium 57% and uranium 0.10.3% Imported ores for production Tin (thorium 0.4-4%) pyrochlore ore (thorium oxide 13.3%, uranium oxide 3.1%) euxenite ore Canada (0.350.54% uranium, 2.54% thorium)
AT OR BELOW	THE 0.05 PERCENT WEIGHT CRITERION
Aluminum	 Bauxite ore: 0.25 Bq/g (6.8 pCi/g) ²³⁸U, 0.3 Bq/g (8.2 pCi/g) ²³²Th, and 0.26-0.29 Bq/g (7-8 pCi/g) ²²⁸Th Red mud slurry: 0.52 Bq/g (14 pCi/g) ²³⁸U
Coal	 U.S. average concentrations: 4.7 ppm thorium and 1.8 ppm uranium anthracite 5.4 ppm thorium and 1.5 ppm uranium bituminous 5 ppm thorium and 1.9 ppm uranium lignite 6.3 ppm thorium and 2.5 ppm uranium Coal ash: 0.06-0.32 Bq/g (1.5-8.6 pCi/g) ²³⁸U, 0.01-0.28 Bq/g (0.4-7.5 pCi/g) ²³²Th Fly ash: 0.2 Bq/g (5.4 pCi/g) ²³⁹U and 0.07 Bq/g (1.9 pCi/g) ²³²Th
Copper	 1–100 ppm uranium in copper ores 36 pCi/g (1.3 Bq/g) ²³⁸U Arizona Miami District, 0.016% mean (U₃O₈)
Phosphate	 ²³⁸U 20–200 ppm (0.26–2.5 Bq/g (7–67 pCi/g)) ²³²Th 1–5 ppm (0.003–0.02 Bq/g (0.1–0.6 pCi/g))

Table 3.2.2 Naturally Occurring Radioactivity Related to Mineral Resources ^a

Table 3.2.2 Naturally Occurring Radioactivi	ty Related to Mineral Resources * (continued)
---	---

Mineral	Mineral or Waste Radioactivity
AT OR BELOW	THE 0.05 PERCENT WEIGHT CRITERION
Titanium-bearing mineral sands	 Rutile: thorium <50–350 ppm, uranium <10–20 ppm Leucoxene: thorium 80–700 ppm, uranium 20–50 ppm Ilmenite: thorium 50–500 ppm, uranium <10–30 ppm Sludge ponds: 2.15 and 1.0 Bq/g (58 and 27 pCi/g) ²³⁸U and ²³⁴U, respectively
Zircon mineral sands	 Thorium 150–250 ppm Uranium 150–300 ppm baddeleyite up to 7.4 Bq/g (200 pCi/g) uranium radionuclides

^a NCRP 118; EPA, RAE–9232/1–2; CRCPD, 1981; CRCPD, 1994; Koperski, 1993; McBride et al., 1978; and Corbett, 1983.

^b Conversions: Natural uranium at 250 Bq/g (6800 pCi/g) per % weight to 0.025 Bq/g (0.68 pCi/g) per ppm. Depleted uranium at 130 Bq/g (3600 pCi/g) per % weight to 0.013 Bq/g (0.36 pCi/g) per ppm. Natural and processed thorium at 40 Bq/g (1100 pCi/g) per % weight to 0.004 Bq/g (0.11 pCi/g) per ppm.

Material	Uranium Concentration pCi/g (ppm) ^b	Thorium Concentration pCi/g (ppm) ^b	
Granite	1.7 (4.7)	0.2 (2)	
Cement	1.2 (3.4)	0.6 (5)	
Byproduct gypsum	5 (13.7)	1.8 (16)	
Dry wallboard	0.4 (1.0)	0.3 (3)	

Table 3.2.3 Uranium and Thorium Concentrations in Common Building Materials ^a

^a NCRP 94.

....

^b 1 pCi/g = 0.037 Bq/g. Conversions: Natural uranium at 250 Bq/g (6800 pCi/g) per % weight to 0.025 Bq/g (0.68 pCi/g) per ppm. Depleted uranium at 130 Bq/g (3600 pCi/g) per % weight to 0.013 Bq/g (0.36 pCi/g) per ppm. Natural and processed thorium at 40 Bq/g (1100 pCi/g) per % weight to 0.004 Bq/g (0.11 pCi/g) per ppm.

Mineral	Products (Intermediate)	Products (Final)
Bauxite and aluminum	Alumina-feedstock, aluminum reduction industry	Aluminum products
Rare earths		Catalysts, ceramics, refractory and metallurgical processes, magnets, lighting, phosphors, glass and optics, electronics
Copper		40% building and construction industries 25% electrical and electronic industries 38% industrial machinery and equipment, power industry, and transportation industry
Zinc		45% automobile industry-galvanized sheet metal 20% brass manufacturing 15% die casting
Titanium	Titanium tetroxide: titanium dioxide + titanium sponge	Titanium dioxide—pigment in paper, paint, plastics, cosmetics, and ceramics
	Rutile: Titanium metal	Titanium sponge—aircraft engines and airframes. Titanium metal—aerospace industry aircraft frames, and jet engines
Zirconium and hafnium	Byproduct of mining and extracting titanium minerals, ilmenite and rutile	Zircon: High production included refractory bricks and shapes, alumina-zirconia abrasives, foundry sands, and investment casting, milled and micronized zircon, zirconium chemicals, and zirconia
	Sponge metal and crystal bar	High-value products are cubic zirconia, technical zirconia ceramics, superalloy castings, zirconia textile refractories, and specialty chemicals
		Zirconium (zircon)—foundry sands, refractory paints, and other refractory materials.
		Hafnium—nuclear reactor control rods
Phosphate	Elemental phosphorus, phosphoric acid, phosphoric s	Fertilizers and soil conditioners, backfill and road-base materials, additives to concrete block, mine reclamation, and chemical feedstock, e.g., sulfur recovery

Table 3.2.4	Examples of	f Mineral-Derived	Products
-------------	-------------	-------------------	----------

...,

Process	Description	Byproduct
Beneficiation	Crushed ores are concentrated to free valuable mineral and metal components from the matrix of less valuable rock (called gangue). Beneficiation processes can include physical and chemical separation techniques, such as gravity concentration, magnetic separation, electrostatic separation, flotation, ion exchange, solvent extraction, electroplating, precipitation, and amalgamation.	Nearly all beneficiation processes generate tailings, which can be considered waste material or can undergo further mineral extraction.
Milling	Concentration of mineral ore further by physical or chemical processing. Titanium: Chlorination in a fluidized-bed reactor-feedstock for production of titanium tetrachloride and titanium sponge.	Tailings.
Smelter and refining	The mineral concentrates, such as copper concentrate, are refined in a smelter. The smelting stage may include roasting, smelting, and converting. Smelting may consist of a pyrometallurgical technique or electrolytic production technique. Other processes include plasma fusion and electric-arc techniques, e.g., zirconium oxide is produced from zircon.	Slag, bag house dust.

Table 3.2.5 Mineral Extraction Processing Methods

Type of Coal	Coal-Uranium Concentration μ g/g (ppm)	Ash-Uranium Concentration µg/g (ppm)
Anthracite	1.5	9
Eastern bituminous	1.9	38
Western bituminous	1.9	16
Lignite	2.3	22

Table 3.2.6 Uranium in Coal and Ash ^a

. . .

^a NCRP 77.

i....

-	Radionuclide Concentration ^a (pCi/g) ^b						
Material	²³⁸ U	²³⁴ U	²³⁵ U	²²⁶ Ra	²³⁰ Th	²³² Th	²²⁸ Th
Phosphate ore	3	3 3		31	5	27	36
Phosphate slag	25	24	1.3	35	32	0.8	0.8
Phospho- gypsum	6	6	0.3	33	13	0.3	1.4
Background soil	0.3	0.3		0.5	0.3	0.5	0.7

Table 3.2.7 Typical Radionuclide Concentrations in Phosphate Ore,Phosphate Slag, and Phosphogypsum

^a EPA, RAE--9232/1--2. ^b 1 pCi/g = 0.037 Bq/g

...

		Annual Dose		We want to a second
Product	Concentrations	Equivalent	Comments	Reference
Porcelain teeth	0.10% weight	6 Sv (600 rem) 0.028 Sv (2.8 rem)	Oral mucosa, α Oral mucosa, β	O'Riordan and Hunt (1974)
	0.05% weight (500 ppm)	4 Sv (400 rem/yr)	Teeth surface	Papastefanou (1987)
Dental products	0.044% (maximum concentration)	1.3 Sv (130 rem) 0.016 Sv (1.6 rem)	Oral mucosa, α β dose rates	Thompson (1976)
	0.02% weight average uranium concentration in U.S. dental	7 mSv (0.7 rem)	Basal cell layer from β (200 μm depth) (U/K-40)	(1976) Thompson n (1976))) s Sairenji et al. (1980) t
	porcelain	5 mSv (0.5 rem)	Basal cell layer from β particles (U)	
Dental porcelains	Shofu–Ace 5.2 ppm Shofu–Real 342 ppm G.C. Livdent 47 ppm Trubyte Bioblend 235 ppm		Higher and more uniform concentration in U.S. product as compared to Japanese	Sairenji et al. (1980)
Dental powders	345–1,090 ppm uranium powders used in superficial part of crown 1.6–2.7 ppm uranium used in core part and top of crown		Determination of 15 elements used in dental porcelain powders sold in Japan	Noguchi et al. (1980)
Dental porcelains	0.05% weight	1–4 Sv (100–400 rem)	Teeth surface, α	NUREG/CR-1775
		14 mSv (1.4 rem)	Basal cell layer, β	
		0.003 μSv (0.3 μrem)	Direct radiation, full dentures	
		0.0005 μSv (0.05 μrem)	Direct radiation, few crowns	

Table 3.2.8 Uranium Concentrations in Dental Products and Estimated Doses

Air Concentrations (µCi/cm³) ^b						······································
Location	²³⁸ U	²³⁵ U	²³⁴ U	²³² Th	²³⁰ Th	²²⁸ Th
Personnel Area on west side of room	1.3×10 ⁻¹² 1.4×10 ⁻¹³	6.0×10 ⁻¹³	1.9×10 ⁻¹² 1.6×10 ⁻¹³	1.4×10 ⁻¹³ 2.3×10 ⁻¹⁴	2.0×10 ⁻¹² 1.1×10 ⁻¹³	5.0×10 ⁻¹³ 3.5×10 ⁻¹⁴
Area near bagger	2.7×10 ⁻¹³		2.7×10 ⁻¹³	6.5×10 ⁻¹⁴	2.2×10 ⁻¹³	1.0×10 ⁻¹³

Table 3.2.9 Measured Air Concentrations at Plant B *

...

.

^a Shimko, 1996. ^b 1 μ Ci/cm³ = 3.7×10⁴ Bq/cm³.

	Activity Concentrations (pCi/g) ^b			
Use	²³² Th	²²⁸ Th	²³⁸ U	²³⁰ Th
Refractories Company 1 Company 2	27 27	54 54	270 270	270 270
Zirconia	27	240	135	27
Glazes	11	135	27	540
Glass	81	135	27	27
Casting	14	14	14	15
Rare earth	270	270	0	0
Special alloys	190	190	160	160
<u>Tin smelting</u> Ore Fumes	27 0	27 0	8 0	8 0
<u>Titanium</u> Ilmenite Rutile Chlorinator bed	27 5 270	27 5 270	14 5 140	14 5 140
Phosphate	0	0	41	41

Table 3.2.10 Typical Activity Concentrations of Materials in Use ^a

^a Hipkin and Paynter, 1991. ^b 1 pCi/g = 0.037 Bq/g.

.

1

Use	Annual Effective Dose Equivalent Due to External Irradiation (mrem) ^b	Annual Effective Dose Equivalent Due to Dust Inhalation (no respiratory protection) (mrem) ^b	Annual Effective Dose Equivalent Due to Dust Inhalation (respiratory protection) (mrem) ^b
Refractories	30	600	60
Zirconia	70	1,500	150
Glazes	70	400	40
Glass	30	500	50
Casting	0	10	<10
Special alloys	50	300	30
<u>Titanium</u> Ilmenite Rutile Chlorinator bed Scale	6 2 0 400	1,500 20 300 0	150 <1 30 0
Phosphate			
Rock Scale	4 400	30 0	<10 0

Table 3.2.11 Individual Exposures: External Irradiation and Dust Inhalation Exposure Routes ^a

^a Hipkin and Paynter, 1991. ^b 1 mrem = 0.01 mSv.

		Radior	nuclide Concer	ntration	
-			(pCi/g)°		
Type of Waste	²³⁸ U	²³⁴ U	²³⁵ U	²³² Th	²²⁸ Th
Special application metals ^b	43	43	2.2	22	22
Coal ash	3.3	3.3	0.2	2.1	3.2
Phosphate slag	25	24	1.3	0.8	0.8
Water treatment sludge	0.1	0.1	7×10⁻⁴	0.005	0.2

Table 3.2.12 Radionuclide Concentrations Used in Storage and Disposal Assessment *

···.

^a EPA, RAE–9232/1–2.
^b Special application metals include zirconium, hafnium, titanium, and tin.
^c 1 pCi/g = 0.037 Bq/g.

Exposure Scenario	Annual Effective Dose Equivalent (mrem) ^b			
	Special Application Metals ^c	Phosphate Slag	Coal Ash	Water Treatment Sludges
Gamma irradiation	2×10 ²	1×10 ²	1×10 ¹	2
Dust inhalation	3	0.4	0.06	7×10⁻⁵

Table 3.2.13 Individual Effective Dose Equivalent to Workers From Storageor Disposal of Wastes *

^a EPA, RAE-9232/1-2.
^b 1 mrem = 0.01 mSv.
^c Special application metals include zirconium, hafnium, titanium, and tin.

		t		
Exposure Scenario	Special Application Metals	Phosphate Slag	Coal Ash	Water Treatment Sludges
		I-SITE INDIVIDUA	L	
Gamma irradiation	3×10⁻⁴	400	40	0.08
Dust inhalation	3	4×10 ⁻¹	6×10 ⁻²	0.007
Total	3	400	40	0.09
	OFI	F-SITE INDIVIDUA	L	
Gamma radiation	7×10 ⁻⁵	100	10	0.02
Dust inhalation	3×10⁻⁴	0.3	0.03	8×10 ⁻⁴
Drink-contaminated well water	0.008	5×10 ⁻⁴	0.04	4×10 ⁻⁶
Food-contaminated well water	0.01	4×10 ⁻⁴	0.1	6×10⁻⁵
Total	0.02	100	10	0.02

Table 3.2.14 Individual Effective Dose Equivalent to Members of the Public FromStorage or Disposal of Wastes a

^a EPA, RAE–9232/1–2. ^b 1 mrem = 0.01 mSv.

...

	Annual Collective Dose Equivalent (person-rem)°			
Exposure Scenario	Special Application Metals	Phosphate Slag	Coal Ash	Water Treatment Sludges
Exposure to resuspended particulates	6×10⁻⁵	0.03	0.2	7×10⁻⁴
River water contaminated by groundwater	3×10⁻⁴	6×10⁻⁴	0.003	3×10⁻⁵
Total	4×10 ⁻⁴	0.03	0.2	7×10⁻⁴

Table 3.2.15 Collective Effective Dose Equivalents From Storage or Disposal of Wastes ^{a,b}

^a EPA, RAE-9232/1-2.

^b The population densities used in the estimation of the collective EDEs are as follows. For special application metals, the population density is assumed to be 65 persons per square mile. The population density assumed for phosphate slag disposal is the average value for Idaho, which is approximately 12 persons per square mile; for coal ash disposal, the population density was based on the average population distribution of four Northeastern states: New Jersey, Connecticut, Rhode Island, and Massachusetts. For water treatment sludges, the population density is assumed to be 206 persons per square mile, which is based on the Illinois population census.

^c 1 person-rem = 0.01 person-Sv.

Exposure Parameter	Annual Effective Dose Equivalent (mrem) ^b	Collective Effective Dose Equivalent (person-rem)°
	WORKER	······································
<u>Direct irradiation</u> Standing on the pile Standing near the pile	33 14	
Particulate emissions	0.007	
Radon emissions	0.005	
Total	47	
	NEARBY RESIDENT	
Direct irradiation	0.6	· · · · · · · · · · · · · · · · · · ·
Particulate emissions	0.02	
Radon emissions	0.003	
Groundwater	0	
Surface water	0.7	
Total	1.3	
	REUSE	
Direct irradiation House foundation	17	510

Table 3.2.16 Estimated Doses From Exposures to a Coal Ash Pile and Coal Ash Product ^a

^a EPA, Cont. No. 68–02–4375. ^b 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

^c Annual collective effective dose equivalent from reuse is estimated assuming that 4 people live in a house and 7,500 houses are constructed with concrete using 1 year's production of coal ash; therefore, 30,000 persons could be exposed.

Table 3.2.17 Gamma Irradiation Dose of Individuals and Populations of Pocatello and Soda Springs, ID *

Community	Average Annual Individual Dose (mrem/yr) ^ь	Maximum Annual Individual Dose (mrem/yr)⁵	Annual Population Dose (person-rem) ^ь
Pocatello	14	150	800
Soda Springs	50	200	200

^a EPA/520/6-90/008.

•••

^b 1 mrem/yr = 0.037 mSv/yr; 1 person-rem = 0.01 person-Sv.

Note: Radionuclide concentrations in Idaho phosphate slag (EPA, RAE–9232/1–2): 1.5 Bq/g (41 pCi/g) 238 U, 1.48 Bq/g (40 pCi/g) 234 U, 1.8 Bq/g (48 pCi/g) 226 Ra, and 0.02 Bq/g (0.5 pCi/g) 232 Th.

Exposure Scenario	Individual Annual Effective Dose Equivalent ^b (mrem) ^d	Annual Collective Effective Dose Equivalent ^c (person-rem) ^d
Waste disposal worker		
Collector ^e	5	20
Landfill operator	2	30
Off-site resident		
Airborne releases	0.04	20
Groundwater releases	0.02	90
Future on-site resident ^f	10	1×10 ⁴

Table 3.2.18 Individual and Collective Effective Dose Equivalents Attributed to the Disposal of Zircon Sands ^a

^a Natural uranium concentration is assumed to be about 8.5 Bq/g (230 pCi/g) and the ²³²Th concentration is 0.65 Bq/g (17.5 pCi/g). The short-lived decay products of ²³⁸U and ²³⁵U are assumed to be present and in activity equilibrium. The decay products of ²³²Th are assumed to be present and in activity equilibrium. See Appendix A.2 for a description of the generic disposal methodology used in this analysis.

^b For the individual annual effective dose equivalent (EDE), it is assumed that there is about 518 MBq (14 mCi) of natural uranium and of 74 Bq (2 mCi) of ²³²Th are disposed in one landfill. It is assumed that 60 metric tons of zircon sand are disposed annually in one landfill. See Section 3.2.4.3 for further information.

^c For the collective EDE, it is assumed that 1850 GBq (50 Ci) of natural uranium and 259 GBq (7 Ci) of ²³²Th are disposed annually in 3,500 landfills. The exposed populations are as follows: 3,500 waste collectors, 17,500 landfill operators, 35,000 future residents, and 700,000 people that drink groundwater. It is assumed 500,000 people reside within 80 km of each landfill. ^d 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

^e It is assumed the waste is transported to the landfill in an off-load vehicle. The volume capacity is 15 m³ and the weight capacity is 10 metric tons.

¹ The annual individual and collective dose equivalent for the future on-site resident takes into account exposure to ²²⁰Rn and its short-lived decay products during indoor residence. However, ²²²Rn is not taken into account since ²³⁸U short-lived decay products only are considered.

Activities and Receptors	Individual Annual Effective Dose Equivalent (mrem) ^b	Collective Effective Dose Equivalent ^a (person-rem) ^b	Percent Weight
	TRANSPC	PRT	
Truck driver	6	5	0.05
	ROUTINE U	SE ^{c,d}	
<u>Worker</u> Zircon flour handling Sandblasting	300–4,000 300	300–700	0.05 0.006
Public Dental products Zircon-derived products	0.05	50	0.05
- Tape dispenser Phosphate slag - Pavement and building	0.5	5 2 000-8 000	0.05
construction			
L	ANDFILL DISPOSAL-	ZIRCON SAND [®]	
<u>Workers</u> Collectors Landfill operators	5 2	20 30	0.05 0.05
Public Off-site resident	0.04	20	0.05
- Groundwater releases Future on-site resident	0.04 0.02 10	90 10,000	0.05 0.05 0.05
	ACCIDEN	T ¹	
<u>Dental products</u> Warehouse Dental technician	6 0.002		0.05 0.05
Zircon flour Warehouse fire - Firefighter	0.4		0.05
- Cleanup worker Transportation fire	3		0.05
- Cleanup worker	30		0.05

Table 3.2.19 Summary of Effective Dose Equivalents According to Percent Weight

See following page for footnotes.

-- .

Footnotes to Table 3.2.19

^a The population sizes used in estimation of the collective effective dose equivalent (EDE) are as follows: 1,000 drivers, zircon flour handlers, and tape dispenser users were assumed. The population assumed for dental products containing uranium was 1 million, and 12 million was assumed for ophthalmic glass wearers. The collective EDE from phosphate slag use was based on the population of Soda Springs and Pocatello, ID.

^b 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

^c In zircon flour and sand, the ²³⁸U concentration is estimated to be 8.5 Bq/g (230 pCi/g) and the ²³²Th concentration is estimated to be 0.65 Bq/g (17.5 pCi/g), whereas in ilmenite, the abrasive used in sand blasting, the ²³⁸U concentration is 93 Bq/g (2.5 pCi/g) and the ²³²Th concentration is 0.2 Bq/g (5.5 pCi/g). Note that the respirable dust concentration for the zircon product handling was 1.4 mg/m³, whereas the average dust concentration for sand blasting was 10.4mg/m³ and 90% of the dust was considered to be respirable particulate.

^d Refer to Section 3.2.4.2 for further information on the estimation of annual individual and collective dose equivalent attributed to routine operations.

^e Refer to Section 3.2.4.3 for further information on the estimation of annual individual and collective dose equivalent attributed to disposal. Also see Appendix A.2 for a description of the generic disposal methodology.

^f Refer to Section 3.2.4.4 for further information on the estimation of the individual dose equivalent attributed to accidents. Also see Appendix A.1 for a description of the generic accident methodology.

3.3 Unrefined and Unprocessed Ore Containing Source Material

3.3.1 Introduction

In 10 CFR 40.13 (b), any person is exempt from the requirements for a license, to the extent that the person receives, possesses, uses, or transfers unrefined and unprocessed ore containing source material. Unless authorized in a specific license, a person shall not refine or process such ore. This exemption was proposed on September 7, 1960 (25 FR 8619), and issued as a final rule on January 14, 1961 (26 FR 284).

The *Federal Register* notices cited above do not contain any information on analyses of radiological impacts on the miner or public from transfer of unrefined and unprocessed ore containing source material. No indication was given that a dose analysis had been performed to support the exemption.

Since source material pertains to both uranium and thorium, the transport and possession of uranium and thorium ore are evaluated in the current assessment. This exemption may also apply to the transfer and possession of other ores containing source material.

3.3.2 Description of Items

Ore is defined in the Bureau of Mines Dictionary as a natural mineral compound of the elements of which at least one is a metal (Bureau of Mines, 1968). In 10 CFR 40.4, unrefined and unprocessed ore is defined as "ore in its natural form prior to any processing, such as grinding, roasting or beneficiating, or refining." Based on these definitions, this exemption applies to an ore that contains a metal that has not been either physically or chemically altered.

The NRC licensing requirement for source material begins "after removal from its place of deposit in nature" (10 CFR 40.3). Therefore, NRC does not regulate mining of ores regardless of their source material content. Ores containing <0.05% by weight uranium, thorium or any combination thereof are not source material by definition (40 CFR 40.4). The unrefined and unprocessed ore exemption, therefore, applies to such ore subsequent to the mining activity. Thus, abandoned mines are not subject to regulation by the NRC. Also, the exemption does not apply to in-situ mining or uranium because the ore is "processed" through chemical treatment underground.

The exemption explicitly pertains to the receipt, possession, use, or transfer of unrefined ore that contains source material. In the notice of proposed rulemaking (25 FR 8619), it was stated that prior to the proposed regulation, "miners are required to have a license to transfer source material after it was mined. Under the proposed regulation ... the possession and transfer of unrefined and unprocessed ores containing source material would be exempted." Based on this text in the *Federal Register* (25 FR 8619) and discussions with knowledgeable individuals, the original application of this exemption was for possession of uranium ore by miners and for transfer to an ore-buying station or directly to the milling facility (Phone call, K. Weaver, Radiation Programs, Colorado State Health Department, Denver, CO, October 1996; phone call, C. Cain, Region IV, Nuclear Regulatory Commission (NRC), Arlington, TX, October 1996). After the ore is mined at either surface or underground mines, it may be stored on-site and then transported to an ore-buying station or taken directly to a milling facility.

Once the ore is delivered to a buying station or mill, it is considered to be an integral part of the milling operation and subject to controls to meet the NRC license requirements, including 10 CFR Part 20.

Currently, no underground or surface uranium mines are producing ore (Phone call, K. Sweeney, National Mining Association, Washington DC, July 1997). Some of the open-pit mines are being held on standby status, while others have closed permanently and either been reclaimed or abandoned. The dominant type of uranium mining method is in situ mining, which does not require the transport of ore from the mine to the mill. There are three operating in situ mines, and it is projected there will be four additional sites in fiscal year 1998 (Phone call, D. Gillan, Uranium Recovery Branch, NRC, Rockville, MD, July 1997). In situ mining involves the injection of a leaching solution (lixiviant) into the uranium-bearing strata to extract uranium. In this process, the uranium is extracted from the lixiviant by ion-exchange. The ion-exchange resin is stripped of uranium, which is then precipitated to produce a yellow cake slurry.

Monazite, a rare earth and thorium phosphate mineral, is the primary source of thorium (Hedrick, 1997). Domestic mine production of thorium-bearing monazite ceased at the end of 1994 as the world demand for thorium-bearing ores remained depressed (Hedrick, 1997). In prior years, monazite had been recovered by dredging methods. Thorium production was a byproduct during the processing for titanium and zirconium minerals and monazite was recovered for its rare earth content (Hedrick, 1997). Mined mineral sands contain very low concentrations of thorium, ranging from 0.0012 to 0.005% by weight. After the initial wet gravity separation process, which occurs at the mining site, the mineral sand concentrate usually contains less than 0.05% by weight of thorium (Hewson, 1990). The mineral sands are further processed via physical processes (e.g., gravity and electromagnetic processes) to produce a monazite product that ranges from 4 to 6% by weight of thorium.

3.3.3 Summary of Previous Analyses and Assessments

The *Federal Register* notices cited above do not contain any information on analyses of radiological impacts on the public from the use of unrefined and unprocessed ore containing source material. Since the exemption was established, a number of studies have described the emissions and doses associated with uranium mining and milling operations. In these studies, off-site air concentrations and potential doses due to ore transport and storage activities were usually grouped with other mine or mill operations. A number of studies have also been conducted on worker exposures associated with the mineral sands processing industry in which monazite is a byproduct. As with uranium mining and milling, workers involved explicitly in the transport and storage of mineral sands are not identified and are grouped with other workers. However, relevant information from these reports is summarized below.

3.3.3.1 Uranium Mining and Milling Operations

In NUREG–0706, one of the accident scenarios evaluated was a truck accident transporting uranium ore from the mine to the mill. It is assumed that the uranium ore is shipped to ore stockpiles in 23-metric ton (MT) batches and the average distance from the mine to the mill is 50 km. The ore was assumed to contain 1% respirable dust by weight. In the accident scenario, all of the dust was released and available for dispersal. An environmental release factor of 9×10^{-3} was used. Based on these assumptions, the quantity of respirable ore
released to the environment in the event of a truck accident was estimated to be 2.1 kilogram (kg). The individual lung dose commitment at 500 meters was 1.3 mSv (130 mrem), and 0.14 mSv (14 mrem) at 2000 meters from the accident scene (NUREG-0706).

The National Park Service abandoned mine land inventory shows 42 abandoned uranium mine sites within units of the National Park System and 2 immediately adjacent to park boundaries (Phone call, J. E. Burghardt, Geologic Resources Division, National Park Service, Lakewood, CO, July 1997). The Orphan uranium mine, which produced high-grade uranium ore, is now abandoned and its surface facilities are along the West Rim Trail near the Grand Canyon Village (Burghardt, 1995). The surface facilities are connected to the major underground mine by a 488-meter compartment vertical shaft. The shaft's headframe, cages, hoists, as well as air compressor building, numerous building foundations, and miscellaneous scrap, remain in the surface yard (Burghardt, 1995). This yard is fenced on three sides and is open on its northeast side abutting the Grand Canyon's South Rim. The surface yard is rectangular in shape, measuring approximately 131 m \times 102 m.

Several gamma-radiation survey measurements have been conducted on the Orphan mine surface yard, but only two preliminary surveys have been conducted outside of the fenced enclosure. A gamma survey conducted by the Bureau of Land Management detected elevated gamma values beyond the fence line (Burghardt, 1996). Elevated gamma levels were also detected by a survey conducted by Burghardt (1995). Background gamma-exposure rate measurements ranging from 8 to 10 nanocoulomb (nC)/kg-h (30 to 40 microroentgen (μ R)/h) were detected within 15 to 30 meters east and south of the fenced enclosure. However, to the west of the fenced enclosure, exposure rates ranged between 13 and 98 nC/kg-h (50 and 380 μ R/h) (gross exposure rate) and some quite elevated gamma levels (77–98 nC/kg-h (300–380 μ R/h)) were detected in the immediate vicinity of the West Rim Trail. The highest value measured was 320 nC/kg-h (1250 μ R/h), which was found 15 meters northwest (outside) of the fenced enclosure, just 30 meters from the West Rim Trail. According to Burghardt (1995), mine operations at some point must have extended beyond the present-day fenced yard.

3.3.3.2 Monazite

Numerous studies focus on the radiological impacts from the dry separation processing of mineral sands, especially for workers involved in monazite processing. In studies of radiation doses to Western Australian mineral sands industry workers involved in mining, wet concentration (initial separation process), and transport were considered to be nondesignated employees (Marshman and Hewson, 1994). Nondesignated workers are defined as those employees working in situations who are estimated to receive annual effective dose equivalents (EDEs) substantially less than 5 mSv (500 mrem) (Hewson, 1990). The focus of these studies was on annual committed EDEs associated with designated workers rather than on nondesignated workers.

In the mineral sands mined in Western Australia, the concentration of thorium in the mineral sands or "ore" and in the wet separation (primary) concentrate is estimated to be 0.005% by weight of thorium and 0.03% by weight of thorium, respectively (Hewson, 1990). Typical absorbed dose rates associated with extraction of the mineral sands range from 0.1 to 0.3 microgray (μ Gy)/h (10 to 30 μ rad/h). For the wet separation process, absorbed dose rates range from 0.2 to 1 μ Gy/h (20 to 100 μ rad/h) (Hewson and Hartley, 1990). Airborne gross

alpha activity levels associated with mining and wet separation activities were less than 0.01 $\mbox{Bq/m}^3$

(< 0.3 pCi/m³) and 0.02 Bq/m³ (0.5 pCi/m³), respectively (Hewson and Hartley, 1990).

The major exposure pathways associated with mineral sands processing are inhalation of dust and external radiation, with only small contributions from radon, thoron, and their daughter products. Activity median aerodynamic diameters (AMADs) of airborne dusts in the dry separation plants range from 2 to 12 μ m, with an overall average value of about 6 μ m (Mason et al., 1988). Investigation into the mineralogy of airborne dust has found that monazite concentrates preferentially in the dust (Hartley and Hewson, 1993). In areas where monazite (6 to 7% by weight of thorium) product is stored, absorbed dose rates may reach 100 μ Gy/h (10 mrad/h) or more, depending on the exposure geometry (Mason et al., 1988). In a monazite storage area containing bulk monazite, the thoron progeny concentration near the breathing zone (1.8 meters from the ground) was measured at 15 mWL (Mason et al., 1988). This is nearly two orders of magnitude smaller than the derived air concentration for thoron progeny, which is 1200 WL (Mason et al., 1988).

3.3.4 Present Exemption Analysis

For the distribution and transportation scenario, doses associated with the transport of uranium and thorium ore from the mine to the mill are calculated. The uranium and thorium ore concentrations used for the transport and ore storage exposure scenarios are 0.2% by weight of U_3O_8 (21 Bq/g ²³⁸U (560 pCi/g)) in equilibrium with its progeny and 0.03% by weight of ²³²Th (1.4 Bq/g (37 pCi/g)) in equilibrium with its progeny, respectively. Off-site routine exposure scenarios evaluated are possession of uranium and thorium ore samples by geologists and the exhibition of ore samples by museums. For these scenarios, the uranium ore concentration was increased to 1% by weight of U_3O_8 (104 Bq/g ²³⁸U (2800 pCi/g)) in equilibrium with its short-lived progeny, since it is likely that geologists may be interested in high-grade mineral products.

Potential doses to park rangers and hikers walking by abandoned uranium mines on public lands are also evaluated. In this scenario, the uranium ore concentration was assumed to be 0.2% by weight. The primary exposure pathway for ore transport and storage are airborne particulates, direct irradiation, and to a lesser extent, radon. The primary exposure pathways for possession of an ore sample in a geologist's office and on display in a museum is direct irradiation. Potential exposure pathways to park rangers and hikers from abandoned mines are airborne particulates and direct irradiation. An accident involving the transport of uranium and thorium ore from the mine to the mill is also evaluated.

3.3.4.1 Distribution and Transport

3.3.4.1.1 Uranium Ore Transport

The transport of uranium ore to uranium mills has been drastically reduced over the years. Since there are no underground or surface uranium mines currently producing ore, the following assessment is hypothetical. In NUREG–0706, it was stated that uranium mills have the capacity to process 8700 MT of ore per day, and typically maintain a 10-day backup supply of on-site ore (NUREG–0706). However, the processing of about 270 and up to 2000 MT of ore per day would be more likely (Phone call, J. Cain, Environmental Coordinator and Radiation Safety Officer, Cotter Corporation, Canyon City, CO, October 1996; phone call, S. Landau, Cotter Corporation, Lakewood, CO, July 1997; phone call, O. Paulson, Facility Supervisor, Kennecott Energy, Rawlins, WY, July 1997). In the current assessment, it is assumed that 907 MT of ore (0.2% by weight of U_3O_8 (21 Bq/g ²³⁸U (560 pCi/g))) would be transported from a uranium mine (160 km on public roads one way) to an operating mill per day and each ore-hauling truck would have a 21-MT capacity. Under these circumstances, there could be about 43 trucks per day or about 10,750 truck loads of ore transported annually from uranium mines to a mill (250 day/yr). Under this assumption, one driver would transport two loads of ore per day, resulting in about 500 trips per year. Ore trucks are covered and the ore is wetted prior to transport. Primary exposure routes for truck drivers are direct irradiation and inhalation during ore pickup and drop operations.

MicroShield (Computer Codes, Grove Engineering, 1996) was used to calculate the EDE rate due to direct irradiation. The dump body dimensions used were $5 \text{ m} \times 2.3 \text{ m}$, the thickness of the dump body was 0.3 cm iron, and the source-to-receptor distance was 1 m. The exposure duration per trip was 2 hours (one way), which results in an approximate annual exposure duration of 1000 hours in which a full load of ore is transported to the mill. Given these conditions, the annual EDE is estimated to be 2 mSv (200 mrem).

Fugitive dust emissions result when the ore truck driver picks up or drops a load of ore. A respirable (<15 μ m) particulate emission factor of 0.005 kg/MT of uranium ore was used to estimate the amount of airborne particulate from dropping operations (EPA, Aggregate Handling, 1988). This particulate emission factor takes into account a wetting efficiency of 50% and a mean wind speed of 5 m/s (Kennecott, 1994). Using a box model (Hanna et al., 1982) and assuming an ore dropping area of 30 m² and a mixing height of 10 m, the estimated respirable dust concentration is 900 μ g/m³. Assuming that a driver is standing outside of the truck cab and is exposed about 170 h/yr (10 minutes per pickup and 10 minutes per dropoff), the estimated annual EDE due to dust inhalation is 0.6 mSv (60 mrem). In many cases, the truck driver would not stand outside of the truck cab or the ore could be discharged into a hopper underneath the truck (bottom load truck) and the driver would remain inside the truck cab. The EDE due to radon is assumed to be negligible, since buildup of radon progeny would be minimal in the highly ventilated space in a moving truck. The annual EDE to a truck driver from direct irradiation and dust inhalation is estimated to be 3 mSv (300 mrem).

3.3.4.1.2 Thorium Ore Transport

In the original mineral sand, the thorium content is about 0.0013% by weight (13 ppm). When the dredging method is used to mine mineral sands, a simultaneous gravity separation process is used to separate the silica sand from the minerals. The discarded sand (about 97% of the material excavated) is pumped back into the mining area. Three percent of the remaining minerals are also separated on-site into mineral components. Of the remaining minerals composite, about 0.6 to 0.8% is monazite and about 4.2% of the monazite is thorium (Phone call, J. Raiser, RGC Minerals and Sands, Inc., Green Cove Springs, FL, July 1997). This results in a thorium concentration of about 340 ppm (0.034% by weight, 1.4 Bq/g) (37 pCi/g) in the initial gravity separated mineral ore concentrate. This is the same thorium concentration value given by Hewson (1990). In 10 CFR 40.4, unprocessed, unrefined ore is defined as "ore in its natural form prior to any processing such as grinding, roasting or beneficiating, or refining." Due to the type of mineral sand extraction process, in which there is no change in

chemical or crystal matrix, it is assumed that the mineral product after this initial gravity separation step is unprocessed, unrefined ore.

Although domestic mine production of thorium-bearing monazite ceased in 1994, for this analysis it was assumed that 907 MT of monazite is processed per day. Twenty-one MT-hauling-capacity trucks would be used to transport the mineral sand to the processing plant, but the assumed distance to the plant is about 16 km (a one-way trip would take 0.5 hour at 32 km/h). Therefore, an exposure duration of about 1000 hours was assumed (one driver could make about 5 trips per day, which results in about 1250 trips per year). Using the same dump truck body dimensions and source-to-receptor distance for the driver as used in Section 3.3.4.1.1, the resulting annual EDE due to direct irradiation is about 0.2 mSv (20 mrem). Since the mineral sands separation process is wet (and using a mean wind speed of 2 m/s), the estimated respirable airborne particulate emission factor for the ore drop operation was 0.0007 kg/MT. Using the box model (Hanna et al., 1982) and the same area source dimensions and mixing height as used for uranium ore drop operations, the estimated airborne respirable dust concentration was 400 µg/m³. Assuming an exposure duration of 420 h/yr (10 minutes per ore pickup or drop operation), the estimated annual EDE is 0.14 mSv (14 mrem). The annual individual EDE due to both direct irradiation and dust inhalation to a truck driver hauling mineral sands is about 0.34 mSv (34 mrem).

3.3.4.2 Routine Exposure

The exemption explicitly pertains to receipt, possession, use, or transfer of unrefined and unprocessed ore. Transport of ore was evaluated in Section 3.3.4.1. The exposure scenarios evaluated are the possession of uranium and thorium ore samples in an office and the viewing of ore specimens in a museum exhibit. Another exposure scenario evaluated is park rangers and hikers walking on trails are near abandoned uranium mines. A summary of the estimated individual and collective annual EDEs associated with these exposure scenarios is provided in Table 3.3.4.

3.3.4.2.1 Sample Possession

Geologists, as well as other types of professionals (e.g., mining engineers) may keep ore samples within their offices or laboratories (Tanner, 1990). Organizations such as the U.S. Geological Survey (USGS) also have ore cores and other geological samples in controlled ore storage areas (Phone call, R. Zielinski, USGS, Lakewood, CO, July 1997). Ore samples are not usually kept in the desk, but rather within the office space (Phone call, R. Zielinski, USGS, Lakewood, CO, July 1997). Assuming that a uranium ore sample (1% weight of U_3O_8 , 111 Bq/g (0.003 μ Ci/g)²³⁸U) 15.2 cm in diameter was kept in an office 1.8 m from the worker for 2000 h/yr, the annual EDE due to direct irradiation is about 4×10⁻⁵ mSv (0.004 mrem). If a monazite sample (0.03% by weight of thorium) 15.2 cm in diameter was kept in an office 1.8 m from the worker for 2000 h/yr, the annual EDE due to direct irradiation is about 9×10^{-5} mSv (0.009 mrem). It is unknown how many professionals (geologists or mining engineers) may have ore samples in their possession (e.g., office or laboratory). However, if there are 1,000 individuals who have uranium and thorium ore samples, the annual collective EDEs were estimated to be 4×10^{-5} person-Sv and 9×10^{-5} person-Sv (0.004 person-rem and 0.009 person-rem), respectively.

Extending this scenario to an amateur geologist with 100 uranium ore samples and 100 monazite samples on display in his home office, again with an exposure time of 2000 h/yr, the annual EDE due to direct irradiation is 0.004 mSv (0.4 mrem) for the uranium ore 0.009 mSv (0.9 mrem) for the thorium ore, or less than 0.01 mSv (<1 mrem) in both cases. However, if there are 100 amateur geologists who have uranium and thorium ore samples in collections of this size, the annual collective EDEs are estimated to be 4×10^{-4} person-Sv and 9×10^{-4} person-Sv (0.04 person-rem and 0.09 person-rem), respectively.

Numerous ore samples are on display in U.S. museums. In the current assessment it is assumed that a uranium (1% by weight of U_3O_8) and monazite (0.03% by weight of thorium) ore sample (5-cm diameter) are in a display case and the museum visitor stands 0.6 m from the display for about 10 minutes. Based on these conditions, the estimated annual EDEs from viewing these uranium and monazite samples are less than 1×10^{-5} mSv (<0.001 mrem) for each, respectively. It is unknown how many visitors view mineral exhibits in museums. If 1 million people view uranium and thorium ore specimens in museums annually, the annual collective dose equivalents would be less than 1×10^{-5} person-Sv (<0.001 person-rem) for each ore type.

3.3.4.2.2 Abandoned Uranium Mines

The National Park Service abandoned mine land inventory shows 42 abandoned uranium mine sites within units of the National Park System and 2 immediately adjacent to park boundaries (Phone call, J. E. Burghardt, Geologic Resources Division, National Park Service, Lakewood, CO, July 1997). The majority of these sites are in the southern Utah parks of Canyonlands, Capitol Reef, and Glen Canyon (Phone call, J. E. Burghardt, Geologic Resources Division, National Park Service, Lakewood, CO, July 1997). There are numerous other abandoned mines on the Colorado Plateau, a physiographic province centered on the "four corners" area of Utah, Colorado, New Mexico, and Arizona (Burghardt, 1996). There are many more abandoned uranium mines on other Federal lands (Phone call, J. E. Burghardt, Geologic Resources Division, National Park Service, Lakewood, CO, July 1997). In many cases, abandoned uranium mines are in remote areas that are not easily accessible by the public. However, some abandoned uranium mines may be accessible to the public.

In the current assessment, a hypothetical abandoned uranium mine, located in a national park in an area accessible by hikers, is evaluated. It is assumed park rangers lead groups of hikers along a trail near an abandoned uranium mine. The trail is on land contaminated with residual uranium ore (from an ore storage pad and truck loading area) and mine spoils. It is assumed that due to weathering, the uranium contamination extends 5 cm into the soil. The uranium ore and mine spoils contain 0.2% by weight of U_3O_8 and are spread over a number of acres.

Dose coefficients for exposure to soil contaminated to a depth of 5 cm from Federal Regulatory Guidance Report No. 12 (EPA-402-R-93-081) were used to estimate annual EDEs due to direct irradiation. The exposure duration for the park ranger leading the hikes is 100 h/yr. For an individual hiker, the exposure duration is 2 h/yr. Based on these assumptions, the annual EDE for the park ranger and hiker due to direct irradiation is estimated to be 0.008 mSv and 2×10^{-4} mSv (0.8 mrem and 0.02 mrem), respectively. To estimate the annual EDE due to dust inhalation from wind erosion, a mass loading factor of 100 μ g/m³ is used (Healy, 1980). The annual individual EDE for park ranger and hiker due to dust inhalation would be about 0.04 mSv (4 mrem) and 8×10⁻⁴ mSv (0.08 mrem), respectively. Doses due to radon emanation at the

abandoned uranium mine site are considered to be negligible. Therefore, the annual EDEs due to direct irradiation and dust inhalation for a hypothetical park ranger and hiker are about 0.05 mSv (5 mrem) and 0.001 mSv (0.1 mrem), respectively.

It is unknown how many park rangers and hikers may walk by abandoned uranium mines on public lands. However, if 100 park rangers and 100,000 hikers have access to abandoned uranium mine property, the annual collective EDEs are 0.005 person-Sv (0.5 person-rem) and 0.1 person-Sv (10 person-rem), respectively.

3.3.4.3 Disposal

Uranium ore, once mined, is typically processed into a commercial product. Rather than be disposed, it may be stockpiled to blend with other ores or be maintained until economic conditions allow for cost-effective ore processing. Large inventories of ore cores and samples, such as those maintained by the USGS, are being sent to a uranium mill for processing (Phone call, D. DePlato, Radiation Safety Officer, RGC Minerals and Sands, Green Cove Springs, FL, July 1997), and it is very unlikely that residual ore and mine spoils at an abandoned uranium mine would be sent to at an off-site disposal site (Phone call, J. E. Burghardt, Geologic Resources Division, National Park Service, Lakewood, CO, July 1997). At one facility that commercially produced thorium, monazite is being reintroduced into the mineral sand mining area (Phone call, W. Cofer, Health Physicist, State of Florida Department of Health and Rehabilitative Services, Orlando, FL, July 1997).

The most likely scenario in which uranium and thorium ore may be disposed in municipal landfills is the disposal of discarded small uranium and thorium ore samples. It is assumed that 3500 uranium (0.2% by weight of U_3O_8) and thorium (0.03% by weight of thorium) 1-kg ore samples are disposed in 1 year. Summarized in Tables 3.3.2 and 3.3.3 are the individual and collective EDEs due to disposal of ore samples, respectively. These EDEs are estimated using the generic disposal methodology described in Appendix A.2.

3.3.4.4 Accidents

For this analysis, it is assumed that an accident occurs during transport, resulting in fire. It is difficult to imagine a fire causing a significant fraction of the ore to become airborne; however, firefighting activities could disperse the material and cleanup workers could encounter contamination. Thus, this should be considered as an upper bounding scenario. It is assumed that 21 MT of uranium (0.2% by weight of U_3O_8) and thorium (0.03% by weight of thorium) ore is being transported. The generic accident methodology described in Appendix A.1 is used for this analysis. In a transport accident involving uranium ore, the estimated EDE due to inhalation and submersion to a firefighter is 0.03 mSv (3 mrem). The estimated EDE to a cleanup worker due to particulate resuspension is 0.3 mSv (30 mrem); however, no respiratory equipment is assumed for the cleanup worker. It is very likely that a cleanup worker for a truck load of uranium ore would wear a respirator. If a dust respirator with an assigned protection factor (APF) of 10 was used, the EDE would be reduced to 0.03 mSv (3 mrem).

In the transportation accident involving thorium ore, the estimated EDE due to inhalation and submersion to a firefighter is 0.01 mSv (1 mrem). The estimated EDE to a cleanup worker due to particulate resuspension is 0.1 mSv (10 mrem); however, no respiratory equipment is assumed for the cleanup worker. As mentioned for the transportation accident involving

uranium ore, cleanup workers would also wear some type of respiratory equipment for accidents involving thorium ore. Even if a dust respirator with an APF of 10 was used, the EDE would be reduced to 0.001 mSv (0.1 mrem).

3.3.5 Summary

It appears that the original application of this exemption was for possession of uranium ore by miners and the transfer of the ore to an ore-buying station or directly to the milling facility. Based on the wording in 10 CFR 40.13 (b), a person would be exempt from the requirements of a license if that person receives, possesses, uses, or transfers any type of unrefined and unprocessed ore containing source material. Summarized in Table 3.3.4 are the results of the current dose assessment, which evaluated the transfer, receipt, possession, and disposal of uranium and thorium ore. In addition, the radiological impact of abandoned uranium mines located on Federal lands that may be accessible to members of the public is evaluated.

Based on the current analysis, the following general conclusions were obtained concerning radiological impacts on truck drivers, and other members of the public who may come in contact with ore samples or residual ore contamination at abandoned uranium mines:

- Hypothetical annual EDEs to ore truck drivers appear to be about 0.3 mSv (300 mrem).
- Doses to members of the public from ore samples in museum exhibits are very low. However, there is a potential of higher doses to members of the public from contamination at abandoned uranium mines that may be accessible to the public.

	Uranium Ore (0.2% by weight)	Thorium Ore (0.03% by weight)
Receptors	Individual Annual Effective Dose Equivalent (mrem)ª	Individual Annual Effective Dose Equivalent (mrem)ª
Truck drivers ^₅	300	34
Geologist with ore sample ^c	0.004	0.009
Museum visitor ^c	<0.001	<0.001
Park ranger ^d	5	
Hiker ^d	0.1	

Table 3.3.1 Individual Doses to Workers and the Public From Unrefined Uranium and Thorium Ore

^a 1 mrem = 0.01 mSv.

....

^b See Section 3.3.4.1.1 for further information on the parameters used to estimate the annual individual and collective effective dose equivalents (EDEs) for ore-hauling truck drivers. ^c See Section 3.3.4.2.1 for further information on the parameters used to estimate the annual

individual EDEs for the individual with ore samples or for museum visitors viewing a mineral exhibit.

^d See Section 3.3.4.2.2 for further information on the parameters used to estimate the annual individual EDEs for park rangers and hikers walking on abandoned uranium mine property located on public lands.

	<u>Uranium Ore (0.2% by weight)</u>	Thorium Ore (0.03% by weight)		
Receptors	Individual Annual Effective Dose Equivalent (mrem)ª	Individual Annual Effective Dose Equivalent (mrem)ª		
Waste collectors	<0.001	<0.001		
Landfill operators	<0.001	<0.001		
<u>Off-site receptor</u> Airborne releases Groundwater	<0.001 <0.001	<0.001 <0.001		
On-site receptor	<0.001	<0.001		

Table 3.3.2 Individual Doses Attributed to Disposal of Uranium andThorium Ore Samples

^a 1 mrem = 0.01 mSv.

	Uranium Ore (0.2% by weight)	Thorium Ore (0.03% by weight)		
Receptors	Collective Effective Dose Equivalent (person-rem)ª	Collective Effective Dose Equivalent (person-rem) ^a		
Waste collectors	0.001	0.001		
Landfill operators	<0.001	<0.001		
<u>Off-site receptor</u> Airborne releases Groundwater	<0.001 <0.001	<0.001		
On-site receptor	<0.001	0.006		

Table 3.3.3 Collective Doses Attributed to Disposal of Uranium and
Thorium Ore Samples

^a 1 person-rem = 0.01 person-Sv.

....

	Individual Annual Effective Dose Equivalent (mrem) ^b		Collective Effective Dose Equivalent ^a (person-rem) ^b	
Activities and Receptors	Uranium Ore	Uranium Ore Thorium Ore		Thorium Ore
	TRAN	ISPORT°		
Truck driver	300	34		
	ROUT	INE USE ^d		
Worker exposures Geologist with sample Park ranger	0.004 5	0.009	0.004 0.5	0.009
<u>Public exposure</u> Museum visitor Hiker Amateur geologist	<0.001 0.1 0.4	<0.001 0.9	<0.001 10 0.04	<0.001 0.09
	LANDFILL	DISPOSAL®		
Occupational Collectors Landfill operators	<0.001 <0.001	<0.001 <0.001	0.001 <0.001	0.001 <0.001
<u>Public exposure</u> Off-site resident - Airborne releases - Groundwater releases	<0.001 <0.001	<0.001 <0.001	<0.001 0.004	<0.001
Future on-site resident	0.001	<0.001	<0.001	0.006
	ACC	IDENT	- ··· ··· <u>······</u> · · · ·	· · · · · · · · · · · · · · · · · · ·
<u>Transportation fire</u> Firefighter Cleanup worker	3 30	1 10		

Table 3.3.4 Summary of Doses Associated With Unrefined Uranium and
Thorium Ore

^a Refer to text discussion for time period for collective dose assessment.

^b 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

^c Refer to Section 3.3.4.1 for further information on the estimation of annual individual and collective dose equivalents attributed to transportation operations.

^d Refer to Section 3.3.4.2 for further information on the estimation of annual individual and collective dose equivalents attributed to routine operations.

^e Refer to Section 3.3.4.3 for further information on the estimation of annual individual and collective dose equivalents attributed to disposal. Also see Appendix A.2 for a description of the generic disposal methodology.

¹ Refer to Section 3.3.4.4 for further information on the estimation of the individual dose equivalent attributed to accidents. Also see Appendix A.1 for a description of the generic accident methodology.

3.4 Incandescent Gas Mantles

3.4.1 Introduction

In 10 CFR 40.13(c)(1)(i), any person is exempt from the requirements for a license to the extent that the person receives, possesses, uses, or transfers any quantity of thorium contained in incandescent gas mantles. This exemption was established on March 20, 1947 (12 FR 1855), and has remained essentially unchanged since that time.

The *Federal Register* notice cited above provided no information on radiological impacts on the public for use or disposal of gas mantles containing thorium. Information published in 1960 by the Atomic Energy Commission (25 FR 8619) indicated that the exemption would not result in an unreasonable hazard to life or property, but a supporting dose analysis was not published.

Estimates of potential radiation doses to members of the public from incandescent gas mantles containing thorium have been published in other reports by Buckley et al. (NUREG/CR–1775), O'Donnell and Etnier (NUREG/CR–1910), and the National Council on Radiation Protection and Measurements (NCRP 95). These dose estimates are discussed in Section 3.4.3 of this report.

3.4.2 Description of Items

Thorium-containing mantles are available in a variety of designs and sizes, each intended to fit into one of the many different lighting devices in use. To function, the mantle must be heated to a temperature of 1870 to 2370°C, which causes the thorium oxide in the mantle to incandesce (NUREG/CR-1910). This is achieved by placing the mantle over, in, or near a gas or kerosene flame that burns during the operation of the device.

Incandescent gas mantles containing thorium have been used for indoor and outdoor lighting since the 1880s. In 1950, gas mantles were still being used to light the streets of some communities (NUREG/CR–1775), and in 1952, 65% of the U.S. production of thorium was used to make gas mantles (Cuthbert, 1958). In 1974, the U.S. production of gas mantles was estimated to be 20 million units per year (NUREG/CP–0001, Barker and Tse), and in 1981, U.S. domestic use of gas mantles was estimated to be 25 million units per year (NUREG/CR–1910). At present, U.S. sales of gas mantles are estimated to be 50 million units per year (Phone call, R. May, The Coleman Company, Inc., Wichita, KS, April 1997). The Coleman Company, a major U.S. supplier of gas mantles, recently replaced thorium in its gas mantles with (nonradioactive) yttrium (Couch and Vaughn, 1995). However, thorium-containing mantles are still available in the United States under a variety of brand names (Couch and Vaughn, 1995), and they account for about half of the current U.S. sales, or about 25 million units per year (Phone call, R. May, The Coleman Company, Inc., Wichita, KS, April 1997).

Thorium–containing mantles are manufactured by soaking a webbed fabric (or stocking) in a solution containing nitrates of thorium and other metals (NUREG/CR–1910). The other metals are added either to improve the luminosity of the mantles (e.g., cerium) or to help harden the fragile structure of the mantle (e.g., beryllium) (Griggs, 1973). When saturated, the fabric is removed from the solution and dried. In some cases, the fabric is also treated to convert the soluble nitrates into insoluble compounds (most likely hydroxides), then it is rinsed and dried (NUREG/CR–1910). The fabric is finally coated with lacquer, dried, cut, and fashioned into

mantles. Most mantles, when flattened, consist of two layers of fabric that are 4 to 11 cm long by 2 to 7 cm wide and contain between 50 and 500 mg of thorium (NUREG/CR-1910). At this stage, the mantles can be finished as either "soft" or "hard" mantles.

To form a soft mantle, the stocking is cut off, sewn shut to form a bag, and either equipped with a drawstring for tying it directly to the burner assembly or affixed to a holder that attaches to the burner assembly (NUREG/CR–1910). Hard mantles are bags that are preshaped by attachment to a rigid support frame that either attaches to or sits over the burner assembly. Some hard mantles are also preburned at the factory to convert the thorium to thorium oxide (NUREG/CP–0001, Cullen and Paschoa). This procedure volatilizes the fabric bag and the lacquer coating, leaving a fragile shell of thorium and other metal compounds. If the hard mantles are not preburned or soft mantles are being used, they are placed over the burner assembly in a lighting device, which may accommodate one to four mantles, then preburned with no fuel being supplied to convert the thorium to thorium oxide (NUREG/CR–1910).

Finished mantles are packaged and distributed as ordinary consumer products (NUREG/CR-1910). Hard mantles are normally boxed singly, and one or two soft mantles are usually sealed in a plastic pack. In both cases, the box or pack may be placed in a carton containing a new lighting device, or they may be combined with other boxes or packs in a carton containing replacement mantles. Except for mantles to be installed in outdoor lighting devices, which are normally purchased and installed by utility companies, most mantles and new lighting devices are purchased from retail stores and installed in the lighting devices by users.

Generally, the lifetime of a gas mantle depends on its use environment (NUREG/CR-1910). In stable environments (e.g., fixed outdoor and indoor lamps), mantles may function for 6 to 8 months before breaking. In less stable environments (e.g., moveable residential lamps), they may function for 1 to 3 months. In unstable environments (e.g., camping lanterns), they may function over a time span that includes 50 hours of actual burning. In all use environments, however, the lifetime of a gas mantle is estimated to be less than 1 year.

3.4.3 Summary of Previous Analyses and Assessments

A variety of distribution, routine use, disposal, accident, and misuse exposure scenarios were considered in the previous assessments by Buckley et al. (NUREG/CR–1775) and O'Donnell and Etnier (NUREG/CR–1910). For distribution, routine use, and disposal, these scenarios include the following:

- Exposures to workers and members of the public during distribution, transport, and installation.
- Exposures to campers during use of portable lanterns and changing of mantles.
- Exposures from residential and commercial outdoor lighting and residential indoor lighting.
- Exposures from lighting in recreational vehicles.

• Exposures from disposal of mantles in landfills and by incineration.

The exposure scenarios involving accidents and misuse of gas mantles that have been considered include:

- Contamination of a campground following breakage of a mantle and transport of thorium and its decay products to a nearby reservoir used to supply drinking water.
- Ingestion of a mantle by a small child.
- Exposure to firefighters to thorium and its decay products released in a warehouse fire.

The dose estimates of O'Donnell and Etnier (NUREG/CR–1910) were based on an annual distribution, use, and disposal of 25 million mantles. Both individual and collective doses were presented as ranges of possible doses because of uncertainties regarding (1) the actual radionuclide content of the mantles, (2) the rate of diffusion of ²²⁰Rn from the mantles and stored lanterns, (3) the identities and quantities of radionuclides that may evaporate from the mantles during use, and (4) the behavior of persons during and after mantle replacement. To account for these uncertainties, doses were calculated with the following assumptions:

- The radionuclides initially present in the mantles had activities corresponding to their activities in natural thorium or in thorium that was purified and aged for 6 months (see Table 3.4.1). Each mantle was assumed to contain 250 mg of thorium, although the thorium content of some mantles was as high as 400 mg (NUREG/CR-1775). The activity of ²³²Th in such a mantle is about 1 kilobecquerel (kBq) (27 nanocurie (nCi)). The total activity in the mantle is about 10 kBq (270 nCi) if the mantle contains natural thorium or 7.8 kBq (210 nCi) if it contains 6-month-old thorium.
- During times the mantles are not burning, 25% of the activities in Table 3.4.1 for ²²⁰Rn and its progeny were assumed to diffuse into the air. The air concentrations used in the dose estimates depended upon the volumes and ventilation rates of the contaminated air spaces. For mantles stored in lanterns in homes, doses were also calculated for the limiting cases when (a) all of the ²²⁰Rn diffuses from the mantles and disperses immediately in the home and (b) only 5% of the diffusing ²²⁰Rn enters the home air.
- To bracket possible effects of nuclide vaporization during mantle use, dose estimates were made for complete vaporization of (a) only ²²⁰Rn and its progeny and (b) all except the two thorium radionuclides, ²³²Th and ²²⁸Th (see Table 3.4.1).
- To estimate doses from changing mantles, the following exposure scenario was used:
 - A. Exposure to two new gas mantles
 - (1) Direct exposure. The installer is exposed for 1 minute at an average distance of 1 meter from two mantles containing a total of 500 mg of 6-month-old thorium that is depleted in ²²⁰Rn and its decay products because of diffusion (25%) from the thorium.

- (2) Inhalation exposure. Twenty-five percent of the ²²⁰Rn escapes into a hemispherical air space with a volume of 7 m³ and a ventilation rate of either 1000 volume changes per hour (an outdoor replacement in a 3-kilometer per hour breeze) or 1 volume change per hour (an indoor replacement).
- B. Exposure to two old mantles
 - (1) *Direct exposure*: Same as above.
 - (2) Inhalation exposure.
 - (a) Same as above.
 - (b) The installer blows broken mantles from the lighting device and 1% (5 mg) of the ash from the broken mantles is dispersed as respirable particles into the air space.
 - (3) Ingestion exposure. While handling the old mantles 5% (25 mg) of the thorium adhered to the installer's hands, and subsequent washing left only 1% (0.25 mg) of the adhered material, which was ultimately ingested.

Depending on which parts of the mantle replacement scenario were used, O'Donnell and Etnier (NUREG/CR–1910) estimated total doses (the sum of the dose equivalents from external exposures plus the 50-year dose commitments from inhalation and ingestion) for outdoor replacement of two mantles that ranged from 0.04 nanosievert (nSv) (4 nrem) to 0.002 mSv (0.2 mrem) to the total body, from 0.04 nSv (4 nrem) to 0.01 mSv (1 mrem) to the bone, and from 0.2 nSv (20 nrem) to 0.4 μ Sv (40 μ rem) to the lungs. Indoor replacement was estimated to give total doses between 0.8 nSv (80 nrem) and 0.007 mSv (0.7 mrem) to the total body, between 0.003 μ Sv (0.3 μ rem) and 0.06 mSv (6 mrem) to the bone, and between 0.4 μ Sv (40 μ rem) to the lungs.

During distribution, routine use, and disposal, O'Donnell and Etnier (NUREG/CR-1910) estimated the dose from both external and internal exposure, as appropriate for each exposure scenario. The best estimates of individual and collective doses, rather than the minimum or maximum values, obtained by O'Donnell and Etnier are summarized as follows:

- For workers during transportation and distribution and installation, annual dose equivalents to individuals were 0.002 mSv (0.2 mrem) to the whole body, 0.004 mSv (0.4 mrem) to the bone, and 0.002 mSv (0.2 mrem) to the lungs. Annual collective dose equivalents were 0.6 person-Sv (60 person-rem) to the whole body, 1 person-Sv (100 person-rem) to the bone, and 0.5 person-Sv (50 person-rem) to the lungs.
- For routine users of gas mantles (including campers and their families), persons exposed during indoor and outdoor residential use, and persons using recreational vehicles, the annual dose equivalents to individuals were 0.001 mSv (0.1 mrem) to the whole body, 0.007 mSv (0.7 mrem) to the bone, and 0.02 mSv (2 mrem) to the lungs. Annual collective dose equivalents were 40 person-Sv (4000 person-rem) to the whole

body, 200 person-Sv (20,000 person-rem) to the bone, and 500 person-Sv (50,000 person-rem) to the lungs.

 For other members of the public who do not use gas mantles but receive exposure while shopping in stores or living along truck routes or near incinerators, annual dose equivalents to individuals were 0.01 μSv (1 μrem) to the whole body, bone, and lungs. Annual collective dose equivalents were 3 person-Sv (300 person-rem) to the whole body and bone and 2 person-Sv (200 person-rem) to the lungs.

The best estimates of the radiation doses for the accident and misuse scenarios are summarized as follows:

- For contamination of a campground following breakage of a mantle and transport of thorium and its decay products to a nearby reservoir used to supply drinking water, O'Donnell and Etnier (NUREG/CR–1910) estimated annual dose equivalents of less than 0.01 μSv (<1 μrem) to the whole body, bone, and lungs, and annual collective dose equivalents of less than 0.001 person-Sv (<0.1 person-rem) to the whole body, bone, and lungs.
- For ingestion of a whole mantle, O'Donnell and Etnier estimated dose equivalents of individuals of 1 mSv (100 mrem) to the whole body, 7 mSv (700 mrem) to the bone, and 0.4 mSv (40 mrem) to the lungs.
- For exposure to firefighters at a warehouse fire involving 10,000 gas mantles, and using conservative assumptions for the amount of thorium released in a fire plus no respiratory protection for firefighters, Buckley et al. (NUREG/CR–1775) estimated dose equivalents to individuals of 3 mSv (0.3 rem) to the whole body, 70 mSv (7 rem) to the bone, and 40 mSv (4 rem) to the lungs.

Both O'Donnell and Etnier (NUREG/CR–1910) and Buckley et al. (NUREG/CR–1775) considered doses from disposal of gas mantles in a landfill. However, neither analysis appears to be adequate because only the mobilization and transport of thorium and its decay products to a nearby source of water was considered. Since the assumed scenario is quite conservative and seems unlikely for such an immobile radionuclide as thorium, a more likely exposure scenario involves inadvertent intrusion into solid waste in a landfill by future on-site residents, but doses from this scenario were not evaluated.

The dose analyses for thorium gas mantles were summarized in NCRP 95 by expressing the dose estimates in terms of effective dose equivalents (EDEs). Based on the results of O'Donnell and Etnier (NUREG/CR–1910), the NCRP came to the following conclusions:

- The average annual EDE to individuals would be about 0.002 mSv (0.2 mrem), with the maximum dose to individuals using worst-case assumptions possibly being a factor of 100 higher, or 0.2 mSv (20 mrem).
- The annual collective EDE from distribution, routine use, and disposal of 25 million gas mantles per year would be about 90 person-Sv (9000 person-rem).

Thus, it can be concluded that the existing analyses of potential doses to the public from use of thorium gas mantles are adequate, for the most part. However, three refinements to the existing analyses appear to be needed. First, the EDEs should be calculated for exposure scenarios involving inhalation and ingestion of thorium and its decay products. Second, accidents involving fire should be reevaluated using the generic accident methodology of Appendix A.1 of this report. Third, realistic intrusion scenarios following disposal of gas mantles in a landfill should be evaluated in addition to more realistic scenarios for release of radioactivity to the general environment using the generic disposal methodology of Appendix A.2 of this report.

3.4.4 Present Exemption Assessment

Table 3.4.2 presents results of the current assessment of potential radiation doses to members of the public from the annual distribution, use, and disposal of 25 million gas mantles containing thorium. The current assessment relies very heavily on the previous work of O'Donnell and Etnier (NUREG/CR–1910) and on a study of radioactivity released from burning gas lantern mantles by Luetzelschwab and Googins (1984).

Based on the work of O'Donnell and Etnier (NUREG/CR–1910), it is assumed that 20.25 million of the gas mantles are used in portable lanterns (e.g., camping lanterns), 3 million in commercial and residential outdoor gaslights, and 1.25 million in indoor residential gas or kerosene lamps. It is also assumed based on their previous work that (1) the amount of thorium contained in the gas mantles is 250 mg, and (2) the thorium has decayed for 6 months after being purified (see Table 3.4.1). The value of 250 mg of thorium per mantle is in close agreement with results from a survey of 15 mantle packages (2 mantles per package) by Luetzelschwab and Googins (1984). They found that the average ²³²Th activity per package was 2.2 kBq (65 nCi), which corresponds to a mass of 600 mg of ²³²Th per package (or 300 mg per mantle). The activity of 6-month-old thorium was also found to provide a conservative estimate of the total activity in the mantles for the first year or two after manufacture, when most mantles are used (Luetzelschwab and Googins, 1984).

If radiation doses are primarily due to external irradiation of the total body by photons from the gas mantles, the EDE can be equated with the "whole body" dose obtained previously by O'Donnell and Etnier (NUREG/CR–1910). This is possible because the external radiation doses from the thorium decay series are due to high-energy gamma rays coming primarily from ²²⁸Ac, ²¹²Pb, ²¹²Bi, and ²⁰⁸Tl (Luetzelschwab and Googins, 1984). However, if internal exposures from ingestion or inhalation are involved, the 50-year committed EDEs are reevaluated here using dose conversion factors from Table 3.1.7 in Section 3.1 of this report. The dose conversion factor in Table 3.1.7 for inhalation of ²²⁰Rn and its short-lived progeny, ²¹⁶Po through ²⁰⁸Tl, assumes that all of these radionuclides are in radioactive equilibrium in air. It must be multiplied by so-called equilibrium factor to account for the normal disequilibrium between the ²²⁰Rn and its short-lived products in air (International Commission on Radiological Protection (ICRP) 50). The value of the equilibrium factor may vary typically between 0.1 and 0.01 for indoor air of homes and by values even smaller than 0.01 for outdoor air.

3.4.4.1 Distribution and Transport

This section considers potential radiation doses for (1) transport of gas mantles to distributors, and (2) distribution of gas mantles by retail stores and both distribution and installation of gas mantles by utility companies. Utility company installers are a special case of distribution workers because they could inhale or ingest radionuclides while replacing gas mantles in commercial and residential outdoor gaslights.

During most distribution and transport activities, the potential doses to workers and members of the public are due primarily to external radiation; thus the results of the previous assessment by O'Donnell and Etnier (NUREG/CR-1910) are used. However, the potential doses to installers are reassessed here to obtain up-to-date estimates of the 50-year committed EDEs for inhalation and ingestion of radionuclides during the installation of gas mantles in a lighting device.

3.4.4.1.1 Transport

For transport of gas mantles to distributors, O'Donnell and Etnier (NUREG/CR–1910) estimated the annual collective dose equivalent to be 0.01 person-Sv (1 person-rem). The annual individual EDEs ranged from a low of less than 1×10^{-5} mSv (<0.001 mrem) to members of the public along truck routes to a high of about 0.002 mSv (0.2 mrem) for truck drivers (NUREG/CR–1910).

3.4.4.1.2 Installation

During mantle installation, the installer may be exposed via (1) direct external irradiation, (2) inhalation of ²²⁰Rn and its short-lived progeny because of diffusion of ²²⁰Rn from the mantles, (3) inhalation of ash from broken mantles that is dispersed in air as respirable particles, and (4) ingestion of ash from the used mantles. Exposure scenarios used in this assessment for mantle replacement are the same as those in the previous work by O'Donnell and Etnier (NUREG/CR–1910) (see Section 3.4.3).

Table 3.4.3 lists total EDEs from both external and internal exposure for mantle changes occurring either outdoors or indoors. For outdoor replacement of two mantles, the EDE to the installer could range from a low of less than 1×10^{-5} mSv (<0.001 mrem) to a high of 0.02 mSv (2 mrem). For indoor replacement of two mantles, the EDEs could range from a low of 2×10^{-4} mSv (0.02 mrem) to a high of 0.04 mSv (4 mrem). The lower doses are due to direct external irradiation and submersion in or inhalation of airborne radionuclides due to diffusion of ²²⁰Rn from the mantles. The higher doses are mainly due to inhalation of airborne ash from the used mantles and, to a lesser extent, to ingestion of ash from the used mantles. For utility company installers, it is assumed that all mantle changes typically occur outdoors rather than indoors.

Each utility company installer is assumed to replace 500 mantles per year in 250 outdoor commercial and residential gaslights (NUREG/CR–1910). Thus, the annual individual EDE to an installer could range between 5×10^{-4} mSv (0.05 mrem) and 5 mSv (500 mrem). It is doubtful, however, that these professional installers would in all cases blow mantle parts from the gaslights and ingest ash from the used mantles. In the latter case, subsequent work activities and hand washing would likely remove all of the adhered mantle ash from the hands.

Hence, doses to installers were arbitrarily chosen to be 10% of the higher values given above for outdoor mantle changes. This yielded annual individual effective doses to installers of 0.5 mSv (50 mrem), and a collective EDE of about 3 person-Sv (300 person-rem) to the 6,000 installers needed to replace the 3 million mantles used each year in commercial and residential outdoor gaslights.

3.4.4.1.3 Distribution

For distribution of gas mantles to routine users, the collective dose equivalent is estimated to be 7 person-Sv (700 person-rem) based on the results from Section 3.4.4.1.2 of this report and the previous assessment by O'Donnell and Etnier (NUREG/CR–1910). This total collective dose is based on estimates of 0.04 person-Sv (4 person-rem) to warehouse workers, 0.006 person-Sv (0.6 person-rem) to truck drivers, 0.6 person-Sv (60 person-rem) to store workers, 3 person-Sv (300 person-rem) to store customers, 0.01 person-Sv (1 person-rem) to utility stock clerks, 3 person-Sv (300 person-rem) to utility company installers, and 6×10^{-5} person-Sv (0.006 person-rem) to the public along truck routes. The annual individual EDEs ranged from a low of less than 1×10^{-5} mSv (<0.001 mrem) for members of the public along truck routes to a high of 0.1 mSv (10 mrem) for warehouse workers (NUREG/CR–1910). For store customers and store clerks, the annual individual EDEs averaged 2×10^{-5} mSv (0.002 mrem) and 0.002 mSv (0.2 mrem), respectively.

3.4.4.2 Routine Use

This section considers potential radiation doses from routine use of gas mantles in (1) camping lanterns, (2) outdoor commercial and residential gaslights, and (3) indoor residential lights.

During routine use, radionuclides may escape from mantles and become airborne via two different processes: vaporization and diffusion (NUREG/CR–1910). Diffusion produces a steady-state flux of ²²⁰Rn from a gas mantle, and the effect of the ²²⁰Rn diffusion on the thorium decay chain is simply to reduce the values in Table 3.4.1 by the fraction of the ²²⁰Rn that escapes. For example, the assumed diffusion rate of 25% for a mantle in storage results in relative activities for ²²⁰Rn and its progeny that are only 75% of those given in Table 3.4.1. Unless a barrier intervenes such as the plastic pack around new mantles, the ²²⁰Rn that diffuses from the mantles will provide a steady flux of ²²⁰Rn into the surrounding air. This ²²⁰Rn will decay and introduce its progeny into the air, and the resulting airborne concentrations are a potential source for exposure, particularly indoors.

During operation of a lighting device, gas mantles may also reach high enough temperatures to volatilize radionuclides contained within the mantles (NUREG/CR–1910). This process was noted in a study by Griggs (1973) and is supported by measurements from a study of Luetzelschwab and Googins (1984). To discuss this process, it is convenient to divide the thorium decay chain of Table 3.4.1 into three groups of nuclides: (1) ²³²Th and ²²⁸Th; (2) ²²⁸Ra, ²²⁸Ac and ²²⁴Ra; and (3) ²²⁰Rn and its progeny (²¹⁶Po through ²⁰⁸Tl). Vaporization is characterized by an initial, pulse-type release of radionuclides in Groups 2 and 3, but not those in Group 1 (Luetzelschwab and Googins, 1984). Unlike diffusion, which is a continuing effect, the effects of vaporization are short-term because the ²²⁰Rn and its progeny (Group 3) will retain radioactive equilibrium within the mantle over a period of a few days. Thus, vaporization of radionuclides from mantles is a complex function of the prior use of the mantles (NUREG/CR–1910).

The effects of vaporization and diffusion of radionuclides from the mantles have been taken into account in the internal dose calculations of this study. Based on the measurements of Luetzelschwab and Googins (1984), it was assumed that 30% of the radium nuclides (Group 2) and 100% of the ²²⁰Rn and its progeny (Group 3) were released from the mantles and into the air by vaporization in an instantaneous, pulse-like release during ignition of the lighting devices. It was assumed further that (1) 100% of the ²²⁰Rn diffused from the mantles and into the air throughout continued operation of the lighting devices, and (2) 25% of the ²²⁰Rn diffused from the mantles and into air while the mantles were simply being stored inside the nonignited lighting devices. The effects of depletion of radionuclides in the mantles by both vaporization and diffusion also were considered in the previous external dose calculations of O'Donnell and Etnier (NUREG/CR–1910).

3.4.4.2.1 Camping Lanterns

Persons who use incandescent gas mantles in camping lanterns may be exposed to them under a multitude of conditions. Thus, O'Donnell and Etnier (NUREG/CR-1910) constructed a list of exposure events for a 2-night camping trip by an avid camper. These exposure events are summarized in Table 3.4.4 of this report. The table identifies the types of absorbers considered in the external dose calculations and the sources of airborne radionuclides considered in the internal dose calculations. The times for the various exposure events during the camping trip are total times, except for those involving the loading (or unloading) of the car. For loading of the car, the times are only those involved in handling the lanterns and mantles. The total time spent loading the car would be longer and would depend to some extent on whether the camper is an avid camper or a one-time camper, as defined below.

Categories of campers considered here include the principal camper and three other members in a family of avid or one-time campers. A principal camper is defined as one who participates in all the exposure events in Table 3.4.4, with the exception of traveling in the rear seat. Other members are defined as those who travel in the vehicle and are not involved in either purchase or ignition of mantles. Avid campers are those who make 26 two-night camping trips per year with two double-mantled lanterns and four replacement mantles; one-time campers are those who make 1 two-night camping trip per year with a single-mantled lantern and one replacement mantle. Also, a principal avid camper is assumed to replace 8 mantles per year (in 2 doublemantle lanterns two times per year); a principal one-time camper is assumed to replace 1 mantle per year. Campers also are assumed to be exposed between camping trips by inhalation of radionuclides in air due to diffusion of ²²⁰Rn from mantles stored in lanterns in the home.

Table 3.4.5 presents estimates of potential individual doses from the various activities associated with camping lantern use. For the camping activities defined in Table 3.4.4, it was estimated that avid campers and their families could receive annual EDEs of between 5×10^{-4} mSv and 0.06 mSv (0.05 and 6 mrem) and one-time camping families could receive between 2×10^{-5} and 6×10^{-4} mSv (0.002 and 0.06 mrem). For the outdoor inhalation exposures, hemispherical air spaces were assumed with the same radii and ventilation rates as those used previously by O'Donnell and Etnier (NUREG/CR–1910). For indoor inhalation exposures, the volume of the car was assumed to be 6 m³ with a ventilation rate of 5 volume changes per hour (see Appendix A.1), and the volume of the tent was assumed to be 20 m³ and its ventilation rate was assumed without basis to be the about the same as that of the car (i.e., 5 volume changes per hour). For these inhalation exposures, committed EDEs were calculated using the

dosimetric data from ICRP 50 (see references) and Federal Guidance Report No. 11 (EPA-520/1-88-020). For the direct external exposures, EDEs were estimated using MicroShield (Computer Codes, Grove Engineering, 1996).

Table 3.4.5 also presents estimates of potential individual doses from mantle replacement (see Section 3.4.4.1.2) and from storage of camping lanterns in the home. Annual EDEs were calculated to be 6×10^{-4} mSv (0.06 mrem) from home storage of a camping lantern with one mantle (by one-time campers) and 0.002 mSv (0.2 mrem) from home storage of eight mantles (by avid campers). For both avid and one-time campers, the EDE for the mantle replacement dominates the annual EDEs given in Table 3.4.5. These calculations assume that a house has a volume of 450 m³ and a ventilation rate of 1 volume change per hour (see Appendix A.1) and that campers are in their homes for 200 effective days or a total of 4,800 h/yr (NUREG/CR–5512).

To estimate the collective dose to members of the public from routine use of mantles in camping lanterns, the annual individual doses in Table 3.4.5 for each of the camper types was multiplied by the number of campers of that type (one principal camper and three other campers) and divided by the number of mantles each type of principal camper purchased and burned during the year (one by one-time principal campers and eight by avid principal campers). The resulting value of 1×10^{-5} person-Sv (0.001 person-rem) per mantle was then multiplied by the number of mantles used for camping, approximately 20.25 million, to give an annual collective EDE of about 400 person-Sv (40,000 person-rem).

3.4.4.2.2 Indoor Residential Lamps

Indoor lamps containing mantles may be used functionally (e.g., in vacation cabins and cottages) and decoratively (e.g., in residential homes). No data are available concerning the relative magnitude of the two uses. O'Donnell and Etnier (1981) assumed that 75% were used functionally in vacation cottages and 25% were used decoratively in permanent residences.

The mantles in decorative lamps will be used only occasionally, and they are assumed to behave primarily like mantles in camping lanterns stored in a home (O'Donnell and Etnier,1981). The average annual EDE from a decorative lamp in a permanent residence containing two mantles is estimated to be about 0.2 mSv (20 mrem). This dose estimate is based on the following assumptions: (1) 25% of the ²²⁰Rn contained in two mantles emanates from the lamp and disperses into the home air; (2) the exposed persons are in the residence for 200 effective days per year (4800 h/yr) (NUREG/CR–5512); (3) the volume of the residence is 450 m³, and the ventilation rate is 1 volume change per hour.

The mantles in functional lamps will be used regularly, and the annual EDE from a functional lamp in a vacation cottage is calculated using the following assumptions: (1) two mantles are purchased and replaced each year; (2) the vacation cottage is used for 52 days (26 weekends) and is occupied for 12 h/day; (3) the lamp is actually lighted for 4 of the 12 hours; and (4) the vacation cottage home has a volume of 225 m³ (one-half of the volume of a residence) and a ventilation rate of 1 volume change per hour. The person who replaces the two mantles during the year receives an additional annual EDE of 0.002 mSv (0.2 mrem). The average annual effective dose from use of mantles in a vacation cottage is 0.1 mSv (10 mrem).

If there are two mantles per residence and the average residence is occupied by four persons, use of 1.25 million mantles could expose 640,000 persons in approximately 160,000 permanent residences and 1.86 million persons in approximately 465,000 vacation cottages or cabins. Thus the annual collective EDE to the public from indoor residential lamps is estimated to be about 300 person-Sv (30,000 person-rem).

There is a special scenario regarding the continuing use of mantles in a permanent residence as the only source of lighting in remote areas. The mantles in four functional lamps are used regularly, and the annual EDE from use is calculated to be 2 mSv (200 mrem). This dose estimate is based on the following assumptions: (1) 25% of the ²²⁰Rn contained in four mantles emanates from the lamp and disperses into the home air; (2) the exposed persons are in the residence for 200 effective days per year (4800 h/yr) (NUREG/CR–5512); (3) the volume of the residence is 500 m³, and the ventilation rate is 1 volume change per hour; (4) during six colder months, the lanterns burn for 10 hours per day; and during six warmer months, the lanterns burn for 2 hours per day. It is unknown how many people actually use mantles as their primary source of lighting; however, the hypothetical annual EDE for four people exposed in each of a nominal 1000 residences would be about 8 person-Sv (800 person-rem).

3.4.4.2.3 Outdoor Gaslights

Outdoor residential gaslights are usually located several meters from residences (NUREG/CR–1910). For a gaslight containing two mantles that is located 18 meters from three homes and swept by a 3-km/h breeze, essentially all of the radiation dose can be attributed to direct external exposure to the mantles contained in the gaslights. Thus, based on the previous results of O'Donnell and Etnier (NUREG/CR–1910), it was estimated that the annual EDE from external irradiation of the whole body of an exposed individual would be less than 1×10^{-5} mSv (<0.001 mrem).

If 1.5 million outdoor gaslights are used to house the 3 million mantles and that 12 persons from the two homes are exposed to the mantles in each gaslight, the collective EDE could be 0.05 person-Sv (5 person-rem). It is assumed here that the collective doses from residential outdoor gaslights and commercial outdoor gaslights are similar. More people would be exposed to commercial outdoor gaslights, but the individual doses would be significantly less since the people would be exposed for much shorter periods of time.

3.4.4.3 Disposal

This section provides estimates of individual and collective doses to members of the public from landfill disposal and by incineration of thorium-containing mantles. These dose estimates are based on the generic disposal methodology in Appendix A.2 of this report.

Used mantles may be discarded as domestic solid waste or as litter at mantle replacement sites (NUREG/CR-1910). The potential doses to campers from used thorium-containing mantles discarded as litter is one of the exposure events considered during a typical 2-night camping trip by an avid camper (see Table 3.4.4). It is assumed here that 10 million mantles per year are discarded as litter at camping sites or other mantle replacement sites and 15 million mantles per year are discarded as domestic solid waste. The amount of thorium contained in the 15 million mantles disposed of as solid waste at landfills and by incineration would be 3.75×10^6 g, and it would be in a form that is readily dispersible in air and readily accessible to

infiltrating ground water at landfills. Thus, the dose-to-source ratios for individual and collective doses in Appendix A.2 are used without modification.

For disposal at landfills, the annual individual EDE would be about 0.01 mSv (1 mrem) to either waste collectors or future on-site residents. Annual individual doses to workers at landfills and off-site members of the public would be substantially less. The total collective dose to members of the public was found to be about 10 person-Sv (1000 person-rem), due almost entirely to exposure to future on-site residents at landfills for 1000 years after loss of institutional controls over the sites. If exposure to future on-site residents is not considered, the collective EDE would be 0.07 person-Sv (7 person-rem), due mainly to exposure to waste collectors and workers at landfills.

For disposal by incineration, the annual individual EDE would be about 0.07 mSv (7 mrem) to waste collectors. Annual individual doses are substantially less to workers at incinerators and off-site members of the public. The total collective dose to members of the public would be about 0.01 person-Sv (1 person-rem), due mainly to exposure to waste collectors at incinerators.

3.4.4.4 Accidents and Misuse

This section considers: (1) the internal dose to a small child who ingests part of a used gas mantle, (2) the external dose to a person who uses a pack of 2 gas mantles as a calibration source, and (3) the total internal and external dose to individuals from a residential fire involving 8 gas mantles (4 mantles installed in lanterns and 4 replacement mantles) and a warehouse fire involving 10,000 gas mantles.

For ingestion by a small child, it was assumed the child would find a discarded mantle at a camp site and digest part of the mantle while playing with it (NUREG/CR-1775). If 20% of the ash from the used mantle (50 mg of thorium) adheres to the child's hands, if subsequent wiping and washing removes all but 5% of the adhered ash (2.5 mg of thorium), and if the 2.5 mg of thorium is ultimately ingested by the child, the child could receive an EDE of about 0.01 mSv (1 mrem), based on dose conversion factors for an adult (see Section 3.1), or 0.02 mSv (2 mrem), based on age-dependent dose conversion factors for a 1- to 5-year-old child (ICRP 69).

For misuse as a calibration source, it was assumed an individual holds the mantles in his or her hands for about 10 minutes per day or 40 h/yr while calibrating radiation survey instruments for photons and beta particles. The dose equivalent rate at 1 meter from a package with two mantles containing 500 to 600 mg (250 to 300 mg per mantle) of 6-month-old thorium is well established as $2 \times 10^{-4} \,\mu$ Sv/h (0.02 μ rem/h) (Luetzelschwab and Googins, 1984). Actual measurements of the dose equivalent rate at the surface of a pack with two mantles containing 300 mg (150 mg per mantle) of thorium have been reported by O'Donnell and Etnier (NUREG/CR–1910). The dose rate near the surface of the package from beta particles and photons was approximately 0.01 mSv/h (1 mrem/h), and the photon dose rate was estimated to be 0.001 mSv/h (0.1 mrem/h). If these dose rates are normalized to a pack with two mantles containing 500 to 600 mg (250 to 300 mg per mantle), then the dose equivalent rates at the surfaces of the pack from beta particles and photons would be 0.02 mSv/h (2 mrem/h), and the photon dose rate would be approximately 0.002 mSv/h (0.2 mrem/h). Thus, the annual individual EDE from external irradiation of the total body by photons would be less than

 1×10^{-5} mSv/h (<0.001 mrem), and the annual dose equivalent to the hands from both beta particles and photons could be 0.08 mSv (8 mrem).

For accidents involving fire, a residential fire involving 8 gas mantles and a warehouse fire involving 10,000 gas mantles were considered. A release fraction of 0.1% is assumed for the radionuclides in each mantle which is assumed further to contain 250 mg of 6-month-old thorium (see Table 3.4.1). Based on these assumptions and the generic accident methodology in Appendix A.1, the estimates of individual dose from fires involving thorium-containing mantles are summarized as follows:

- For a person trying to escape from a residential fire or a neighbor trying to rescue a person from a residential fire, the individual EDE from eight mantles containing a total of 2 g of thorium could be 0.002 mSv (0.2 mrem).
- For a firefighter wearing a respirator at a residential fire, the individual EDE from eight mantles containing a total of 2 g of thorium could be less than 1×10⁻⁵ mSv (<0.001 mrem). The individual EDE could be 2×10⁻⁵ mSv (0.002 mrem) to a worker who is involved in the cleanup following the fire and does not wear a respirator.
- For a firefighter wearing a respirator at a warehouse fire, the individual effective dose from 10,000 mantles containing a total of 2.5 kilogram (kg) of thorium (in equilibrium with its daughters) could be 0.001 mSv (0.1 mrem). The individual EDE could be 0.005 mSv (0.5 mrem) to a worker who is involved in the cleanup following the fire and does not wear a respirator.

3.4.5 Summary

Table 3.4.2 presents the results of the current assessment of potential radiation doses to the public for an annual distribution, use, and disposal of 25 million gas mantles containing thorium. It is assumed each gas mantle contains 250 mg of 6-month-old thorium (see Table 3.4.1). However, Buckley et al. (NUREG/CR-1775) reported that the thorium content could be as high as 400 mg. In this case, the estimated individual doses for routine use, accidents, and misuse would be about two times greater than the doses for mantles containing 250 mg of thorium.

For routine use, the most highly exposed individual (a resident whose only source of lighting was the use of mantles) was estimated to receive an annual EDE of 2 mSv (200 mrem). The total collective dose to the public was estimated to be 700 person-Sv (70,000 person-rem).

For misuse, this analysis estimated an individual EDE of 0.02 mSv (2 mrem) to a small child who plays with a used mantle and subsequently ingests part of the ash from the mantle. The individual effective doses were found to be less for other potential misuse and accident scenarios.

Previous studies (NCRP 95) estimate a total population dose of about 90 person-Sv (9000 person-rem) to the public from gas mantles containing thorium, and this assessment finds a total population dose that is three times larger. These differences can be attributed to the following factors:

- Dose equivalents to the lungs and to bone (the red marrow in bone and the endosteal cells on the bone surfaces) were not treated properly by the older dosimetry used in the previous studies (EPA-520/1-88-020).
- Previous estimates of collective dose were based on averages over rather large dose ranges. Point values used in the current estimates of collective dose tend to be closer to the maximum dose values calculated in the previous studies (NUREG/CR-1910).
- The standard assumption used in estimating collective doses in this study is four individuals per family. Previous studies assumed only three individuals per family (NUREG/CR-1910).

Radionuclide	Activity in Natural Thorium (nCi) ^a	Activity in 6-Month-Old Thorium (nCi) ^a
²³² Th	27	27
²²⁸ Ra	27	2
²²⁸ Ac	27	2
²²⁸ Th	27	23
²²⁴ Ra	27	23
²²⁰ Rn	27	23
²¹⁶ Po	27	23
²¹² Pb	27	23
²¹² Bi	27	23
²¹² Po	17	15
²⁰⁸ TI	10	8
Total	270	192

Table 3.4.1 Radioactivity in a Gas Mantle Containing 250 mg of Natural Thorium orThorium That Has Been Chemically Extracted From Ore and Aged for 6 Months

^a 1 nCi = 37 Bq.

Exposure Pathway	Individual Annual Effective Dose Equivalent (mrem) ^ь	Collective Effective Dose Equivalent ^a (person-rem) ^b
<u>Distribution and transport</u> Transport Distribution	0.2° 50⁴	1 700
<u>Routine use</u> Portable camping lanterns Indoor residential lamps Permanent cabin Outdoor gas lights	10° 20 200 <0.001 ^f	40,000 30,000 800 5
<u>Disposal</u> Landfills Incineration	1 ^g 7 ^h	1,000 1
Accidents and misuse Use as a calibration source Fire Ingestion of mantle ash by small child	0.008 ⁱ 0.5 ⁱ 2 ^k	

Table 3.4.2 Potential Radiation Doses From Incandescent Gas Mantles Containing Thorium

^a Collective doses are based on the annual distribution, use, and disposal of 25 million gas mantles containing 250 mg each of thorium.

^b 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

^c Dose estimate applies to truck drivers; dose estimates are less for workers in truck terminals and for members of the public along truck routes (see Section 3.4.4.1.1).

^d Dose estimate applies to utility company installers; dose estimates are less for distribution warehouse workers, truck drivers, store workers and customers, stock clerks, and members of the public along truck routes (see Sections 3.4.4.1.2 and 3.4.4.1.3).

^e Dose estimate applies to principal avid-camping users of thorium-containing mantles; dose estimates for all campers using thorium-containing mantles are less (see Section 3.4.4.2.1). ^f Dose estimate applies to persons exposed to thorium-containing mantles in outdoor residential gaslights; dose estimates would be less for persons exposed to thorium-containing mantles in outdoor commercial gaslights (see Section 3.4.4.2.4).

⁹ Dose estimate applies to waste collectors or future on-site residents at landfills; dose estimates are less for workers at a landfill and off-site members of public (see Section 3.4.4.3). ^h Dose estimate applies to waste collectors at incinerators; dose estimates would be less for workers at incinerators and off-site members of public (see Section 3.4.4.3).

ⁱ Dose estimate applies to photon irradiation of total body of a person using two mantles in a pack as a calibration source; dose estimate for beta-particle and photon dose to hands is 0.06 mSv/yr (6 mrem/yr).

¹ Dose estimate applies to a worker who is involved in the cleanup following a warehouse fire and does not wear a respirator; dose estimates would be less for a firefighter involved in a warehouse fire and for a firefighter or others involved in a residential fire (see Section 3.4.4.4).

^k Dose estimate applies to 1- to 5-year-old child; dose estimates would be less for older children or adults (see Section 3.4.4.4).

	Effective Dose Equivalent		
Exposure Condition	Outdoor Changes (mrem) ^a	Indoor Changes (mrem) ^a	
1 ^b	<0.001	0.02	
2°	2	2	
3 ^d	2	4	

Table 3.4.3 Summary of Potential Radiation Doses From Changing Two Mantles

a 1 mrem = 0.01 mSv.

^b Condition 1: Includes both external irradiation of the whole body by photons from mantles and inhalation of radionuclides from diffusion of ²²⁰Rn from mantles (see Section 3.4.3).
 ^c Condition 2: Includes Condition 1 plus ingestion of ash from used mantles (see Section 3.4.3).
 ^d Condition 3: Includes Condition 2 plus inhalation of ash from blowing used mantle parts from a

lighting device (see Section 3.4.3).

Exposure Event ^a	Duration of Exposure	Number of Mantles	Average Distance From Mantles (meters)	Identity of Absorbers	Sources of Airborne Radionuclides
		FIR	ST DAY		
Purchase	1 h	4	1	None	None
Load car	0.5 min	4	1	Lantern	None
		4	1	Box	None
Travel to camp					
Front seat	1.5 h	4	2	Lantern, seats, cargo	Diffusion
		4	2	Box, seats, cargo	None
Back seat	1.5 h	4	1	Lantern, seats, cargo	Diffusion
		4	1	Box, seats, cargo	None
Unload car	0.5 min	4	1	Lantern	None
		4	1	Box	None
Set up camp	2 h	2	3	Lantern	Diffusion
		2	6	Lantern	Diffusion
		4	3	Box	None
Ignite lanterns	1 min	4	1	Lantern	Vaporization & diffusion
		2	6	Lantern	Vaporization &
		4	6	Box	None
Under light	3 h	4	3	Lantern	Diffusion
outside		4	3	Box	None

Table 3.4.4 Exposure Conditions for a 2-Night Camping Tripfor an Avid Camper and Family

Exposure Eventª	Duration of Exposure	Number of Mantles	Average Distance From Mantles (meters)	ldentity of Absorbers	Sources of Airborne Radionuclides
		FIRST	DAY		
Inside tent Lantern on	0.5 h	2 2 4	1 6 6	Lantern Lantern Box	Diffusion None None
Lantern off	7.5 h	2 2 4	1 6 6	Lantern Lantern Box	Diffusion None None
		SECONE	DAY	·····	
At site	6 h	4 4	6 6	Lantern Box	Diffusion None
Ignite lanterns (same as above) Under lights outside (same as above) Inside tents (same as above)					
**** <u>-</u>		THIRD	DAY	········	
At site (same as above) Load car (same as above) Travel home (same as travel to camp) Unload car (same as above)					

Table 3.4.4 Exposure Conditions for a 2-Night Camping Tripfor an Avid Camper and Family (continued)

^a It is also assumed each person is exposed to ground contamination of 0.6 μ g/cm² from discarded mantles for a total of 38 hours during stay at camping site.

	One-Time	e-Time Campers Avid Campe		ampers
Exposure Events	Principal (mrem) ^ь	Others ^a (mrem) ^b	Principal (mrem)⁵	Others ^a (mrem) ^b
Camping activities	0.06	0.002	6	0.05
Mantle replacement	1		8	
Home storage	0.06	0.06	0.2	0.2

Table 3.4.5 Estimates of Annual Effective Dose Equivalent for One-Time and Avid Campers

^a Includes other family members. ^b 1 mrem = 0.01 mSv.

- • •

3.5 Thorium in Vacuum Tubes

3.5.1 Introduction

In 10 CFR 40.13(c)(1)(ii), any person is exempt from the requirements for a license to the extent that the person receives, possesses, uses, or transfers any quantity of thorium contained in vacuum tubes. This exemption was established on March 20, 1947 (12 FR 1855), and has remained essentially unchanged since then.

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

3.5.2 Description of Items

Many vacuum tubes used in applications requiring high output powers contain directly heated cathodes (Ewell, 1981). These cathodes are frequently made from thoriated-tungsten wires containing 1 to 2% by weight of thorium. The thorium is used because of its ability to emit electrons at relatively low temperatures when heated in a vacuum (Davis, 1977).

The vacuum tubes containing thoriated-tungsten cathodes are of many varied designs (Davis, 1977). High-output power tubes may range from tens to hundreds of centimeters in dimension and from a few to 150 kg in mass. The cathodes may have masses between 100 μ g and 100 g and contain between 1 μ g and 2 g of thorium. Tube envelopes may be metal, glass, ceramic, or any combination thereof. The envelopes will normally contain any ²²⁰Rn that diffuses from the thoriated-tungsten wire.

The use of vacuum tubes in electronics has been in decline since the development of the transistor and other solid-state devices after World War II (Liao, 1985). In spite of the success of solid-state devices in most electronics applications, there are certain specialized functions that only vacuum tubes can perform (Liao, 1985; Parker and Abrams, 1992). These functions usually involve operation at extremes of power or frequency. However, sales of such vacuum tubes during the 1980s and beyond are estimated to be 100,000 units per year or less (Ewell, 1981; Garoff, 1979).

The current market for vacuum tubes is dominated by the magnetrons in microwave ovens sold for home and restaurant use. It is currently estimated that more than 90% of all U.S. homes own at least one microwave oven (Reda, 1995; Kulman, 1997). A single magnetron is typically used in the low-wattage ovens for home use (i.e., 600 to 1000 watts) (Microwave Ovens, 1996), while as many as three or four magnetrons may be used in the heavy-duty, high-wattage ovens for restaurant use (i.e., 2000 to 2600 watts). Sales of such magnetrons are estimated to be approximately 10 million units per year.

The typical cathode in magnetrons for home use has 8 to 10 turns of thoriated-tungsten wire on a diameter of 200 mils, the diameter of the wire is 15 to 20 mils, and the amount of thorium in the wire is 1 to 2% by weight (Phone call, J. Osepchuk, Full Spectrum Consulting, Concord,

MA, June 1997). If the density is taken as 19.3 g/cm³ for tungsten and 11.2 g/cm³ for thorium, the mass of the thoriated-tungsten wire ranges from 300 to 600 mg. The typical wire mass in a microwave oven magnetron is about 400 mg per tube, and the amount of thorium is typically 4 to 8 mg per tube (Phone call, Amana Consumer Affairs, Amana Refrigeration, Inc., Amana, IA, 55204, June 1997).

3.5.3 Summary of Previous Analyses and Assessments

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

3.5.4 Present Exemption Analysis

Table 3.5.1 presents results of the current assessment of potential radiation doses from vacuum tubes containing thorium. These results are based on an annual distribution of 10 million magnetrons in microwave ovens containing 8 mg of natural thorium each, and an annual distribution of 100,000 other types of vacuum tubes containing an average of 0.25 g of natural thorium each. The effective lifetime of both the magnetrons used in microwave ovens and the vacuum tubes used in other electronic applications is assumed to be 10 years.

3.5.4.1 Distribution and Transport

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

3.5.4.1.1 Magnetrons in Microwave Ovens

In applying the generic methodology in Appendix A.3 to magnetrons in microwave ovens, the following assumptions have been made. The microwave ovens, each containing one magnetron, are assumed to be shipped from the manufacturers or importers to distribution centers. The number of microwave ovens in a typical shipment is assumed to be 1000 magnetrons containing 8 mg of natural thorium each. It is assumed further that commercial semi-trucks are used to ship the microwave ovens between distribution centers, and microwave ovens pass through an average of three distribution centers before being delivered to large retail stores. In addition, it is assumed that (1) a semi-truck driver may pick up a shipment as often as once per week (50 shipments per year) from the same manufacturer during the year, (2) a stockhandler in a distribution center may be exposed to 3 shipments of microwave ovens per week (150 shipments per year), and (3) a store clerk may be exposed continuously to a display of 20 microwave ovens in a retail store.

Based on the above assumptions and the generic methodology in Appendix A.3, the annual individual effective dose equivalent (EDE) would be less than 1×10^{-5} millisievert (mSv) (<0.001 mrem) to a stockhandler in a large distribution center (i.e., large warehouse). Individual doses would be less to semi-truck drivers, stockhandlers in retail stores, store clerks and customers in retail stores, and members of the public along truck routes. The collective EDE

from all distribution and transportation activities could be 0.2 person-Sv (20 person-rem), due almost entirely to exposure to retail store clerks and customers.

3.5.4.1.2 Other Types of Vacuum Tubes

In applying the generic methodology of Appendix A.3 to vacuum tubes used in other applications, the following assumptions have been made. The vacuum tubes are assumed to be shipped from the manufacturer or importer to the customer by commercial semi-truck, and the number of vacuum tubes in a typical shipment is assumed to be 10 tubes containing 0.25 g of natural thorium each. It is assumed further that (1) the vacuum tubes pass through an average of three truck terminals before reaching their final destination, (2) the exposure to workers in a truck terminal is the same as those estimated for workers in a large warehouse, and (3) the same semi-truck driver could pick up a shipment as often as once per week (50 shipments per year) from the same manufacturer.

Based on the above assumptions and the generic methodology in Appendix A.3, the annual individual EDE would be less than 1×10^{-5} mSv (<0.001 mrem) to a semi-truck driver who picks up a number of shipments from the same manufacturer during the year. Individual doses would be less to other semi-truck drivers, workers in truck terminals, and members of the public along truck routes. The collective EDEs from all distribution and transportation activities could be 1×10^{-4} person-Sv (0.01 person-rem).

3.5.4.2 Routine Use

This section estimates potential radiation doses to routine users from magnetrons and other vacuum tubes containing thorium. The exposure scenarios used here may be somewhat out of date, particularly for vacuum tubes other than magnetrons in microwave ovens, but the results are still considered to be indicative of the potential radiation doses from typical routine use of vacuum tubes in radio and TV broadcasting and in various radar applications. Routine users will not receive any internal doses because ²²⁰Rn should not diffuse through the walls of the vacuum tubes. Exposures will be from photons emitted by thorium and its radioactive decay products (see Section 3.1).

3.5.4.2.1 Magnetrons in Microwave Ovens

Because of concern over microwave leakage from early microwave ovens, surveys were made of the spatial distance of a microwave oven user as a function of operating time (Osepchuk, 1979). Results of these surveys can be summarized as follows: (1) typical microwave oven usage in the home involved about seven starts per day, with 2- to 3-minute cooking times per operation, (2) distance from the microwave oven during the preparation of a meal ranged from 0.7 meter for a 1-minute operation to 2 meters for a 30-minute or longer operation, and (3) the most probable average distance from the microwave oven during the preparation of a meal was 1.2 to 1.5 meters.

A typical magnetron in a microwave oven for home use has a wall thickness of approximately 0.2 cm of copper and contains 8 mg of thorium, which is assumed to be 20-year-old natural thorium (see Section 3.1). To estimate potential radiation doses to users of home microwave ovens, it is assumed that the principal user is exposed at a distance of 0.7 meter for 5 min/day during two short cooking operations and at an average distance of 1.2 meters for 1 h/day during

meal preparation, and other family members are exposed for 5 min/day at an average distance of 0.7 meter during two short cooking operations (i.e., one of these could be during lunchtime at work and the other during a snacktime at home). All family members also are assumed to be exposed at an average distance of 3 meters for 30 min/day during breakfast and dinner. Thus, the individual effective dose to the principal microwave user and to other family members could be less than 1×10^{-5} mSv (<0.001 mrem), based on calculations with MicroShield (Computer Codes, Grove Engineering, 1996). The annual collective dose over the 10-year effective lifetime of the ovens could be 0.4 person-Sv (40 person-rem).

The above estimates of individual dose apply to home users of low-wattage microwave ovens with a single magnetron. The maximum dose scenario for a high-wattage, heavy-duty microwave oven containing four magnetrons (see Section 3.5.2). It is assumed that a restaurant cook is exposed at an average distance of 1.2 meters for 1000 h/yr (i.e., 4 h/day for 250 day/yr) during the preparation of meals. The annual individual EDE to the cook could be 2×10^{-5} mSv (0.002 mrem) based on calculations with MicroShield (Computer Codes, Grove Engineering, 1996).

3.5.4.2.2 Other Types of Vacuum Tubes

Because vacuum tubes other than the microwave oven magnetron vary widely in design and in application, it is not feasible to attempt a detailed assessment of all potential applications. Thus, for this analysis it was decided to indicate the potential radiation doses using the following representative scenarios: Scenario I, an individual exposed at an AM broadcasting station to 10 vacuum tubes containing 0.1 g of thorium each; and Scenario II, an individual exposed at a large radar installation to 22 vacuum tubes containing 0.4 g of thorium each. It is assumed here that 50,000 of the 100,000 such vacuum tubes distributed per year are used in radio and TV broadcasting and the other 50,000 are used in radar surveillance systems.

Scenario I. An AM radio broadcasting station with a maximum power of 50 kW contains five operating tubes and five spare tubes. Each tube contains 0.1 g of thorium, which is assumed to be 20-year-old natural thorium (see Section 3.1). The tube envelopes are cylinders of aluminum with a wall thickness of 0.2 cm, and all tubes are in an equipment room that is separated from the operator by a cinder block wall containing a 2.5-cm-thick glass window. The operating tubes also are enclosed in a steel cabinet with a wall thickness of 0.1 cm.

One operator is assumed to work in the control room for 2000 h/yr at an average distance of 1 meter from each tube, and four operators are required for continuous broadcasting. Accounting for photon absorption in the aluminum cylinders, glass window, and steel cabinet (for the five operating tubes), each operator is estimated to receive an annual individual EDE of 0.002 mSv (0.2 mrem) based on calculations with MicroShield (Computer Codes, Grove Engineering, 1996). The annual collective dose from 50,000 tubes could be 0.04 person-Sv (4 person-rem), and the total collective dose over the assumed 10-year effective lifetime of the vacuum tubes could be 0.4 person-Sv (40 person-rem).

Scenario II. A large radar surveillance station contains 11 operating vacuum tubes and 11 spare tubes. Each tube contains 0.4 g of thorium, assumed to be natural thorium. The tube envelope is inside a 0.75-cm-thick ceramic material that is covered by a 0.2-cm-thick aluminum cylinder. The operating tubes are also enclosed in a 0.1-cm-thick iron cabinet.

A five-person crew operates the station and rotates duties so that each operator is in the control room about 5 hours per shift. Continuous operation of the station requires four operating crews. Hence, each operator is assumed to be in the control room for about 1300 h/yr. Approximately 25% of this duty time (325 h/yr) is spent working at an average distance of 0.7 meter from 2 operating tubes, 2 meters from the other 9 operating tubes, and 3 meters from the 11 spare tubes. During the remainder of the duty time (975 h/yr), each operator is assumed to be exposed at an average distance of 3 meters from all 22 tubes. Each operator is estimated to receive an annual individual EDE of 0.002 mSv (0.2 mrem) based on calculations with MicroShield (Computer Codes, Grove Engineering, 1996). The annual collective dose from 50,000 tubes could be 0.09 person-Sv (9 person-rem). The total collective dose over the assumed 10-year effective lifetime of the vacuum tubes could be 0.9 person-Sv (90 person-rem).

3.5.4.2.3 All Types of Vacuum Tubes

As mentioned previously, the above exposure scenarios for vacuum tubes other than the magnetrons in home microwave ovens may be outdated because they were developed from data relevant to the early 1980s. Nevertheless, the results are still considered to be indicative of potential radiation doses from the routine use of vacuum tubes in radio and TV broadcasting and in various radar applications. The total collective dose from routine use of 1 year's distribution of all types of vacuum tubes containing thorium is estimated to be 2 person-Sv (200 person-rem).

3.5.4.3 Disposal

· · · ·

This section estimates potential radiation doses from disposal of magnetrons in microwave ovens and other types of vacuum tubes based on the generic disposal methodology in Appendix A.2. The amount of natural thorium in the magnetrons of microwave ovens is assumed to be 80 kg, and the amount of thorium in other types of vacuum tubes is assumed to be 25 kg. Further, it is assumed here that all of the microwave ovens and equipment containing the other vacuum tubes are sent to landfills for disposal and that the thorium remains within the tubes during landfill disposal. Thus, the following adjustments are made to the source-to-dose ratios in Appendix A.2: (1) there is no inhalation or ingestion by waste collectors or workers at a landfill, and (2) there is no exposure to off-site members of the public due to airborne releases during landfill operations.

Based on the above assumptions and the generic disposal methodology in Appendix A.2, the individual and collective doses can be summarized as follows:

- The annual individual EDEs to waste collectors at landfills could be 2×10⁻⁴ mSv (0.02 mrem) from microwave oven magnetrons, 5×10⁻⁵ mSv (0.005 mrem) from other types of vacuum tubes. Individual doses to workers at landfills, off-site members of the public, and future on-site residents would be less.
- The collective EDEs could be 0.3 person-Sv (30 person-rem) from microwave oven magnetrons, 0.1 person-Sv (10 person-rem) from other types of vacuum tubes, and 0.4 person-Sv (40 person-rem) from both sources, due almost entirely to exposure to future on-site residents at landfills for 1000 years after the loss of institutional controls over the sites.
If the exposure to future on-site residents is not considered, the collective dose would be 4×10^{-4} person-Sv (0.04 person-rem) from microwave oven magnetrons, 1×10^{-4} person-Sv (0.01 person-rem) from other types of vacuum tubes, and 5×10^{-4} person-Sv (0.05 person-rem) from both sources, due mainly to exposure to workers at landfills.

3.5.4.4 Accidents and Misuse

This section estimates the potential radiation doses from accidents and misuse of vacuum tubes using the generic accident methodology in Appendix A.1.

In the case of misuse, the exposure to a person who finds and vandalizes a large vacuum tube containing 2 g of natural thorium (see Section 3.5.2) is considered. The person is assumed to destroy the vacuum tube out of curiosity, inhaling radioactive material released from the tube (i.e., ²²⁰Rn and its short-lived progeny). It is assumed further that the person is exposed for 30 minutes in a small room with an enclosed volume of 18 m³ and a ventilation rate of 1 volume change per hour (see data for a small watch repair shop in Tables A.1.2 and A.1.9 of Appendix A.1), and the thorium is 20-year-old thorium and the amount of ²²⁰Rn in the tube is 8 kilobecquerels (kBq) (0.22 microcurie (μ Ci)) (i.e., all of the ²²⁰Rn and daughters escape from the cathode of the tube). Based on these assumptions and the generic accident methodology of Appendix A.1, the individual EDE to this person could be 0.01 mSv (1 mrem). For a microwave oven size tube, the individual EDE could be less than 1×10⁻⁵ mSv (<0.001 mrem).

In the case of accidents, the following assumptions have been made: (1) a residential fire involving a single microwave oven magnetron containing 8 mg of natural thorium, (2) a transportation fire involving a typical shipment of 1000 microwave ovens containing 8 mg of natural thorium each, and (3) a typical shipment of 10 vacuum tubes of other types containing 0.25 g of natural thorium each. It is also assumed that the release factor for the thorium contained in the microwave oven magnetrons and other types of vacuum tubes is 0.01%. Based on these assumptions and the generic accident methodology in Appendix A.1, the individual doses can be summarized as follows:

- For a person trying to escape from a residential fire or a neighbor trying to rescue a
 person from a residential fire, the individual EDE from a microwave oven magnetron
 containing 8 mg of 20-year-old natural thorium could be less than 1×10⁻⁵ mSv
 (<0.001 mrem).
- For a firefighter wearing a respirator at a residential fire, the individual EDE from a microwave oven magnetron containing 8 mg of natural thorium could be less than 1×10⁻⁵ mSv (<0.001 mrem). The individual EDE could be less than 1×10⁻⁵ mSv (<0.001 mrem) to a worker who is involved in the cleanup following the fire and who does not wear a respirator.
- For a firefighter wearing a respirator at a transportation fire, the individual EDE from a shipment of 1000 microwave oven magnetrons containing a total of 0.8 g of natural thorium could be 2×10^{-5} mSv (0.002 mrem). The individual EDE from a shipment of 10 vacuum tubes containing a total of 2.5 g of natural thorium could be less than 1×10^{-5} mSv (<0.001 mrem).

 For a worker who is involved in the cleanup following a transportation fire and who does not wear a respirator, the individual EDE from the shipment of 1000 microwave oven magnetrons could be 2×10⁻⁴ mSv (0.02 mrem). The individual EDE from the shipment of 10 vacuum tubes of other types could be 5×10⁻⁵ mSv (0.005 mrem).

3.5.5 Summary

Table 3.5.1 presents results of the current assessment of the potential radiation doses to the public from vacuum tubes containing thorium. These results are based on an assumed annual distribution of 10 million magnetrons in microwave ovens containing 8 mg of natural thorium each and 100,000 other types of vacuum tubes containing an average of 0.25 g of natural thorium each. The effective lifetime of both the magnetrons used in microwave ovens and the vacuum tubes used in other electronic applications is assumed to be 10 years.

For routine use of these vacuum tubes, including distribution and disposal in landfills, the annual EDE could be 0.002 mSv (0.2 mrem) to a routine user, and the total collective dose from 1 year's distribution of vacuum tubes containing thorium could be 3 person-Sv (300 person-rem), due mainly to exposure to individuals during routine use. For routine use of microwave ovens in the home, the annual individual effective dose was estimated to be less than 1×10^{-5} mSv (<0.001 mrem).

For an accident involving fire, the individual EDE could be as much as 5×10^{-5} mSv (0.005 mrem), and for misuse involving vandalism of a vacuum tube, the individual effective dose could be 1×10^{-4} mSv (0.01 mrem). Thus, the potential radiation doses from accidents and misuse appear to be very low.

Exposure Pathway	Individual Annual Effective Dose Equivalent (mrem) ^b	Collective Effective Dose Equivalent ^a (person-rem) ^b
Distribution and transport	<0.001°	20
Routine use Radar systems Microwave oven Total	0.2 ^d <0.001 ^e	200 ^f
Landfill disposal	0.02 ⁹	40
Accidents and misuse Fire Vandalism of vacuum tube	0.02 ^h 1 ⁱ	

Table 3.5.1 Potential Radiation Doses From Vacuum Tubes Containing Thorium

^a Collective doses are based on an assumed annual distribution of 10 million magnetrons in microwave ovens containing 8 mg of natural thorium each and 100,000 other high-powered vacuum tubes containing 0.25 g of natural thorium each. A 10-year effective lifetime also is assumed for both the magnetrons and other vacuum tubes. Refer to text discussion for time period for collective dose assessment.

^b 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

^c Dose estimate applies to a stockhandler in a distribution center or large warehouse; dose estimates are less for semi-truck drivers, workers and customers in retail stores, and members of the public along truck routes (see Section 3.5.4.1.1).

^d Dose estimate applies to operators at radio and TV broadcasting stations or at radar surveillance systems (see Section 3.5.4.2.2)

^e Collective dose from routine use of one year's distribution of all types of vacuum tubes containing thorium (see Section 3.5.4.2.3).

^t Dose estimate applies to use as of magnetrons in microwave ovens for homes (see Section 3.5.4.2.1). Dose estimate for a cook in a restaurant is higher (see Section 3.5.4.2.1). ⁹ Dose estimate applies to waste collectors at landfills; dose estimates are less for workers at landfills, off-site members of the public, and future on-site residents at landfills (see Section 3.5.4.3).

^h Dose estimate applies to a worker who is involved in cleanup following a transportation fire; dose estimates are less for firefighters and others involved in a residential or transportation fire (see Section 3.5.4.4).

ⁱ Dose estimate applies to inhalation of ²²⁰Rn and its short-lived progeny by a person who finds and vandalizes a vacuum tube containing 2 g of natural thorium (see Section 3.5.4.4).

3.6 Welding Rods Containing Thorium

3.6.1 Introduction

In 10 CFR 40.13(c)(1)(iii), persons who receive, possess, use, or transfer welding rods containing any amount of thorium are exempted from licensing requirements for source material. An exemption for thoriated tungsten containing not more than 3% by weight of thorium, but with no restrictions on the products or devices in which the exempted material could be used, was published on March 15, 1949 (14 FR 1156). The exemption in its present form, which applies only to welding rods but with no limit on the amount of thorium that may be contained in this product, was proposed on September 7, 1960 (25 FR 8619), and was issued as a final rule on January 14, 1961 (26 FR 284).

The *Federal Register* notices establishing the present exemption for thorium in welding rods cited above states that the exemption would not result in an unreasonable hazard to life or property. However, quantitative information on radiation doses to the public associated with the exemption was not provided. Assessments of radiological impacts on members of the public from distribution, use, and disposal of welding rods containing thorium have been performed by McDowell-Boyer (NUREG/CR–1039) and Buckley et al. (NUREG/CR–1775). The assessment by Buckley et al. relied extensively on the previous results of McDowell-Boyer, which also were adopted in a review by the National Council on Radiation Protection and Measurements (NCRP 95). Results of the previous dose assessments are described in Section 3.6.3.

3.6.2 Description of Items

Tungsten inert-gas (TIG) arc welding is a process in which an electrical arc is struck between an inert, gas-cooled, nonconsumable electrode (i.e., an electrode that does not provide filler material), also called a welding rod, and the metal work pieces. The electrical arc heats the work pieces and causes them to coalesce (i.e., melt together). Most TIG welding machines are used in automated processes that do not require the continuous presence of an operator. However, a significant amount of manual TIG welding, which requires the presence of an operator, also is conducted.

Many electrodes used in TIG welding consist of tungsten wire that contains thorium dioxide (ThO_2) or another metal oxide (e.g., magnesium, lanthanum, cerium, zirconium, or yttrium). The metal oxide additions provide several benefits, including increased electron emissivity, current-carrying capacity, resistance to contamination of the electrode, and an increased useful life by decreasing the rate of electrode erosion and promoting retention of the desired tip geometry. These characteristics result in easier initiation of the arc, promotion of arc stability during welding operations, and a reduction in the frequency of electrode-tip regrinding and electrode replacement.

The TIG welding process is one of the more expensive welding processes and, thus, is used primarily in industries that require high-quality, low-contamination welds. Examples include the aircraft industry, the petrochemical industry, manufacturing of food processing equipment, and nuclear power plant construction and maintenance. Other industries that use TIG welding to a lesser extent include gas and oil piping and ship building (NUREG/CR-1775).

Although the exemption allows any amount of thorium in welding electrodes, the industry standard is nominally 1 to 2% by weight of ThO_2 , and most electrodes in current use appear to contain about 2% by weight of ThO_2 . Thoriated-tungsten electrodes are produced in nine standard diameters between 0.25 and 6.35 mm, and in six standard lengths between 7.6 and 61 cm. In the assessment by McDowell-Boyer (NUREG/CR–1039), a typical electrode was assumed to be 2.4 mm in diameter and 15.2 cm in length, and to contain 0.23 g of thorium.

The number of thoriated-tungsten welding electrodes in current use in the United States is not known. Based on information available at the time, Buckley et al. (NUREG/CR-1775) estimated an annual production of 5.2 million such electrodes. However, the annual production may have declined since that time (Hedrick, 1985; Hedrick, 1991) due to the increased use of other metal oxides in welding rods. The assessment by McDowell-Boyer (NUREG/CR-1039) assumed an annual production of 1 million electrodes containing thorium. The assessment by Jankovic et al. (1999) reported the U.S. annual production at 4 to 5 million.

3.6.3 Summary of Previous Analyses and Assessments

A variety of scenarios for routine distribution, use, and disposal of thoriated-tungsten welding rods, as well as certain scenarios for accidents and misuse, were considered in previous analyses and assessments by McDowell-Boyer (NUREG/CR–1039) and Buckley et al. (NUREG/CR–1775). The thorium content of the welding rods was assumed to be 2% by weight, and the thorium was assumed to have been aged for 20 years after chemical separation. This section summarizes the results of the previous assessments. Unless otherwise noted, the assessment by Buckley et al. was based on the results obtained previously by McDowell-Boyer. Therefore, the results of Buckley et al. are discussed only when the exposure scenarios differ from those assumed by McDowell-Boyer.

Additionally, there are four studies performed in the past decade of the airborne thorium activity caused by thoriated-tungsten welding operations that are considered pertinent to this study and are summarized below.

3.6.3.1 Recent Airborne Activity Measurements

3.6.3.1.1 Study by Ludwig et al.

For routine use of thoriated-tungsten welding rods, Ludwig et al. (1999) conducted field tests in 26 different shops in Germany. About half of the study dealt with electrodes containing twice the amount of thorium currently believed to be in use in the United States (4% versus 2%). Samples included some for both alternating current (AC) and direct current (DC) welding. The airborne activity generated through welding and grinding of electrodes was measured using personal air samplers and stationary air samplers that collected the inhalable (respirable) dust fraction. For welding, a modified air sampler was integrated into the protective shield, and samples were taken inside the shield during work including preparation time, but not grinding. Samples were collected separately during grinding. Volume flow rates were selected to best match inhalable fraction conventions. Ludwig et al. provided data about the presence or absence of suction during welding and grinding, annual working times, the number of instances of grinding per working hour, AC or DC power supply, and the assayed quantities of ²³²Th during

welding as measured by Ludwig et al. ranged from a low of less than 0.1 to 91 millibecquerel (mBq)/m³ (<7×10⁻⁹ to 5×10⁻⁶ microcurie (μ Ci)/m³) with a geometric mean of about 0.9 mBq/m³ (3×10⁻⁸ μ Ci/m³). For measurements where suction was not used (i.e., no local exhaust), the geometric mean ²³²Th airborne level was 1.2 mBq/m³ (3.2×10⁻⁸ μ Ci/m³). The data did not show significant differences in airborne levels for 2% versus 4% thorium rods; however, AC welding yielded results that were about a factor of 30 higher than DC welding. For AC welding, the geometric mean airborne level without local exhaust was 10 mBq/m³ (2.8×10⁻⁷ μ Ci/m³); and for DC welding the geometric mean was 0.34 mBq/m³ (9.2×10⁻⁹ μ Ci/m³).

Ludwig et al. (1999) reported that in a room with volume of about 100 m³, and without any ventilation or suction system, 35 electrodes (45 by weight thorium) were ground in 15 minutes. Their graph indicates an activity concentration for ²³²Th during the grinding of about 180 mBq/m³ (5×10⁻⁶ μ Ci/m³) with the airborne activity concentration decreasing soon after the end of grinding. The only case from the Ludwig et al. (1999) study that involved local exhaust ventilation indicated a reduction factor of 100.

The study estimated levels of ²³²Th intakes for welders from less than 0.1 to 144 Bq/yr ($<2.7\times10^{-3}$ to 3.9 nCi/yr) during welding and from 0.02 to 30.2 Bq/yr (5.4×10^{-4} to 0.82 nCi/yr) during grinding. In six cases the estimated total intake was estimated to exceed the annual limit for intake to the public of thorium in oxide form, as derived from ICRP Publication 71 (ICRP 71), based on the projections from limited sample data. Four of the six cases involved welders, working with alternating current, where the main exposure was caused by the welding process. The other two welders used direct current welding with the grinding causing the estimated intake to exceed the intake limit.

3.6.3.1.2 Study by Vinzents et al.

The study by Vinzents et al. (1994) was performed to estimate airborne levels while grinding thoriated-tungsten welding rods. To provide sufficient sample size, the measurement was made during the grinding of 4% thorium electrodes for 139 minutes by one person. No exhaust hood was used and the jet of the grinding was directed towards the floor. Vinzents determined a ²³²Th respirable airborne mean concentration of 440 mBq/m³ (2×10⁻⁵ μ Ci/m³) (with approximately 0.3 respirable fraction). However, he acknowledged the limited data and believed it to represent a worst case estimate. Vinzents reported that electrodes are typically ground once each day for approximately one minute.

3.6.3.1.3 Study by Crim and Bradley

Crim and Bradley (1995) performed measurements of airborne thorium concentrations during the grinding of 2% thorium oxide welding rods. In order to collect sufficient sample size, the total grinding time was 6 minutes which is substantially longer than the reported typical grinding time of 30-90 seconds. Air samples were collected to the sides during one grinding and above and to the side during another. The range for the 4 air samples was 1.3 - 300 mBq/m³ ($3.6 \times 10^{-8} - 8.2 \times 10^{-6} \,\mu$ Ci/m³) with a geometric mean of 23 mBq/m³ ($6.3 \times 10^{-7} \,\mu$ Ci/m³). Breathing zone air samples collected showed no detectable activity at a minimum detectable level of 15 mBq/m³ ($4 \times 10^{-7} \,\mu$ Ci/m³). The data presented by Crim and Bradley (1995) did not specify if the represented total thorium or ²³²Th.

3.6.3.1.4 Study by Jankovic et al.

The study by Jankovic et al.(1999) provided results of air sampling conducted during 15 different welding operations with 2% thorium; four (4) with dilution ventilation and air samplers outside the welding helmet, two (2) with dilution ventilation and air samplers inside the welding helmet, four (4) with local exhaust and air samplers outside the helmet, and five (5) with local exhaust and air samplers were all carried out with direct current.

The data for welding with only dilution air yielded the largest values for airborne ²³²Th with the geometric mean of 0.7 mBq/m³ (2×10⁻⁸ μ Ci/m³) for the sample outside the helmet and 0.15 mBq/m³ (4×10⁻⁹ μ Ci/m³) for the sample inside the helmet. With local exhaust, the values were 0.04 mBq/m³ (1×10⁻⁹ μ Ci/m³) for the sample outside the helmet and 0.1 mBq/m³ (3×10⁻⁹ μ Ci/m³) for the sample inside the helmet. The geometric mean for all sample data was 0.15 mBq/m³ (4×10⁻⁹ μ Ci/m³) unadjusted for respirable fraction (10 μ m or less). Jankovic reported an activity median aerodynamic diameter (AMAD) of welding aerosols of 3.5 μ m at 30 cm from the point of operation. Measurements during grinding with no local exhaust ranged from 14 to 190 mBq/m³ (3.8×10⁻⁷ to 5.1×10⁻⁶ μ Ci/m³) with a geometric mean of 52 mBq/m³ (1.4×10⁻⁶ μ Ci/m³), unadjusted for a measured 60% respirable fraction. The respirable fraction for electrode sharpening was 60% (10 μ m or less) and the AMAD was 5 μ m in the breathing zone. This fraction is somewhat higher than that reported by Vinzents but could easily be a function of sampling method (i.e., effects of gravitational settling relative to sampler location).

3.6.3.2 Assessment by McDowell-Boyer

3.6.3.2.1 Distribution and Transport

Doses from distribution and transport of thoriated-tungsten welding rods were estimated by McDowell-Boyer (NUREG/CR–1039), based on a methodology essentially the same as that presented in Appendix A.3 of this report. Distribution and transport were assumed to involve handling by parcel-delivery workers, over-the-road trucking workers, and retail store workers; also considered were exposures to customers in retail stores. Estimates of individual and collective doses resulting from an annual distribution of 1 million electrodes obtained by McDowell-Boyer are summarized in Table 3.6.1.

During distribution and transport, external exposure to thorium and its decay products was assumed to be the only exposure pathway of concern. For external exposure, the dose to the whole body in Table 3.6.1 was essentially the same as the effective dose equivalent (EDE). For packaged electrodes, there is no credible scenario for ingestion or inhalation of radionuclides in particulate form. The decay product ²²⁰Rn could be released from the welding rods, and inhalation exposure to radon and its short-lived decay products could occur. However, for the worst-case assumption that all radon produced in the electrodes would be released into the air, the estimated annual individual dose equivalents to the lungs for all scenarios were less than 1×10^{-10} millisievert (mSv) (<1×10⁻⁸ mrem) and, thus, were many orders of magnitude less than the estimated doses from external exposure.

3.6.3.2.2 Routine Use

For routine use of thoriated-tungsten welding rods, McDowell-Boyer (NUREG/CR-1039) considered scenarios describing exposure to heavy-use welders, including welding at home as well as in the workplace, occasional-use welders, and nonwelders in the vicinity of welding rods during use. Exposures during grinding of electrodes by welders also were considered.

Assumptions about release of radioactive material during welding and grinding used by McDowell-Boyer include the following: (1) an average loss rate of thorium during welding with normal rods of 0.8 to 6.0 mg/h, (2) an average loss rate of thorium during welding with contaminated rods (either oxidized or contaminated with the weld metal) 10 times higher than for normal rods, with welding using contaminated rods occurring 10% of the time, (3) a percent of material released during welding that is present in the fumes of 0.2 to 1%, and (4) an average loss of thorium during a single grinding of 8 to 16 mg. In addition, the ventilation rate in welding shops was assumed to be 3420 m³/h, in accordance with an American National Standards Institute standard for a welder's workplace (ANSI Z49.1–1973). The ventilation rate in the work area in a home was assumed to be 230 m³/h for a normal-sized room with an air turnover rate of 1 change per hour. Finally, 25,000 welders were assumed to use thoriated-tungsten electrodes for all welds and 90% of the population assumed to use thoriated-tungsten electrodes for 30% of the welds.

Based on the assumptions described above, the estimates of individual and collective doses to bone from inhalation of thorium and its decay products, obtained by McDowell-Boyer (NUREG/CR-1039), are summarized as follows (bone is the critical organ for inhalation of thorium):

- For 12,500 heavy-use welders who are assumed to be exposed to air concentrations of thorium of 0.048 to 1.8 μ g/m³ for 900 hours while welding in a work shop, 12 to 460 μ g/m³ for 4.2 hours while grinding tips in the work shop, 0.48 to 18 μ g/m³ for 100 hours while welding with a contaminated electrode in the work shop (50 of which are also assumed to be exposed to concentrations of 0.7 to 26 μ g/m³ for 180 hours while welding in a home shop), and 7 to 260 μ g/m³ for 20 hours while welding with a contaminated electrode annual individual dose equivalent to bone would be 0.2 to 10 mSv (20 to 1,000 mrem), the average annual individual dose equivalent to bone would be 0.04 to 1.6 mSv (4 to 160 mrem), and the annual collective dose equivalent to bone would be 0.5 to 20 person-Sv (50 to 2,000 person-rem).
- For 12,500 occasional-use welders who are assumed to be exposed to air concentrations of thorium of 0.048 to 1.8 μ g/m³ for 180 hours while welding in a work shop, 12 to 460 μ g/m³ for 0.83 hour while grinding tips in the shop, and 0.48 to 18 μ g/m³ for 20 hours while welding with a contaminated electrode in the shop, the maximum annual individual dose equivalent to bone would be 0.02 to 0.6 mSv (2 to 60 mrem), the average annual individual dose equivalent to bone would be 0.009 to 0.3 mSv (0.9 to 30 mrem), and the annual collective dose equivalent to bone would be 0.1 to 4 person-Sv (10 to 400 person-rem).
- For 75,000 nonwelders who are assumed to be exposed to air concentrations of thorium of 0.048 to 1.8 μ g/m³ for 540 hours during normal welding and 0.48 to 18 μ g/m³ for

60 hours during welding with contaminated electrodes, the maximum annual individual dose equivalent to bone would be 0.20 to 1.5 mSv (20 to 150 mrem), the average annual individual dose equivalent to bone would be 0.07 to 0.5 mSv (7 to 50 mrem), and the annual collective dose equivalent to bone would be 5 to 40 person-Sv (500 to 4,000 person-rem).

Ranges of estimated doses reflect the assumed ranges in the release of thorium from electrodes during welding or grinding and the fraction of the releases from welding that would be present in the inhaled fumes.

Doses to the whole body from inhalation of thorium and its decay products also were calculated by McDowell-Boyer (NUREG/CR–1039). These doses were about an order of magnitude less than the doses to bone given above. However, this calculation of dose to the whole body from internal exposure represented the total energy absorbed in all body tissues divided by the nominal body mass of 70 kg. The calculation is not a meaningful quantity when highly nonuniform irradiations of the whole body occur, as is the case for inhaled thorium. This calculated dose to the whole body for internally deposited thorium is not related to the EDE and does not provide a meaningful basis for comparison.

McDowell-Boyer (NUREG/CR-1039) also estimated external doses to the whole body for heavy-use welders, occasional-use welders, and nonwelders in the vicinity of welding rods during use. As noted previously, dose to the whole body from external exposure to aged thorium essentially was the same as the EDE. These dose estimates are summarized as follows:

- For 12,500 heavy-use welders who are assumed to be exposed at a distance of 30 cm during welding or grinding, 50 of whom are also assumed to be exposed at a distance of 90 cm during welding at home, the maximum annual individual dose equivalent to the whole body would be 1.5×10⁻³ mSv (0.15 mrem), the average annual individual dose equivalent would be 8×10⁻⁴ mSv (0.08 mrem), and the annual collective dose equivalent would be 0.01 person-Sv (1 person-rem).
- For 12,500 occasional-use welders who are assumed to be exposed at a distance of 30 cm during welding or grinding, the maximum annual individual dose equivalent to the whole body would be 8×10^{-5} mSv (0.008 mrem), the average annual individual dose equivalent would be 5×10^{-5} mSv (0.005 mrem), and the annual collective dose equivalent would be 6×10^{-4} person-Sv (0.06 person-rem).
- For 75,000 nonwelders who are assumed to be exposed at a distance of 3.6 meters while working in a shop, the maximum annual individual dose equivalent to the whole body would be 2×10⁻⁵ mSv (0.002 mrem), the average annual individual dose equivalent would be less than 1×10⁻⁵ mSv (<0.001 mrem), and the annual collective dose equivalent would be 6×10⁻⁴ person-Sv (0.06 person-rem).

The importance of external exposure compared with inhalation exposure during routine use of welding electrodes can be evaluated as follows. For inhalation of ThO_2 , which is assumed to be Class Y material, the ratio of the EDE to the dose equivalent to bone is about 0.06 (EPA-520/1-88-020). If the doses to bone calculated by McDowell-Boyer (NUREG/CR-1039) and summarized above are multiplied by this factor, the resulting EDEs are approximately an

order of magnitude or more greater than the calculated doses to the whole body (i.e., EDEs) from external exposure. Therefore, based on the assessment by McDowell-Boyer, inhalation exposure appears to be considerably more important than external exposure during routine use and the grinding of welding electrodes.

3.6.3.3 Disposal

Individual and collective doses from disposal of thoriated-tungsten welding rods in landfills or by incineration were estimated by McDowell-Boyer (NUREG/CR–1039) and Buckley et al. (NUREG/CR–1775). Different approaches or assumptions were used in the dose calculations for disposal in the two assessments. Therefore, results from both assessments are summarized in the following sections.

3.6.3.3.1 Assessment by McDowell-Boyer

McDowell-Boyer (NUREG/CR–1039) assumed disposal of 1 million welding electrodes per year, with 50% distributed equally among 100 landfills and 50% distributed equally among 100 incinerators. Doses were estimated for individuals residing 1 km from a landfill or incinerator, and for an assumed population within a distance of 72 km. The amount of thorium and its decay products present in the welding rods was assumed to be equal to the initial amount (2% by weight of thorium, with the material aged for 20 years), less the total amount assumed to be lost during welding and grinding. Dose estimates for disposal are summarized as follows.

For disposal in landfills, doses were estimated only for releases of ²²⁰Rn, based on an assumption that all of the radon in the welding rods would be released to the atmosphere following disposal. The estimated annual individual dose equivalent to the lungs from inhalation of radon and its short-lived decay products was 2×10⁻⁴ mSv (0.02 mrem). The estimated annual collective dose equivalent was 0.03 person-Sv (3 person-rem). Estimated doses to bone from ingestion pathways were more than four orders of magnitude lower.

For disposal by incineration, doses were estimated by assuming that all of the thorium and its decay products in the welding rods would be released to the atmosphere. The estimated annual individual dose equivalent to bone from inhalation and ingestion was 0.002 mSv (0.2 mrem). The estimated annual collective dose equivalent was 2.2 person-Sv (220 person-rem). About 85% of the dose from incineration was due to ingestion of contaminated food following deposition of the atmospheric release, with the remainder due to inhalation.

3.6.3.3.2 Assessment by Buckley et al.

Buckley et al. (NUREG/CR–1775) considered doses from waste collection, disposal in landfills, and incineration. An annual disposal of 5.2 million welding rods was assumed, but the estimates of collective dose summarized below have been adjusted to represent an annual disposal of 1 million. In all disposal scenarios, the amount of thorium remaining in the welding rods was assumed to be one-half of the initial content.

For waste collectors exposed to welding rods in municipal refuse, the annual individual dose equivalent to the whole body from external exposure was estimated to be less than

 1×10^{-6} mSv (< 1×10^{-4} mrem), and the annual collective dose equivalent was less than 1×10^{-4} person-Sv (<0.01 person-rem). Doses from inhalation of radionuclides in particulate form or inhalation of radon and its short-lived decay products were not estimated, but were presumed to be insignificant compared with the doses from external exposure.

For disposal in a landfill, collective doses were estimated for a nearby population resulting from releases of thorium to groundwater. Based on the "worst-case" assumption that the release rate of thorium from the landfill would be 1% per year for previously incinerated sources or 0.1% per year for intact sources, the collective dose equivalent to bone from disposal of 1 million welding electrodes was estimated to be 400 person-Sv (40,000 person-rem) over a 1000 year period. Individual doses were not estimated for this scenario.

For disposal by incineration, individual and collective doses in a nearby population were estimated by assuming that one-half of the material incinerated would be released to the atmosphere. The estimated annual individual dose equivalent to bone from incineration of 1 million welding rods was 2×10^{-4} mSv (0.02 mrem), and the annual collective dose equivalent was 2 person-Sv (200 person-rem). Doses from inhalation of radon and its short-lived decay products were several orders of magnitude lower. The individual dose estimate is about an order of magnitude greater than the previous estimate by McDowell-Boyer (NUREG/CR-1039) for essentially the same scenario, but the collective doses estimated in the two assessments are nearly the same.

3.6.3.4 Accidents or Misuse

Individual doses resulting from accidents or misuse involving thoriated-tungsten welding rods were considered by McDowell-Boyer and Buckley et al. Assumed exposure scenarios and resulting dose estimates are summarized in the following sections.

3.6.3.4.1 Assessment by McDowell-Boyer

McDowell-Boyer (NUREG/CR–1039) considered an accident scenario involving a warehouse fire and a scenario for misuse involving carrying extra welding electrodes in a shirt pocket. The scenario for a fire assumed that a warehouse would contain 5000 electrodes, which is the same number that was assumed to be incinerated at any site (see Section 3.6.3.3.1). Assuming complete volatilization of the thorium and its decay products in a fire, the inhalation dose to a member of the public near the burning warehouse would be about the same as the dose to an individual near an incinerator. Thus, the estimated dose equivalent to bone for an individual near a warehouse fire was 0.002 mSv (0.2 mrem). Doses to persons in the warehouse also were discussed but were believed to be insignificant because of the difficulty in volatilizing thorium at temperatures obtained in a small fire and the likelihood of rapid evacuation of the warehouse after a fire starts. Doses to firefighters during the fire were not considered.

For the scenario for misuse, an individual was assumed to carry three welding rods in a shirt pocket for 8 h/day and for 250 days (1 working year). The estimated dose equivalent to the maximally exposed portion of the skin for this scenario was 0.04 mSv (4 mrem).

3.6.3.4.2 Assessment by Buckley et al.

Buckley et al. (NUREG/CR–1775) considered a fire in a warehouse containing 50,000 welding electrodes. Doses to firefighters were estimated based on the conservative assumptions of exposure for 8 hours and no use of respiratory protection. The estimated dose equivalent to bone during the warehouse fire was 1.3 Sv (130 rem).

3.6.4 Present Exemption Analysis

This section presents estimates of doses to members of the public from distribution and transport, routine use, and disposal of welding electrodes containing thorium. Estimates of dose also are presented for accident or misuse scenarios involving a fire in a storage warehouse or the inadvertent carrying of welding electrodes in a shirt pocket.

Based on the current practices discussed in Section 3.6.2, each thoriated-tungsten welding rod is assumed to contain 2% by weight of thorium, unless otherwise stated. Furthermore, based on a survey of welders at Oak Ridge National Laboratory (ORNL) performed for this assessment (Jankovic et al.), a typical welding rod is assumed to be 18 cm in length and 0.24 cm in diameter. Thus, for a density of tungsten of 19.35 g/cm³, each welding rod is assumed to contain 0.3 g of thorium. The assumed thorium content per electrode is 30% greater than the value assumed previously by McDowell-Boyer (NUREG/CR–1039), due to the greater assumed length of an average electrode. Except as noted in the dose assessment for disposal, contributions to dose from thorium decay products are taken into account in all exposure scenarios by assuming that the thorium has been aged for 20 years after chemical separation.

The number of thoriated-tungsten welding rods distributed annually is not known, but is assumed in this assessment to be 5 million. The assumed value could be too low, based on information that suggests the annual distribution may be about 10 million (Phone call, G. Goodwin, ORNL, Oak Ridge, TN, June 1997). However, the higher value represents foreign as well as domestic sales, and the fraction sold domestically was not provided. The individual doses for distribution and transport and disposal and the collective doses for all scenarios estimated in this assessment can be adjusted for the actual distribution if it is known.

3.6.4.1 Distribution and Transport

As described in Section 3.6.3.1, the doses from distribution and transport of thoriated-tungsten welding rods were estimated previously by McDowell-Boyer (NUREG/CR–1039) using a methodology that is essentially the same as that presented in Appendix A.3 of this report. Thus, the previous results are considered suitable for the present assessment. The scenario for distribution and transport of welding rods assumed by McDowell-Boyer is described as follows. Of the total number of welding rods distributed, 80% are assumed to be shipped by truck to United Parcel Service (UPS) terminals over distances greater than 400 km, 10% to UPS terminals over distances between 32 and 400 km, and 10% directly to warehouses and retail stores over distances less than 32 km. Of the welding rods shipped to UPS terminals, 80% then are assumed to be shipped to warehouses and 20% to retail stores. Thus, for the assumed total distribution of 5 million welding rods per year, each of 80 warehouses is

assumed to receive twenty-five shipments of 2,000, or a total of 50,000 per year, and each of 200 retail stores is assumed to receive ten shipments of 500, or a total of 5,000 per year.

During distribution and transport, only external exposures are assumed to occur. There are no credible scenarios by which radionuclides in particulate form in packaged electrodes could be inhaled or ingested. Releases of ²²⁰Rn could occur, resulting in inhalation exposure to the radon and its short-lived decay products. However, an assessment by McDowell-Boyer (NUREG/CR–1039), based on the conservative assumption that all of the radon would be released from the packaged electrodes, indicated that the dose from exposure to radon would be many orders of magnitude less than the dose from external exposure (see Section 3.6.3.1). Therefore, doses from inhalation or ingestion during distribution and transport can be neglected.

The dose estimates for external exposure during distribution and transport that are adopted in the present assessment are obtained from the results of McDowell-Boyer, given in Table 3.6.1, by increasing the doses by 30% to reflect the higher thorium content per welding rod assumed in this assessment. Collective doses are also increased by a factor of 5 to reflect current distribution. In estimating individual dose, the maximum values given in Table 3.6.1 are used. As noted previously, the dose to the whole body from exposure to thorium and its decay products is essentially identical to the EDE. Results are summarized as follows:

- The annual EDE to individual truck drivers, UPS terminal employees, retail employees, and warehouse employees would be 8×10⁻⁴ to 0.003 mSv (0.08 to 0.3 mrem), with the dose to truck drivers being the highest. The dose to individual customers in retail stores would be considerably less.
- The collective EDE for 5 million welding rods distributed per year, including the dose to truck drivers, UPS terminal employees, retail store employees, warehouse employees, and customers in retail stores, would be 0.05 person-Sv (5 person-rem).

3.6.4.2 Routine Use

Doses from routine use of thoriated-tungsten welding rods would be experienced primarily during welding and grinding (sharpening) of the electrodes. This assessment considers individual and collective doses during these activities.

3.6.4.2.1 Welding in Shops

Professional welders working in a shop should receive the highest individual doses during welding. In this assessment, inhalation doses for welders are estimated based primarily on the results of a study of air concentrations by Ludwig et al. (1999). In addition, data is used from a study by Jankovic et al. (1999) which included measurements of airborne concentrations of ²³²Th around and inside the welders' masks, measurements of the particle size distribution of the total aerosol, and modeling of airborne concentrations of ²³²Th. Doses from external exposure during welding also are estimated, based on the results of a previous assessment by McDowell-Boyer (NUREG/CR–1039) discussed in Section 3.6.3.2.

These studies indicate that the dose to a welder from inhalation exposure depends on several factors, including (1) the release rate of thorium and its decay products from the electrode

during welding, (2) the fraction of the working time spent welding, (3) the effect of the welder's face mask on concentrations of airborne particulates in the breathing zone, (4) diffusion and sedimentation of airborne particulates, (5) the ventilation rate in the welding area, (6) the presence or absence of local exhaust ventilation at the welding site, (7) the distribution of particle sizes in the inhaled aerosols, and (8) the weight % of thorium in the welding electrode.

Based on the concentration measurements from Ludwig et al. (1999) for DC welding using 2% and 4% by weight ThO₂, the geometric mean air concentration of ²³²Th in the breathing zone area of a welder, without the benefit of local exhaust ventilation, would be about 0.34 mBq/m³ ($9.2 \times 10^{-9} \ \mu \text{Ci/m}^3$). This value is essentially the same as that measured by Jankovic, where the geometric mean for similar measurements was $2 \times 10^{-8} \ \mu \text{Ci/m}^3$ uncorrected for a 45% respirable fraction. As measured by Ludwig et al. (1999), AC welding yielded results that were about a factor of 30 higher than DC welding. For AC welding, the geometric mean airborne level without local exhaust was 10 mBq/m³ ($2.8 \times 10^{-7} \ \mu \text{Ci/m}^3$).

Because of the differences in the reported data, an AMAD particle size of 1 μ m was selected for this dose assessment. This value is considered reasonable because there is only about a 35% change in the inhalation coefficient from an AMAD of 0.2 μ m to 1 μ m or from 1 μ m to 3.5 μ m.

Based on the measurements and assumptions described above, the inhalation dose to a welder who is assumed to be exposed while welding for 4 hours/day and 250 days/yr, for a total exposure time of 1000 h/yr, is estimated as follows. For thorium that has been aged for 20 years following chemical separation, the activity of ²²⁸Th is 87% of the activity of ²³²Th and no other decay products of ²³²Th contribute significantly to the inhalation dose (see tables 3.1.5 and 3.1.7). For inhalation of thorium in oxide form, which is Class Y material, and for an AMAD of 1 μ m, the dose coefficients are 3.1×10^{-4} Sv/Bq (1.15×10^{3} rem/ μ Ci) for ²³²Th and 9.23×10⁻⁵ Sv/Bq (3.42×10^{2} rem/ μ Ci) for ²²⁸Th (EPA–520/1–88–020). For a breathing rate appropriate for light activity of 1.2 m³/h (see Appendix A.1), the resulting EDE from inhalation exposure would be 0.2 mSv (20 mrem). For AC welding, the annual dose is calculated to be 5 mSv (500 mrem), using a geometric mean airborne level of 10 mBq/m³ ($2.8 \times 10^{-7} \mu$ Ci/m³) as measured by Ludwig et al. (1999).

The use of local exhaust ventilation at the welding site, in addition to normal room ventilation, is potentially important. The study by Ludwig et al. (1999) and Jankovic et al. (1999) suggests that local exhaust ventilation, which is used in some welding shops, would reduce average inhalation exposure to welders. Based on the summary of measurements presented in Section 3.6.3.1, the average welder breathing zone concentration is estimated to be a factor of 10 less if local exhaust is utilized, which would yield a calculated dose of 10 lower than that estimated above.

Estimating inhalation dose obtained above is subject to considerable uncertainty attributable in part to limited and often inconsistent data. Also, there is a great deal of variation in the concentrations of thorium that occur during welding under a variety of actual conditions. In the study by Ludwig et al. (1999), for data evaluating 26 welding shops, the concentrations varied by a factor of about 900 from low to high and no data were presented regarding ventilation flow rates.

The type and amperage of the current during welding are also sources of variability in airborne exposures during welding. Increasing the welding arc current from 100 to 250 amperes (A) causes the electrode to be consumed faster and significantly increases the thorium airborne concentrations (according to Jankovic et al., a factor of seven higher at 250A). The study by Ludwig et al. (1999) did not address arc current but the data indicates that alternating current (AC) welders can be subjected to airborne concentrations that are about a factor of 30 above those made with direct current (DC) welding.

Another potential uncertainty in the actual dose is whether significant activity of ²³⁰Th is in the welding rods due to the presence of uranium in the ores from which the thorium was obtained. The presence of ²³⁰Th would increase the dose estimate obtained in the present assessment, because the estimate is based on measurements of airborne concentrations of ²³²Th. However, in a set of measurements on electrodes from three different manufacturers, Jankovic et al. (1999) concluded that the mass abundance of ²³⁰Th in the welding rods used in their studies was less than 1 ppm. Therefore, based on the half-lives given in Tables 3.1.2 and 3.1.3, the activity of ²³⁰Th would be less than 20% of the activity of ²³²Th. Based on the relative activities of the two isotopes and the dose coefficients for inhalation given in Tables 3.1.6 and 3.1.7, the dose from inhalation of ²³⁰Th then would be less than 4% of the dose from inhalation of ²³²Th and its decay products. This result is not particularly sensitive to the clearance class for the inhaled aerosols (EPA–520/1–88–020). Therefore, if the welding rods used in the studies by Jankovic et al. (1999) are representative of electrodes in use at the present time, the dose contribution from ²³⁰Th would be insignificant.

Welders also would receive an external exposure during welding. Based on the assumed welding time of 4 h/day and an assumed average distance from the electrode of 30 cm, the dose to the whole body estimated by McDowell-Boyer (NUREG/CR–1039), when increased by 30% to take into account the higher amount of thorium per welding rod assumed in this assessment, would be 0.002 mSv/yr (0.2 mrem/yr). Thus, as expected, external doses to welders should be insignificant compared with inhalation doses.

3.6.4.2.2 Welding in Homes

In this assessment, doses to an occasional home welder are estimated based on (1) the dose estimate for a welder in a shop given above and, (2) assumptions about the exposure time for a home welder. The home welder is assumed to be exposed during a single 4-hour work session per week, for a total of 200 h/yr (NUREG/CR–1039). The assumed exposure time in a home is a factor of 5 less than the assumed exposure time in a shop. Therefore, if the AMAD is assumed to be the same in the two work environments, and without the benefit of a local exhaust system, the EDE to a home welder would be about a factor of 5 lower than the dose to a welder in a shop obtained previously, or about 1 mSv (100 mrem) for AC welding and 0.03 mSv (3 mrem) for DC welding. Use of local exhaust is estimated to reduce the doses by a factor of 10.

3.6.4.2.3 Grinding

Inhalation exposures also can occur during grinding (sharpening) of welding rods. Two types of individuals may grind electrodes: (1) dedicated grinders who spend most of their working time at this activity in support of many welders, and (2) welders who typically grind electrodes only a few times per day for their own use. The latter situation presumably is far more common.

However, Sinclair and Thind (1992) observed two workers at a nuclear construction site who spent most of their time grinding as many as 250 electrodes per day. Doses for both groups of workers are considered in this assessment.

Doses to welders who grind their own electrodes are estimated as follows. Ludwig et al. (1999) reported an activity concentration for ²³²Th during the grinding of about 180 mBq/m³ ($5\times10^{-6} \ \mu \text{Ci/m}^3$) with the airborne activity concentration decreasing soon after the end of grinding. Crim and Bradley reported an average airborne concentration during grinding of 23 mBq/m³ ($6.3\times10^{-7} \ \mu \text{Ci/m}^3$) and Jankovic et al. reported a value of 52 mBq/m³ ($1.4\times10^{-6} \ \mu \text{Ci/m}^3$). Vinzents reported value of 440 mBq/m³ ($1.2\times10^{-5} \ \mu \text{Ci/m}^3$) is considered atypical due to the method utilized for the measurement. Based on these data, an average airborne activity for grinding of 85 mBq/m³ ($2.3\times10^{-6} \ \mu \text{Ci/m}^3$) is determined.

Ludwig et al. (1999) reported that studies performed by others indicated that the AMAD of grinding particles is about 1 μ m. Jankovic reported a value of 5 μ m. The concentration and measured particle size would depend upon many factors such as: the presence of any local exhaust ventilation, the grinding wheel or belt and its direction and speed, the presence of any particle guards, the application pressure, and the required grinding time. An AMAD size of 1 μ m was selected for this dose assessment. This value is considered reasonable as there is an increase in the inhalation factor by only about 1.6 from an AMAD range of 10 to 0.2 μ m.

The annual EDE for these assumptions for a welder performing his own sharpening would be 0.2 mSv (20 mrem). Use of local ventilation is estimated to reduce this dose by about a factor of 10. As in the case of welding, uncertainty in the assessments of estimated dose during grinding is due to variables in the airborne concentration of thorium.

Occasional home welders also would grind welding electrodes for their own use. In the assessment of exposures in a home, the inhalation dose for welders was estimated to be a factor of 5 times lower due to the reduced work time. This ratio also should apply to grinding in a home shop. Therefore, the estimated EDE to a home welder from grinding electrodes is 0.03 mSv (3 mrem) to the home welder.

Doses to dedicated grinders are estimated based on the average airborne activity described above during grinding by normal welders without benefit of local exhaust ventilation. The sharpening efficiency of a dedicated grinder is estimated to be 20 seconds per sharpening. This is about one third of the time typically reported for a welder sharpening his own welding rod. For a hypothetical large welding operation consisting of 50 welders, about 150 welding rods would have to be sharpened per work day. For one person performing the sharpening this amounts to 200 hours per year. Assuming the same AMAD, inhalation dose coefficient, and breathing rate as in the analysis for grinding by welders given above, the estimated EDE to dedicated grinders would be about 8 mSv (800 mrem). All of the uncertainties discussed above apply to this scenario. Parameters were developed to be on the conservative side, with certain of them set at the maximum expected, e.g., no local exhaust ventilation. This scenario was developed to review the significance of this exposure pathway.

The average air concentration of thorium over an 8-hour work day for the dedicated grinders, with local exhaust ventilation, estimated by Sinclair and Thind (1992), is 1.4 μ g/m³, which corresponds to an activity concentration of ²³²Th of 5.6 mBq/m³ (1.5×10⁻⁷ μ Ci/m³). Assuming the same AMAD, inhalation dose coefficient, and breathing rate as in the analysis for grinding

by welders given above, and assuming exposure for 200 working hours per year, the estimated EDE to dedicated grinders would be 0.5 mSv (50 mrem). This estimate is more than an order of magnitude less than the dose for a dedicated grinder in a welding shop given above. This reduction presumably is due primarily to the use of local exhaust ventilation in this assessment, but not by those in the hypothetical operation described above.

3.6.4.2.4 Combined Doses from Welding and Grinding and Collective Doses

The total dose estimate for a professional welder (without benefit of local exhaust) would be the sum of the contributions from welding and from grinding and is 5 mSv (500 mrem) for AC welding and 0.3 mSv (30 mrem) for DC welding. The collective dose to professional welders for an assumed equal use of AC and DC welding and use of 4 million electrodes per year (reserving 1 million of the 5 million annual production for home welders and automatic welders) can be estimated as follows. Using the ratios of welders to the number of electrodes established by McDowell-Boyer (NUREG/CR–1039), there would be 100,000 welders for 4 million electrodes (allowing a portion of the annual production to be used for home or automated welding). The collective EDE would be 100,000 times the average of the AC and DC welding doses or 300 person-Sv (3×10^4 person-rem).

Following the model developed earlier, the total dose estimate for the home welder (without benefit of local exhaust) would also be the sum of the contributions from welding and from grinding and is 1 mSv (100 mrem) for AC welding and 0.06 mSv (6 mrem) for DC welding. The collective EDE for the home welder would be 10,000 times the average of the AC and DC welding doses or 5 person-Sv (500 person-rem).

3.6.4.3 Disposal

Following their useful lifetime, unused portions of welding rods and grinding fines from floors may be sent to landfills or incinerators for disposal. Jankovic et al. (1999) concluded that only 3% of a rod is actually consumed in welding. Therefore, this dose assessment assumes that disposal is made of 5 million pieces of left-over welding rods and floor fines. The quantity disposed could be 97% of the annual distribution or about 1.5×10^3 kg of thorium. Recycling of used electrodes is considered unlikely.

In this assessment, the generic methodology in Appendix A.2 is used to estimate individual and collective doses from disposal of welding rods containing thorium. Dose estimates obtained previously by McDowell-Boyer (NUREG/CR–1039) and Buckley et al. (NUREG/CR–1775), and described in Section 3.6.3.3, are not used. For disposal in landfills, McDowell-Boyer estimated doses to off-site individuals and populations due only to releases of ²²⁰Rn, based on the conservative assumption that all radon produced in the electrodes would be released to the atmosphere, and Buckley et al. estimated collective doses from groundwater transport using very conservative assumptions about thorium release rates. For incineration, McDowell-Boyer and Buckley et al. based their estimates of individual and collective doses on the very conservative assumption that 50 to 100% of the thorium in the electrodes would be released to the atmosphere.

The survey of welders at ORNL performed for this assessment indicated that a typical electrode with an initial length of about 18 cm is about 4 cm in length at the end of its useful life. All decay products of ²³²Th are assumed to be present in activity equilibrium.

3.6.4.3.1 Disposal in Landfills

Of the 1.5×10^3 kg of thorium assumed to be disposed per year, 80% are assumed to be sent to landfills (see Appendix A.2). In the generic methodology described in Appendix A.2, doses are estimated for waste collectors, workers at landfills, off-site individuals who reside near landfills during operations or after closure, and individuals who might reside on the landfill sites after closure. In estimating doses, no allowance is made for the non-dispersibility of the solid pieces of welding rods which amounted to 20% of the total quantity disposed.

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

- The EDE to individual waste collectors would be 0.004 mSv (0.4 mrem). The annual EDE to a future on-site resident would be about 0.004 mSv (0.4 mrem), and the annual EDE to individual workers at landfills would be about 5×10⁻⁴ mSv (0.05 mrem). The dose to off-site individuals after closure of the landfills would be many orders of magnitude less.
- The collective EDE from 1 year's disposals of welding rods would be 4 person-Sv (400 person-rem), due almost entirely to exposure to future on-site residents for 1000 years after facility closure. If exposure to future on-site residents are not taken into account, the collective EDE from exposure to waste collectors and workers at landfills would be 0.02 person-Sv (2 person-rem) per year.

3.6.4.3.2 Disposal in Incinerators

Of the 1.5×10^3 kg of thorium assumed to be disposed per year, 20% are assumed to be sent to incinerators (see Appendix A.2). In the generic methodology for disposal in incinerators described in Appendix A.2, doses are estimated for waste collectors, workers at incinerators, and off-site individuals who reside near the incinerators during operations. In estimating doses, no allowance is made for the non-dispersibility of the solid pieces of welding rods which amounted to 20% of the total quantity disposed.

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

- The EDE to individual waste collectors would be 0.02 mSv (2 mrem). The annual EDE to individual workers at incinerators would be about three orders of magnitude less. The dose to individual off-site residents near incinerators would be about six orders of magnitude less.
- The collective EDE from 1 year's disposals would be 0.3 person-Sv (0.03 person-rem), due almost entirely to exposure to waste collectors.
- The estimated dose for an off-site member of the public was estimated to be less than 1×10⁻⁵ mSv (<0.001 mrem) with an off-site collective EDE of 7×10⁻⁵ person-Sv (0.007 person-rem).

3.6.4.4 Accidents or Misuse

As described in Section 3.6.3.4, two scenarios involving accidents or misuse of welding rods containing thorium were considered in previous assessments by McDowell-Boyer and Buckley et al.: (1) a fire in a storage warehouse, and (2) the carrying of extra electrodes in a shirt pocket. Both of these scenarios are considered in the present assessment.

3.6.4.4.1 Warehouse Fire

Individual doses resulting from a fire in a warehouse containing welding rods are estimated using the generic methodology in Appendix A.1. For this scenario, doses are estimated for firefighters during the fire and for individuals involved in cleanup activities after the fire.

In this assessment, the number of welding rods stored in a warehouse is assumed to be 100,000, or 2% of the assumed distribution of 5 million electrodes per year. For an assumed mass of thorium per electrode of 0.3 g, the 100,000 electrodes in a warehouse would contain 30 kg of thorium. Application of the generic methodology for a warehouse fire to this amount of thorium gives the following results:

• For a fire in a warehouse containing 100,000 welding rods, the EDE to a firefighter using respiratory protection during the fire would be 0.001 mSv (0.1 mrem). The EDE to an individual not using respiratory protection during cleanup after the fire would be 0.06 mSv (6 mrem).

The estimated dose to a firefighter given above is much less than the value obtained previously by Buckley et al. (NUREG/CR–1775), as given in Section 3.6.3.4.2. The large reduction in estimated dose, compared with the previous result, is due primarily to the assumptions that (1) 0.01% of the thorium in welding rods would be released in a fire, instead of 100%; (2) the firefighter would use a respirator with a protection factor of 1000, instead of no respiratory protection; and (3) the firefighter would be exposed for 0.5 hour, instead of 8 hours.

3.6.4.4.2 Carrying of Electrodes in a Shirt Pocket

McDowell-Boyer (NUREG/CR–1039) previously considered a scenario for misuse of welding rods involving carrying of extra electrodes in a shirt pocket (see Section 3.6.3.4.1). An individual was assumed to be exposed to three electrodes for 2000 h/yr. Such an exposure could occur, for example, if the electrodes were carried inadvertently in the pocket of a lab coat.

In this assessment, the EDE to an individual is estimated for the same exposure conditions assumed by McDowell-Boyer, as described above. For external exposure to thorium in welding rods carried next to the body, the EDE is assumed to be given by the dose equivalent at a depth of 10 cm in tissue (see Appendix 4). For exposure to three electrodes containing a total of 0.9 g of thorium that has been aged for 20 years, the EDE, as calculated using CONDOS II (Computer Codes, O'Donnell et al., 1981), is 0.08 mSv (8 mrem). This estimate agrees reasonably well with the dose equivalent to the skin of 0.04 mSv (4 mrem) estimated previously by McDowell-Boyer for welding rods containing 30% less thorium.

3.6.5 Summary

This assessment has considered doses to members of the public from distribution, use, and disposal of welding rods containing thorium. The amount of thorium contained in welding rods was assumed to be 2% by weight, in accordance with current practices, and the amount of thorium in a typical electrode was assumed to be 0.3 g. The number of thoriated-tungsten welding rods distributed, used, and disposed per year was assumed to be 5 million. Results of the assessment are presented in Table 3.6.2 and may be summarized as follows:

- The highest individual and collective doses should occur during routine use of welding rods by welders and grinders. Doses during distribution and transport and from disposal in landfills and incinerators should be very small by comparison.
- The estimated collective dose to professional welders is based on an assumption that 5 million welding rods containing thorium are used per year. This estimate would be conservative if fewer than 5 million electrodes were used per year or if a substantial fraction of the electrodes were used on automatic welding machines.
- The estimates of individual and collective doses to welders and grinders during routine use are subject to considerable uncertainty, due primarily to the variability and uncertainty in the concentrations of thorium and its decay products in air during welding and grinding. This uncertainty could be as large as an order of magnitude. In addition, the doses during welding could be substantially different depending upon the arc-current and the type of current used. On the other hand, the doses during welding and grinding could be substantially lower when local exhaust ventilation is used in addition to normal room ventilation.

Doses for particular scenarios involving accidents and misuse also were estimated in this assessment, including doses resulting from a fire in a warehouse and from inadvertent carrying of extra welding rods in a shirt pocket. However, the estimated doses for the scenarios considered are quite low, and the results suggest there are no credible scenarios for accidents or misuse that could result in doses comparable to the doses that would be experienced by welders and grinders during routine use of welding rods. This is a reasonable conclusion when one considers that the scenarios for routine exposure to welders and grinders involve release into the air of most of the thorium contained in welding rods.

_	Annual Individual Dose Equivalent (mrem) ^b		Annual Collective
Population Group	Maximum	Average	Dose Equivalent ^c (person-rem) ^b
Truck drivers	0.2	0.002	0.0052
UPS terminal employees	0.15	<0.001	0.01
Retail employees	0.06	0.06	0.3
Warehouse employees	0.15	0.15	0.062
Customers in retail stores	<0.001	<0.001	0.35
Total			0.72

Table 3.6.1 Individual and Collective Doses From Distribution and Transport of Thoriated-Tungsten Welding Rods Obtained in Previous Assessments ^a

^a Dose estimates for welding rods containing 2% by weight of thorium that has been aged for 20 years after chemical separation and 0.23 g of thorium per welding rod, as obtained by McDowell-Boyer (NUREG/CR–1039), based essentially on methodology presented in Appendix A.3 of this report. Estimates are doses to whole body from external exposure, which are essentially the same as effective dose equivalents. Doses from inhalation or ingestion are negligible by comparison (see Section 3.6.3.2.1).

^b 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

^c Doses assuming annual distribution of 1 million welding rods.

Exposure Scenario	Individual Annual Effective Dose Equivalent (mrem)°	Collective Effective Dose Equivalent ^b (person-rem) ^c
Distribution and transport	0.3 ^{b,d}	5
Routine use (without local exhaust) ^e		
Professional welders AC Current DC Current Grinding	500 20 20	30,000
Home welders AC Current DC Current Grinding Dedicated grinder	100 3 3 800 ^f	500
<u>Disposal</u> Landfills Incinerators	0.4 ^g 2 ⁱ	400 ^h 0.3
Accidents or misuse ⁱ Warehouse fire Carrying electrodes	6 ^k 8 ⁱ	NA ^m

Table 3.6.2 Summary of Potential Radiation Doses From Use of Welding Rods Containing Thorium ^a

^a Dose estimates are based on the assumption that each welding rod contains 0.3 g of thorium prior to use. Dose estimates for distribution and transport, routine use, and accidents or misuse assume that thorium has been aged for 20 years after chemical separation. Dose estimates for disposal assume that all decay products of ²³²Th are in activity equilibrium.

^b Dose estimates are based on assumption that 5 million welding rods are distributed, used, and disposed per year.

° 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 UPS terminal employees, retail employees, or warehouse employees, and considerably less for customers in retail stores (see Table 3.6.1).

^e Dose estimate for inhalation dose during welding and during grinding of welding rods. Use of local exhaust is estimated to reduce doses by a factor of 10.

[†] Dose estimate applies to individuals who are assumed to spend most of their working time grinding electrodes, but this activity apparently is not a common occurrence.

⁹ Dose estimate applies to waste collectors and future on-site residents at landfills; doses are considerably less for workers at landfills and future on-site residents (see Section 3.6.4.3.1).

^h Dose estimate applies to future on-site residents at landfills for 1000 years after disposal due to 1 year's disposals; if dose to future on-site residents is not taken into account, estimated annual collective dose to waste collectors and workers at landfills is 0.02 person-Sv (2 person-rem).

¹Dose estimate applies to waste collectors at incinerators; dose estimates are considerably less for workers at incinerators and off-site members of the public (see Section 3.6.4.3.2).

Dose estimates apply to single occurrence of accident or misuse scenario.

^k Dose estimate applies during cleanup after fire in warehouse containing 100,000 welding rods; dose estimate for firefighter using respiratory protection is considerably less.

¹ Dose estimate assumes that individual carries three electrodes in shirt pocket for 2,000 hours. ^m Not applicable.

3.7 Thorium in Electric Lamps for Illuminating Purposes

3.7.1 Introduction

In 10 CFR 40.13(c)(1)(iv), persons who receive, possess, use, or transfer electric lamps containing thorium for illuminating purposes are exempted from licensing requirements for source material, provided each lamp does not contain more than 50 mg of thorium. The maximum amount of thorium (50 mg) allowed under this exemption corresponds to an activity of 0.2 kilobecquerel (kBq) (5.5 nanocurie (nCi)). This exemption was proposed on August 17, 1965 (30 FR 10203), and issued as a final rule on December 22, 1965 (30 FR 15802).

Quantitative and qualitative information on radiation doses to the public resulting from the use and disposal of electric lamps containing thorium in amounts allowed under this exemption, including assessments for accident scenarios, are available from two sources. First, the proposed rule cited above contains information on doses from routine use and disposal of electric lamps, and potential doses from accidental breakage of lamps are also discussed. Second, Buckley et al. (NUREG/CR–1775) performed a dose assessment for fluorescent lamp starters containing thorium that considered routine exposure to workers during distribution, routine exposures during use of lamps, exposures from disposal in landfills or by incineration, and accidental exposure to firefighters during a warehouse fire. Results of these assessments are discussed in Section 3.7.3.

3.7.2 Description of Items

A variety of electric lamps used for illuminating purposes may contain thorium in amounts allowed under this exemption. In the past, thoriated-tungsten filaments were used extensively in incandescent lamps intended for general lighting purposes. However, most such lamps now use rhenium-tungsten filaments. Thorium has also been widely used in fluorescent lamp starters to produce ionization within the starter (NUREG/CR–1775) and in certain specialty illuminating lamps, such as those requiring high electrode emissivities or hot strength, and lamps that emit intense light or light with specific spectra. For example, photoflash lamps, which are attached to cameras by the user, and lamps used in vehicles (e.g., directional-signal lamps in automobiles and train lights) have been identified as important types of specialty lamps containing thorium. However, detached photoflash lamps for cameras are no longer are used.

Thorium used in electric lamps is in an insoluble oxide form. The thorium normally is enclosed by an airtight glass envelope that is penetrated by the necessary electrical connections, and additional containment materials (e.g., an aluminum housing) may be included. These envelopes ensure that, during normal use, radioactive material (including ²²⁰Rn) would not be released from the lamps. Therefore, unless the lamps are broken, only external exposure to thorium and its decay products would be of concern.

3.7.3 Summary of Previous Analyses and Assessments

This section summarizes the results of previous analyses and assessments of doses to members of the public from use of electric lamps—for illuminating purposes—that contain

amounts of thorium allowed under this exemption. As indicated in Section 3.7.1, quantitative and qualitative estimates of dose are provided in the proposed rule for this exemption and in the report by Buckley et al. (NUREG/CR–1775).

3.7.3.1 Assessments by Atomic Energy Commission

The proposed rule for this exemption cited in Section 3.7.1 discusses dose assessments performed by the Atomic Energy Commission (AEC) (30 FR 10203). Exposure scenarios considered include external dose during routine use of lamps, exposures following routine disposal of broken or burned-out lamps, reclamation and recycling of the thorium for use in other products, and ingestion or inhalation of thorium following accidental breakage of a lamp. Exposures during distribution and transport of lamps were not considered. The assessments performed by the AEC are summarized as follows:

- Routine Use. The dose equivalent from external exposure for an individual using a lamp should not exceed a few tenths of a microsievert (µSv)/yr (few hundredths of a mrem/yr). This result was based on an estimated external exposure rate of 1.5 nanocoulomb (nC)/kg-h (6 microroentgen (µR)/h) at a distance of 1 meter from a lamp containing 50 mg of thorium that had been aged for 20 years after chemical separation. Inhalation and ingestion exposures would not be of concern.
- Disposal and Recycling. Routine disposal of broken or burned-out lamps in normal refuse disposal facilities is highly unlikely to result in any significant radiological impact, and reclamation and recycling of thorium in other products was considered very unlikely.
- Accidental Breakage. Ingestion or inhalation exposures could result from accidental breakage of lamps containing thorium. However, ingestion of significant quantities of thorium should be highly unlikely, due to the care that normally would be taken in cleaning up and discarding broken glass. In addition, experiments in which lamps were broken in small, unventilated rooms indicated it is very unlikely that the resulting concentration of thorium in air would exceed the maximum permissible concentration (MPC) for members of the public that had been established in Appendix B, Table II, of 10 CFR 20 (prior to the 1991 revision of 10 CFR20). The MPC in air was based on an annual committed dose equivalent to the lungs of 15 mSv (1.5 rem) from continuous exposure.

3.7.3.2 Assessments by Buckley et al.

Buckley et al. (NUREG/CR–1775) performed an assessment of doses to the public from use of fluorescent lamp starters containing thorium. Based on available product information, each starter was assumed to contain 45 μ g of thorium, but the results described below, have been adjusted to represent the maximum thorium content of 50 mg per lamp allowed under this exemption. Buckley et al. also assumed that 5 million lamp starters were distributed and disposed of annually and that 50 million starters were in use at any time.

The scenarios for routine exposure to fluorescent lamp starters considered by Buckley et al. include exposure to workers during distribution, exposures during use of the lamps, and exposures from disposal of lamps in landfills or by incineration. In addition, an accident

scenario involving exposure to firefighters during a warehouse fire was considered. Results of the assessments are summarized as follows:

- *Distribution.* External doses to individual workers during distribution of lamp starters were assumed to be very low compared with doses to individual users. Internal doses to workers during distribution were presumed to be essentially zero (0), because no radioactive material should be released from unbroken lamps.
- Routine Use. Based on assumptions that a user of fluorescent lamps would spend an average of 2 h/day at a distance of 1 meter from two lamps, and that the thorium had been chemically separated from its decay products 20 years previously, the dose equivalent to whole body from external exposure would be 7×10⁻⁵ mSv/yr (0.007 mrem/yr). For an assumed population of users of 70 million, the collective dose equivalent then would be 5 person-Sv/yr (500 person-rem/yr).
- Incineration. Based on an assumption that 5 million lamp starters would be incinerated per year, the committed dose equivalents to maximally exposed individuals would be 7×10⁻⁵ mSv/yr (0.007 mrem/yr) to bone and 4×10⁻⁵ mSv/yr (0.004 mrem/yr) to the lungs. The collective dose equivalents would be 0.7 person-Sv/yr (70 person-rem/yr) to bone and 0.4 person-Sv/yr (40 person-rem/yr) to the lungs.
- Disposal in Landfills. Based on assumptions that 5 million fluorescent lamp starters per year would be sent to landfills, and that half of the thorium in the starters would be transported in groundwater to assumed receptor locations, the collective dose equivalents to bone and the lungs from disposal in landfills would each be 30 person-Sv/yr (3,000 person-rem/yr). Doses to off-site individuals, workers at the landfill, or future residents at the disposal site were not estimated.
- Accidental Fires. Based on assumptions that 500,000 lamp starters would be stored in a warehouse, all of the thorium would be released during the fire, and the exposure time for firefighters would be 8 hours without respiratory protection, the committed dose equivalent to bone would be 5.2 Sv (520 rem).

Several of the assumptions used by Buckley et al. should be very conservative, including the assumption for landfill disposal that half of the thorium in the starters would be transported in groundwater to off-site locations, and the assumptions for the accident scenario that all of the thorium would be released during a fire and that a firefighter would be exposed for 8 hours without respiratory protection. In addition, as noted previously, the average amount of thorium in lamp starters is only about 0.1% of the maximum amount allowed under this exemption, and the dose estimates based on the amount of thorium actually used would be reduced accordingly.

3.7.4 Present Exemption Analysis

...

This section presents estimates of doses to the public from routine distribution and transport, use, and disposal of electric lamps containing thorium in amounts allowed by this exemption. Estimates of dose for accident scenarios involving breakage of lamps and a fire in a storage warehouse also are presented.

The types of electric lamps containing thorium considered in this analysis include incandescent lamps, fluorescent lamp starters, and automobile signal lamps. Some of these lamps apparently are not widely used at the present time, including, for example, incandescent lamps with thoriated-tungsten filaments. However, the estimates of individual and collective doses do not depend significantly on the type of electric lamp being considered, particularly in regard to the details of lamp construction. Rather, the estimates of individual and collective doses depend primarily on assumptions about the conditions of exposure, and the dose estimates presented in this section should be reasonably representative of any type of thorium-containing electric lamp that might be widely used at the present time.

The amount of thorium per lamp used in most dose assessment is the same as that assumed by Buckley et al., 45 μ g, while the maximum allowed by the exemption of 50 mg is reviewed for routine use, accident and misuse scenarios. If information were available regarding the thorium content in a product, it is stated. As noted in discussing the previous assessment for fluorescent lamp starters (see Section 3.7.3.2), the amount of thorium actually used in a lamp may, in some cases, be only a small fraction of the maximum allowable amount.

During routine use, as well as distribution and transport, of any type of electric lamp containing thorium, external exposure to photons emitted by the thorium and its decay products is the only exposure pathway of concern. This is because the radioactive material normally is sealed in an airtight glass envelope, and ingestion or inhalation exposures, including exposures to ²²⁰Rn, would not normally occur. Furthermore, the external dose rate from any type of lamp can be estimated by assuming a point source, because the thorium is confined to a small region of the lamp. Finally, for many types of lamps, a reasonable estimate of the dose rate can be obtained by assuming that the source is unshielded, except for the presence of air between the source and receptor locations, because the high-energy photons emitted by some of the decay products of thorium require considerable shielding to reduce the external dose significantly.

3.7.4.1 Distribution and Transport

During the distribution of electric lamps that are assumed to be intact, external doses could be received by distribution workers and by members of the public who purchase the products. During the transport of intact electric lamps, external doses could be received by truck drivers and terminal workers. Members of the public along the truck routes also could receive external exposure, but their doses would be much less than the doses to workers.

Individual and collective doses during distribution and transport of electric lamps containing 45μ g of thorium were estimated using the methodology described in Appendix A.3. The actual distribution of electric lamps containing thorium is not known and probably varies considerably depending on the type of lamp. For example, the annual distribution of fluorescent lamp starters presumably is considerably greater than the annual distribution of automobile signal lamps. In this assessment, collective doses are estimated for 1 million lamps distributed per year, but dose estimates could be adjusted to represent actual distributions if they are known.

The distribution and transportation network for electric lamps containing thorium also is not known, and it probably varies depending on the type of lamp. In this assessment, the distribution and transportation network assumed for welding rods containing thorium, which was developed by McDowell-Boyer (NUREG/CR-1039) and is described in Section 3.6.4.1, is assumed to apply to electric lamps containing thorium. The essential features of the assumed

distribution and transportation network are the following. First, 80% of the lamps are assumed to be shipped by truck to United Parcel Service (UPS) terminals over distances greater than 400 km, 10% to UPS terminals over distances between 32 and 400 km, and 10% directly to warehouses and retail stores over distances less than 32 km. Second, 80% of the lamps shipped to UPS terminals are assumed to be shipped to warehouses and 20% to retail stores. Finally, each warehouse is assumed to receive 1% of the total annual distribution of lamps, and each retail store is assumed to receive 0.1%.

The assumed distribution and transportation network is appropriate for products which, first, are mostly transported by truck over appreciable distances and, second, are not sold in retail stores to a large extent. These assumptions should be reasonable for such specialty products as fluorescent lamp starters and lamps used in vehicles. The assumptions may be less reasonable for incandescent lamps and photoflash lamps containing thorium, which normally would be sold mainly in retail stores, but these types of lamps are not widely distributed at the present time. Therefore, the assumed distribution and transportation network should be reasonable for the types of lamps subject to this exemption that are most commonly distributed at the present time.

Based on the assumptions described above, the individual and collective doses from distribution and transport of 1 million lamps per year can be obtained based on an assumed distribution of 1 million lamps each containing 45 μ g. The following results are as follows:

- The effective dose equivalent (EDE) to individual truck drivers, UPS terminal employees, retail employees, and warehouse employees would be less than 1×10⁻⁵ mSv (<0.001 mrem). The dose to individual customers in retail stores would be substantially less.
- The collective EDE for 1 million lamps distributed per year, including the dose to truck drivers, UPS terminal employees, retail store employees, warehouse employees, and customers in retail stores, would be less than 1×10⁻⁵ person-Sv (<0.001 person-rem).

3.7.4.2 Routine Use

For routine use of electric lamps containing thorium, estimates of individual and collective doses are obtained based on one of two approaches, depending on the type of lamp. In some cases, estimates were based on the external dose rate at a distance of 1 meter from an unshielded point source representing a lamp, using the external dose coefficients given in Table 3.1.7. In other cases, results were obtained using CONDOS II (Computer Codes, O'Donnell et al., 1981), which takes into account the appropriate amount of shielding between the source and receptor locations when the shielding is significant. In all assessments for routine use, the thorium is assumed to have been aged for 20 years after chemical separation. The types of lamps considered in this assessment again include incandescent and fluorescent lamps used for lighting, photoflash lamps used by amateur photographers, and automobile signal lamps. As noted previously, the doses from routine use of these types of lamps should be reasonably representative of the doses from use of any other types of lamps.

3.7.4.2.1 Incandescent and Fluorescent Lamps

For incandescent and fluorescent lamps used in the home or workplace, an individual is assumed to spend 8 h/day at a distance of 1 meter from a single lamp, and 20 h/day at a distance of 2 meters from two lamps. Based on the external dose rate at a distance of 1 meter from a point source containing 45 μ g of thorium of 7.4×10⁻¹¹ mSv/h (7.4×10⁻⁹ mrem/h), as obtained from Table 3.1.7, and taking into account that the dose rate varies inversely as the square of the distance from the source, the EDE to an individual would be less than 1×10⁻⁵ mSv (<0.001 mrem).

In estimating collective dose during routine use, it is unreasonable to assume that the exposure times and distances for an individual given above would apply to an average member of the exposed population. Rather, an average individual is assumed to spend 2 h/day at a distance of 1 meter from a single lamp, and 20 h/day at an average distance of 2 meters from a single lamp. Based on the external dose rate given above at a distance of 1 meter from a point source, the collective EDE for 1 million lamps in use would be 2×10^{-4} person-Sv (0.02 person-rem).

As noted in Section 3.7.2, most incandescent lamps no longer contain thorium. In addition, as noted in Section 3.7.3.2, the average amount of thorium in fluorescent lamps is only about 0.1% of the maximum amount allowed under this exemption. Dose estimates given above should be increased proportionally to represent doses from maximum amounts. At the maximum amount allowed, the EDE to an individual would be about 2×10^{-4} mSv (0.02 mrem).

3.7.4.2.2 Automobile Signal Lamps

Estimates of external dose were obtained for individuals who drive automobiles with signal lamps containing 170 μ g of thorium as estimated by one manufacturer. (Phone call, A. Zielinski, GE Lighting, Cleveland, OH, October 1999). An automobile was assumed to have four such lamps, two mounted near the front and two near the rear. The front lamps were assumed to be located 0.9 meter from the driver and shielded by an equivalent of 1 cm of iron. The rear lamps were assumed to be 2.4 meters from the driver and shielded by an equivalent of 0.75 cm of iron.

For the conditions described above, the EDE rate to the driver of an automobile calculated using CONDOS II (Computer Codes, O'Donnell et al., 1981) is 0.009 nSv/h (0.9 nrem/h). If a driver is assumed to spend an average of 4 h/day in an automobile, a reasonable maximum value for most individuals, the EDE would be 1×10^{-5} mSv (0.001 mrem) in a year. Increasing the amount of thorium to the maximum allowed in all signal lamps and two headlamps, would represent a scenario for the maximum individual EDE as 3×10^{-5} mSv (0.03 mrem).

The number of automobile signal lamps in use containing thorium is not known. The collective dose for 1 million lamps was estimated by assuming that 69% of all automobiles are occupied by four persons for 1 h/day, 30% are occupied by two persons for 2 h/day, and 1% are occupied by one person for 4 h/day. The resulting collective EDE for 1 million lamps in use would be 9×10^{-4} person-Sv (0.09 person-rem).

3.7.4.3 Disposal

Following their useful lifetime, electric lamps may be sent to landfills or incinerators for disposal. As noted in Section 3.7.3.1, the AEC previously concluded that recycling and reuse of thorium in electric lamps is relatively unimportant, and there is no evidence that this conclusion is inappropriate. Thus, recycling and reuse is not considered further in this assessment.

In this section, the generic methodology described in Appendix A.2 is used to obtain estimates of individual and collective doses from disposal of electric lamps containing thorium. The previous dose estimates of Buckley et al. (NUREG/CR-1775) described in Section 3.7.3.2 are not used in this assessment. Particularly for disposal in landfills, Buckley et al. used unreasonably conservative assumptions about release and transport of thorium to assumed receptor locations. Doses to waste collectors, workers at a landfill, and future on-site residents were not considered.

The number of lamps that might be disposed during a year is not known. As in the analysis of collective doses from routine use in Section 3.7.4.2, doses are estimated assuming that 1 million lamps are disposed per year. It is further assumed that each lamp contains 45 μ g of thorium so then, 45 g of thorium is assumed to be disposed per year. Finally, the dose estimates assume that 80% of the lamps are sent to landfills and 20% to incinerators.

3.7.4.3.1 Disposal in Landfills

In the generic methodology described in Appendix A.2, doses are estimated for waste collectors, workers at landfills, off-site individuals who reside near landfills during operations or after closure, and individuals who might reside on the landfill sites after closure. In estimating inhalation and ingestion doses to waste collectors and workers at landfills and in estimating doses to off-site individuals during landfill operations, the thorium in the lamps is assumed to be dispersed in amounts 10 times less than loose materials in trash, due to the physical form of thorium in the lamps (see Section A.2.3.1.5). Similarly, for exposure to future on-site residents at landfills, the amount of material inhaled or ingested is assumed to be a factor of 10 less than for loose materials in the waste. However, for exposure to off-site individuals after closure of landfills, the leaching of thorium from the lamps into groundwater is assumed to be the same as the leaching of finely dispersed thorium in waste.

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

- The EDE to individual waste collectors, a future on-site resident, individual workers at landfills, and to off-site individuals during landfill operations and after closure of the landfills would each be less than 1×10⁻⁵ mSv (<0.001 mrem).
- The collective EDE from 1 year's disposals of lamps would be about 1×10⁻⁴ person-Sv (0.01 person-rem), due almost entirely to exposure to future on-site residents for 1000 years after facility closure. If exposure to future on-site residents are not taken into account, the collective EDE from exposure to waste collectors and workers at landfills would be less than 1×10⁻⁵ person-Sv (<0.001 person-rem).

3.7.4.3.2 Disposal in Incinerators

In the generic methodology for disposal in incinerators described in Appendix A.2, doses are estimated for waste collectors, workers at incinerators, and off-site individuals who reside near the incinerators during operations. In estimating doses to waste collectors and workers at incinerators, the thorium in the lamps is assumed to be 10 times less dispersible than thorium in loose materials, as in the dose assessment for disposal in landfills, due to the physical form of the lamps.

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

- The EDE to individual waste collectors, individual workers at incinerators, and individual off-site residents near incinerators would each be less than 1×10⁻⁵ mSv (<0.001 mrem).
- The collective EDE from 1 year's disposals of lamps would be less than 1×10⁻⁵ person-Sv (<0.001person-rem).

3.7.4.4 Accidents

As described in Section 3.7.3, two accident scenarios involving electric lamps containing thorium were considered in previous assessments by the AEC and Buckley et al. (NUREG/CR–1775): accidental breakage of lamps and a fire in a large warehouse. Both situations could result in exposures to thorium and its decay products, and both are considered in this assessment. Doses resulting from a fire in a home and in a small storage room are also considered. In each case the lamps are assumed to contain an average quantity of 45 μ g or 50 mg of thorium, which is the maximum amount allowed under this exemption, and both doses are estimated using the generic methodology described in Appendix A.1. All dose estimates include the contributions from ²²⁰Rn, although in most cases this contribution is negligible compared with the dose from other radionuclides in the ²³²Th decay chain.

3.7.4.4.1 Breakage of Lamps

In this assessment, a single lamp is assumed to be broken in a room in a home. An upper-bound estimate of the dose to an individual who is in the room when the lamp breaks, and who cleans up the broken lamp, is obtained by assuming that the model developed in Appendix A.1 for the dose from a spill of material in powder form applies to this scenario. This assumption should be conservative, because the amount of thorium transferred from a broken lamp into the air or to the hands of an individual should be considerably less than the amounts for a spill of material in powder form. In estimating the dose following breakage, the room volume is assumed to be 30 m³, the air ventilation rate is 1 volume change per hour, the breathing rate is $1.2 \text{ m}^3/\text{h}$, and the exposure time is 0.5 hour. Based on these assumptions, the EDE from inhalation and ingestion would be 0.002 mSv (0.2 mrem) for the maximum quantity and less than 1×10^{-5} mSv (<0.001 mrem) for the minimum quantity.

3.7.4.4.2 Fire in Home, Storage Room, or Warehouse

In the generic methodology described in Appendix A.1, doses are estimated for a fire in a home, a small storage room, or a warehouse. For each scenario, doses are estimated for individuals during the fire, either the occupant of a home or storage area or a firefighter during a warehouse fire, as well as individuals involved in cleanup activities after the fire. The results of the dose assessment based on the generic methodology are as follows.

For a fire in a home, a room in which the fire occurs is assumed to contain 10 lamps with maximum allowed quantities the dose to an individual in the room during the fire would be 4×10^{-4} mSv (0.04 mrem), and the dose during cleanup after the fire would be less than 1×10^{-5} mSv (<0.001 mrem). Doses for average use quantities are obtained by multiplying by 0.0009.

For a fire in a small storage room, the room is assumed to contain 1000 lamps with maximum allowed quantities the dose to an individual during the fire would be 0.09 mSv (9 mrem), and the dose during cleanup after the fire would be 0.001 mSv (0.1 mrem). The dose estimate during the fire would be quite conservative if the individual uses effective respiratory protection. Doses for average use quantities are obtained by multiplying by 0.0009.

For a fire in a warehouse, the storage area is assumed to contain 10,000 lamps. The dose to a firefighter using respiratory protection during the fire would be 2×10^{-4} mSv (0.02 mrem), and the dose to an individual who is not using any respiratory protection during cleanup after the fire would be 0.001 mSv (0.1 mrem). Doses for average use quantities are obtained by multiplying by 0.0009.

Buckley et al. used one million lamps for their assessment and our doses must be adjusted higher by a factor of 100 for comparison. With this adjustment, the estimated dose to a firefighter during a warehouse fire is much less than the value obtained previously by Buckley et al. (NUREG/CR-1775), as given in Section 3.7.3.2. The large reduction in estimated dose, compared with the previous result, is due primarily to the assumptions in this assessment that 0.1% of the thorium in the lamps would be released in a fire, instead of 100%; the firefighter would use a respirator with a protection factor of 1000, instead of no respiratory protection; and the firefighter would be exposed for 0.5 hour, instead of 8 hours.

3.7.5 Summary

This assessment has considered doses to the public from use —for illuminating purposes and disposal of electric lamps containing thorium. Results of the assessment are presented in Table 3.7.1 and may be summarized as follows:

Doses to individuals during routine use from lamps with typical amounts of thorium (45 μg) appear to be quite low, i.e., less than 0.001 mSv/yr (<0.1 mrem/yr), due primarily to the small amount of thorium that each lamp is allowed to contain. Doses to individuals during distribution and transport and disposal also appear to be low, for a maximum thorium context of 50 mg. The dose to an individual from routine use could be 3×10⁻⁴ mSv/yr (0.03 mrem).

• Estimates of collective dose for routine exposure scenarios are based on an assumption that 1 million lamps per year would be distributed, transported, used, and disposed. For some products covered by this exemption, the actual number could be considerably greater than 1 million per year. If the actual number of lamps containing thorium were known, dose estimates for these scenarios could be adjusted by the appropriate amount.

This assessment also considered doses to individuals resulting from accidental breakage of a lamp, from a fire in a home, a small storage room, and a large warehouse. Potential doses from breakage of a single lamp appear to be very low, due to the likelihood that only a small fraction of the thorium in a lamp would be inhaled or ingested. Potential doses from a fire in a home are low, unless a large number of lamps would be involved. Potential doses from a fire in a large warehouse are also low, if the building is assumed to contain 1 million lamps and firefighters are assumed to use respiratory protection. Potential doses from a fire in a small storage room are substantially higher if the room is assumed to contain 1,000 lamps and an individual fighting the fire does not use respiratory protection.

Exposure Scenario	Individual Annual Effective Dose Equivalent (mrem)°	Collective Effective Dose Equivalent ^b (person-rem) ^c
Distribution and transport ^b	<0.001	<0.001
Routine use	-0.001	0.00
Auto signal lamps	<0.001 ^g	0.02
<u>Disposal^b</u>		
Landfills	<0.001 ^h	0.01 ⁱ
Incinerators	<0.001 ⁱ	<0.001
Accidents or misuse ^k		NA ^m
Lamp breakage	<0.001	
Home fire	<0.001	
Storage room fire	0.008 ⁱ	
Warehouse fire	<0.001	

Table 3.7.1 Summary of Potential Radiation Doses From Use of Electric Lamps Containing Thorium ^a

See following page for footnotes.

...,

Footnotes to Table 3.7.1

^a Dose estimates are based on assumption that all lamps contain average amount of thorium of 45 μ g. Dose estimates for distribution and transport, routine use, and accidents or misuse assume that thorium has been aged for 20 years after chemical separation. Dose estimates for disposal assume that all decay products of ²³²Th are in activity equilibrium.

^b Dose estimates are based on assumption that 1 million electric lamps containing thorium are distributed, used, or disposed per year. Actual use of some types of lamps could be substantially higher, but other types of lamps apparently are not used to a significant extent at the present time.

^c 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

^d Dose estimate applies to truck drivers; UPS terminal employees, retail employees, or warehouse employees, and customers (see Section 3.7.4.1 and Table 3.6.1).

^e Dose estimates apply to use of incandescent or fluorescent lamps in home and workplace. ^f Dose estimates for individual of the maximum thorium content of 50 mg would be 4×10⁻⁴ mSv (0.04 mrem).

⁹ Dose estimate based a 4 signal lamps each containing 45 μ g. At the maximum content of 50 mg the dose would be 3×10⁻⁴ mSv (0.03 mrem).

^h Dose estimate applies to waste collectors at landfills; workers at landfills, off-site members of the public, and future on-site residents (see Section 3.7.4.3.1).

ⁱ Dose estimate applies to future on-site residents at landfills for 1,000 years after disposal due to one year's disposals.

¹ Dose estimate applies to waste collectors at incinerators; workers at incinerators and off-site members of the public (see Section 3.7.4.3.2).

^k Dose estimates apply to single occurrence of accident or misuse scenario; number of lamps assumed to be involved in each scenario is described in Section 3.7.4.4.

¹ Dose estimate applies during firefighting without respiratory protection; dose estimate during cleanup after fire is considerably less (see Section 3.7.4.4.2).

^m Not applicable.

3.8 Thorium in Germicidal Lamps, Sunlamps, and Lamps for Outdoor or Industrial Lighting

3.8.1 Introduction

In 10 CFR 40.13(c)(1)(v), persons who receive, possess, use, or transfer thorium-containing germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting are exempted from licensing requirements for source material, provided each lamp does not contain more than 2 g of thorium. The maximum amount of thorium (2 g) allowed by the exemption corresponds to an activity of 8 kilobecquerel (kBq) (0.22 microcurie (μ Ci)). This exemption was proposed on August 17, 1965 (30 FR 10203), and issued as a final rule on December 22, 1965 (30 FR 15802).

Quantitative and qualitative information on radiation doses to the public resulting from use and disposal of germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting that contain thorium in amounts allowed under the exemption, including assessments for accident situations, has been published only in the proposed rule cited above. This information is discussed in Section 3.8.3.

3.8.2 Description of Items

Thorium used in germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting is in an insoluble oxide form. The thorium normally is incorporated in the cathodes of the lamps or, alternatively, as a coating on the cathodes. In addition, the thorium normally is enclosed by an airtight glass envelope, which ensures that radioactive material (including ²²⁰Rn) would not be released from the lamp during normal use. Therefore, unless the lamps are broken, only external exposure to thorium and its decay products would be of concern.

The most common type of outdoor and industrial lamps containing thorium appears to be high-intensity discharge (HID) lamps, including mercury-vapor, metal-halide, and mercury-xenon arc lamps. These lamps are constructed with thick-walled glass envelopes designed to withstand considerable temperature variations and rough use. In addition to general outdoor or industrial lighting, HID lamps are used for roadway lighting and for lighting in large indoor structures. The lamps normally are mounted at least 3 to 4 meters above the ground or floor level, but they sometimes are mounted much higher, e.g., in sports arenas.

HID light is produced by the passage of an electric current through a gas or vapor under pressure. According to a manufacturer, two electrodes are used per light and typically for industrial lighting each electrode will contain 200 mg of tungsten of which 2% is thorium (Phone call, A. Zielinski, GE Lighting, Cleveland, OH, October 1999).

The annual distribution of thorium-containing germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting is not known. According to one manufacturer, the annual distribution of HID lamps may constitute less than 3% of the total distribution of all lamps used for lighting purposes, but the total distribution of all lamps is unknown.

3.8.3 Summary of Previous Analyses and Assessments

As indicated in Section 3.8.1, the proposed rule for this exemption contains the only previous assessments of radiation doses from use of germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting that contain allowable amounts of thorium. These assessments were performed by the Atomic Energy Commission (AEC) (30 FR 10203).

Exposure scenarios considered by the AEC included external dose during routine use of lamps, exposures following routine disposal of broken or burned-out lamps, reclamation and recycling of the thorium for use in other products, and ingestion or inhalation of thorium following accidental breakage of a lamp. Results are summarized as follows:

- Routine Use. External exposure would be the only pathway of concern, and the dose equivalent to an individual using a lamp should not exceed a few tenths of a microsievert (μSv)/yr (few hundredths of a mrem/yr). This result was based on an estimated external exposure rate of 6.5×10⁻² nanocoulomb (nC)/kg-h (0.25 microroentgen (μR)/h) at a distance of 1 meter from a lamp containing 2 g of thorium with its decay products in activity equilibrium.
- Disposal and Recycling. Routine disposal of broken or burned-out lamps in normal refuse disposal facilities is highly unlikely to result in any significant radiological impact, and reclamation and recycling of thorium in other products was considered very unlikely.
- Accidental Breakage. Ingestion or inhalation exposures could result from accidental breakage of lamps containing thorium. However, ingestion of significant quantities of thorium should be highly unlikely, due to the care that normally would be taken in cleaning up and discarding broken glass. In addition, experiments in which lamps were broken in small, unventilated rooms indicated that it is very unlikely that the resulting concentration of thorium in air would exceed the maximum permissible concentration (MPC) for members of the public that had been established in Appendix B, Table II, of 10 CFR 20 (prior to the 1991 revision of 10 CFR20). The MPC in air was based on an annual committed dose equivalent to the lungs of 15 mSv (1.5 rem) from continuous exposure.

The AEC did not consider individual doses from distribution of lamps containing thorium, and estimates of collective dose from routine use were not provided.

3.8.4 Present Exemption Analysis

This section presents estimates of dose to members of the public from routine distribution and transport, use, and disposal of germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting that contain thorium in amounts allowed by this exemption. Estimates of dose for accident scenarios involving breakage of lamps and a fire in a storage warehouse also are presented.

Dose assessments for all scenarios assume that each outdoor or industrial lamp contains 8 mg of thorium, or 32 Bq (0.88 nCi) of ²³²Th. The actual amount of thorium contained in the different
types of lamps covered by this exemption is not known but, in most cases, presumably would be much less than the maximum allowed.

During routine use, as well as distribution and transport, of any type of lamp containing thorium, external exposure to photons emitted by thorium and its decay products is the only exposure pathway of concern, because the radioactive material is normally sealed in a glass envelope. Therefore, ingestion or inhalation exposures, including exposures to ²²⁰Rn, would not normally occur. These exposure pathways would be of concern only for disposal and accidents.

3.8.4.1 Distribution and Transport

During distribution of lamps that are assumed to remain intact, external doses could be received by distribution workers and by members of the public who purchase the products. During the transport of intact lamps, external doses could be received by truck drivers and terminal workers. Members of the public along the truck routes also would receive external exposure, but their doses would be much less than the doses to workers.

Individual and collective doses during distribution and transport of germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting that contain 8 mg of thorium were estimated using the methodology described in Appendix A.3. The actual distribution of these lamps is not known, and it probably varies considerably depending on the type of lamp. In this assessment, collective doses are estimated for 1 million lamps distributed per year. Dose estimates could be adjusted to represent actual distributions if they were known.

In this assessment, the distribution and transportation network assumed for welding rods containing thorium, which was developed by McDowell-Boyer (NUREG/CR-1039) and is described in Section 3.6.4.1, is assumed to apply to a mixture of germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting.

The essential features of the assumed distribution and transport network are the following. First, 80% of the lamps are assumed to be shipped by truck to United Parcel Service (UPS) terminals over distances greater than 400 km, 10% to UPS terminals over distances between 32 and 400 km, and 10% directly to warehouses and retail stores over distances less than 32 km. Second, 80% of the lamps shipped to UPS terminals are assumed to be shipped to warehouses and 20% to retail stores. Finally, each warehouse is assumed to receive 1% of the total annual distribution of lamps, and each retail store is assumed to receive 0.1%. Thus, the assumed distribution and transportation network is appropriate for products which, first, are mostly transported by truck over appreciable distances and, second, are distributed mainly to the user but are sold in retail stores only to a lesser extent. These assumptions should be reasonable for the mixture of lamps subject to this exemption (see also Section 3.8.4.2.4).

Based on the assumptions described above, the individual and collective doses from distribution and transport of 1 million lamps per year can be obtained by scaling of the results for welding rods containing thorium given in Table 3.6.1 by the total amount of thorium distributed per year. Dose estimates for welding rods in Table 3.6.1 were based on an assumed distribution of 2.3×10^5 g of thorium per year, and the amount of thorium distributed under this exemption is assumed to be 8×10^3 g/yr. Therefore, the estimates of individual and collective doses for this exemption are a factor of 0.035 lower than the corresponding dose estimates for welding rods in Table 3.6.1. The following results are obtained:

- The effective dose equivalent (EDE) to individual truck drivers, UPS terminal employees, retail employees, and warehouse employees would be less than 1×10⁻⁵ to 7×10⁻⁵ mSv (<0.001 to 0.007 mrem), with the dose to truck drivers being the highest. The dose to individual customers in retail stores would be considerably less.
- The collective EDE for 1 million lamps distributed per year, including the dose to truck drivers, UPS terminal employees, retail store employees, warehouse employees, and customers in retail stores, would be 2×10⁻⁴ person-Sv/yr (0.02 person-rem/yr).

The results given above may be somewhat conservative, because they do not consider any self-shielding provided by the types of lamps subject to this exemption. However, the thicker glass envelopes and generally more rugged construction, when compared, for example, with electric lamps for illuminating purposes, should not reduce the dose estimates by more than a few tenths of a percent for the high-energy photons of concern in the decay of some of the thorium decay products. Such adjustments are negligible compared with uncertainties in the exposure times and distances from sources used in defining the exposure scenarios.

3.8.4.2 Routine Use

As indicated in Section 3.8.3, the AEC concluded that annual dose equivalents to individuals from exposure to a single lamp containing 2 g of thorium should not exceed a few tens of a μ Sv (few hundreds of a mrem). However, more detailed assessments of doses for specific uses of lamps and estimates of collective dose were not provided.

In this assessment, individual and collective doses to the public were estimated using Micro Shield (Computer Codes, Grove Engineering, 1996) for three routine use situations: HID lamps in a sports arena, HID lamps in a factory, and sunlamps in a home. In each case, the lamps are assumed to contain 8 mg of thorium for the average lamp or 2 g of thorium, the maximum amount allowed under this exemption, and the thorium is assumed to have been aged for 20 years after chemical separation. During routine use of lamps, only external exposure is assumed to be important. Although other exposure scenarios can be postulated (e.g., exposure to motorists traveling along streets lit by HID lamps and exposure to individuals in homes located near street lights), the scenarios evaluated should be reasonably representative and, furthermore, may provide conservative estimates of dose for other routine exposure scenarios. Dose assessments for the three exposure scenarios considered are described in the following sections.

3.8.4.2.1 HID Lamps in Sports Arena

The following assumptions were made in estimating external doses from routine use of HID lamps in a sports arena. First, the lamps were assumed to be mounted above the center of the arena in a single square array 24 meters on a side containing 81 lamps, each lamp located 3 meters from its nearest neighbor; the array was assumed to be located 23 meters above the occupants of the arena. Second, an individual was assumed to occupy the arena for 3 hours per event and to attend one event per week for an entire year; the arena was assumed to be occupied by 5000 persons per night for each night of the year. Third, all exposures were assumed to occur at a location beneath the center of the array of lamps.

For the assumptions described above, the EDE for an individual attending one event per week for an entire year would be less than 1×10^{-5} mSv (<0.001 mrem). For an attendance of 5000 persons every night for a year at the arena, the collective EDE per arena would be less than 1×10^{-5} person-Sv/yr (<0.001 person-rem/yr).

For the maximum allowed quantity of 2 g, the EDE for an individual attending one event per week for an entire year would be 4×10^{-5} mSv (0.004 mrem).

The dose estimates given above are believed to be conservative, because it is unlikely that an individual would attend an average of one event per week during a year and that an arena would be used each day of the year. In addition, most individuals in an arena would not be located beneath the center of an array of lamps, where the external dose would be the highest.

3.8.4.2.2 HID Lamps in Factory

In estimating external doses from routine use of HID lamps in a factory, the lamps were assumed to be arranged as described above for HID lamps in a sports arena, except the array of lamps was assumed to be located 6 meters above the factory workers. All exposures were assumed to occur at a location beneath the center of the array of lamps.

For the assumptions described above, the EDE for an individual worker who spends 2000 h/yr in the factory would be less than 1×10^{-5} mSv (<0.001 mrem). For a factory that is assumed to employ 50 workers, the collective EDE per factory would be less than 1×10^{-5} person-Sv (<0.001 person-rem).

For the maximum allowed quantity of 2 g per lamp, the EDE for an individual worker would be 0.003 mSv (0.3 mrem).

The dose estimates given above are believed to be conservative, because it is unlikely that individual workers would spend as much as 2000 h/yr beneath the lamps. Furthermore, when exposed to the lamps, the workers would not always be located beneath the center of an array.

3.8.4.2.3 Sunlamp in Home

In estimating external doses from routine use of a sunlamp in a home, an individual user was assumed to be located at a distance of 1.8 meters from the lamp, while actually using it, for 18 h/yr, and at an average distance of 6 meters from the lamp, while not using it, for 7300 h/yr. Other occupants of the home were assumed to be located at an average distance of 6 meters from the lamp for 4400 h/yr.

For the assumptions described above, the estimated individual and collective doses from use of a sunlamp in a single home are summarized as follows. First, for sunlamp containing 8 mg Th, the EDE for a sunlamp user and other occupants of the home, would be less than 1×10^{-5} mSv (<0.001 mrem). For the maximum allowed quantity of 2 g per lamp, the EDE to the user would be 4×10^{-4} mSv (0.04 mrem).

The dose estimates given above are believed to be conservative, primarily because it is unlikely that average individuals, either users of a sunlamp or other occupants of the home, would spend as much time in the home as assumed in this analysis.

3.8.4.2.4 Collective Dose From Mixture of Lamps

Based on the dose estimates given above for use of HID lamps in a sports arena or factory and use of a sunlamp in a home, the collective dose from a mixture of 1 million lamps can be estimated by assuming a relative mixture of lamps among the different uses considered in this assessment. The actual mixture of lamps in sports arenas, factories, and homes is unknown. In this assessment, it is arbitrarily assumed, first, that 90% of the lamps are HID lamps and 10% are sunlamps and, second, that 20% of the HID lamps are used in sports arenas and 80% in factories. If lamp replacement is ignored, these levels of lamp use would require approximately 2,200 arenas and 9,000 factories, assuming that each contains a single array of 81 lamps, and 100,000 homes, each containing a single sunlamp.

For the assumptions described above, and using values not rounded, the estimated collective EDE for 1 million lamps in use would be 0.004 person-Sv/yr (0.4 person-rem/yr). About 65% of the collective dose would result from exposures in sports arenas, somewhat less than 32% from exposures in factories, and only about 3% from exposures to sunlamps.

The estimate of collective dose for 1 million lamps in use given above should be conservative, primarily because, for the scenarios considered, the duration of exposure to an average individual to lamps and the average distance from lamps are likely to be overestimates.

3.8.4.3 Disposal

Following their useful lifetime, germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting may be sent to landfills or incinerators for disposal. In addition, these types of lamps could be recycled for salvage and reuse of valuable materials, although it is not known if this actually occurs. As noted in Section 3.8.3, the AEC previously concluded that recycling and reuse of thorium are relatively unimportant in these types of lamps.

In this section, the generic methodology described in Appendix A.2 is used to obtain estimates of individual and collective doses from disposal of the types of lamps covered by this exemption in landfills, by incineration, or by recycling.

The number of lamps subject to this exemption that might be disposed during a year is not known. As in the analysis of collective doses from routine use in Section 3.8.4.2, doses are estimated assuming that 1 million lamps are disposed per year. It is further assumed that each lamp contains 8 mg of thorium, so that 8×10^3 g of thorium is assumed to be disposed per year. Finally, the dose estimates assume that 60% of the lamps are sent to landfills, 20% to incinerators, and 20% to a metal smelter.

3.8.4.3.1 Disposal in Landfills

In the generic methodology described in Appendix A.2, doses are estimated for waste collectors, workers at landfills, off-site individuals who reside near landfills during operations or after closure, and individuals who might reside on the landfill sites after closure. In estimating inhalation and ingestion doses to waste collectors and workers at landfills and in estimating doses to off-site individuals during landfill operations, the thorium in the lamps is assumed to be dispersed in amounts 10 times less than loose materials in trash, due to the physical form of thorium in the lamps (see Section A.2.3.1.5). Similarly, for exposure to future on-site residents

at landfills, the amount of material inhaled or ingested is assumed to be a factor of 10 less than for loose materials in the waste. However, for exposure to off-site individuals after closure of landfills, the leaching of thorium from the lamps into groundwater is assumed to be the same as the leaching of finely dispersed thorium in waste.

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

- The EDE to individual waste collectors would be 1×10⁻⁵ mSv (0.001 mrem). The annual EDE to a future on-site resident, to individual workers at landfills, off-site individuals during landfill operations and after closure of the landfills would be less than 1×10⁻⁵ mSv (<0.001 mrem).
- The collective EDE from 1 year's disposals of lamps would be about 0.02 person-Sv (2 person-rem), due almost entirely to exposure to future on-site residents for 1000 years after facility closure. If exposure to future on-site residents are not taken into account, the collective EDE from exposure to waste collectors and workers at landfills would be about 8×10⁻⁵ person-Sv (0.008 person-rem).

3.8.4.3.2 Disposal in Incinerators

In the generic methodology described in Appendix A.2 for disposal in incinerators, doses are estimated for waste collectors, workers at incinerators, and off-site individuals who reside near the incinerators during operations. In estimating doses to waste collectors and workers at incinerators, the thorium in the lamps is assumed to be 10 times less dispersible than thorium in loose materials, as in the dose assessment for disposal in landfills, due to the physical form of the lamps.

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

- The EDE to individual waste collectors would be 1×10⁻⁴ mSv (0.01 mrem). The annual EDE to individual workers at incinerators, and the dose to individual off-site residents near incinerators would be less than 1×10⁻⁵ mSv (<0.001 mrem).
- The collective EDE from 1 year's disposals of lamps would be 2×10⁻⁵ person-Sv (0.002 person-rem), due almost entirely to exposure to waste collectors.

3.8.4.3.3 Reclamation and Recycling

As indicated in Section 3.8.3, the AEC concluded that reclamation and recycling of thorium in germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting were very unlikely. However, given the much greater emphasis in recent years on reuse of discarded materials, compared with the time when this exemption was established, and considering the presence of significant amounts of reclaimable material in some of the types of lamps covered by this exemption, an assessment of potential doses from reclamation and recycling has been performed in this analysis.

The dose assessment for reclamation and recycling of lamps containing thorium is based on the generic methodology presented in Appendix A.2. Potential doses to individual workers at a smelter and members of the public residing near a smelter are considered.

The extent to which the types of lamps covered by this exemption are recycled is not known. In this assessment, lamps containing a total of 1600 g are assumed to be recycled.

Based on the assumptions described above, the estimates of individual doses from reclamation and recycling of 1 million lamps per year are summarized as follows:

- For individual workers at a smelter, the EDE would be 8×10⁻⁵ mSv/yr (0.008 mrem/yr).
- For off-site members of the public, the individual EDE would be less than 1×10⁻⁵ mSv (<0.001 mrem).

These dose estimates should be quite conservative, because it seems highly unlikely that as many as 1 million lamps of the type covered by this exemption would be recycled per year.

3.8.4.4 Accidents

Two types of accident scenarios involving lamps are considered in this assessment: (1) breakage of a lamp and (2) a fire in a home, small storage room, or warehouse. In each case, the lamps are assumed to contain 8 mg of thorium for the average lamp or 2 g of thorium, the maximum amount allowed under this exemption, and doses are estimated using the generic methodology described in Appendix A.1. All dose estimates include the contributions from ²²⁰Rn although, in most cases, this contribution is negligible compared with the dose from other radionuclides in the ²³²Th decay chain.

3.8.4.4.1 Breakage of a Lamp

In this assessment, a single lamp is assumed to be broken in a room in a home. An upper bound estimate of the dose to an individual who is in the room when the lamp breaks and who cleans up the broken lamp is obtained by assuming that the model developed in Appendix A.1 for the dose from a spill of material in powder form applies to this scenario. This assumption should be conservative, because the amount of thorium transferred from a broken lamp into the air or to the hands of an individual should be considerably less than the amounts for a spill of material in powder form. In estimating the dose following breakage, the room volume is assumed to be 30 m³, the air ventilation rate is 1 h⁻¹, the breathing rate is 1.2 m³/h, and the exposure time is 0.5 h.

Based on the assumptions described above, the EDE to an individual from inhalation and ingestion following breakage of a lamp containing the average 8 mg would be 3×10^{-4} mSv (0.03 mrem). The EDE to an individual for breakage of a lamp containing 2 g of thorium would be 0.07 mSv (7 mrem).

3.8.4.4.2 Fire in Home, Storage Room, or Warehouse

In the generic methodology described in Appendix A.1, doses are estimated for a fire in a home, a small storage room, or a large warehouse. For each scenario, doses are estimated for

individuals during the fire, either the occupant of a home or storage area or a firefighter during a warehouse fire, as well as individuals involved in cleanup activities after the fire. Results of the dose assessment based on the generic methodology are as follows.

For a fire in a home, a room (27 m³) where the fire occurs is assumed to contain a single lamp containing 8 mg of thorium. The EDE to an individual without respiratory protection in the room during the fire would be 3×10^{-4} mSv (0.03 mrem), and the dose during cleanup after the fire would be less than 1×10^{-5} mSv (<0.001 mrem). For a fire involving a lamp with 2 g of thorium, the EDE would be 0.07 mSv (7 mrem).

For a fire in a small storage room, the room is assumed to contain 100 lamps. The EDE to an individual during the fire would be 0.001 mSv (0.1 mrem), and the dose during cleanup after the fire would be 2×10^{-5} mSv (0.002 mrem). For a fire involving lamps containing 2 g of thorium, the EDE to a firefighter would be 0.4 mSv (40 mrem). The dose estimate during the fire would be lower if the individual uses effective respiratory protection.

For a fire in a large warehouse, the storage area is assumed to contain 10,000 lamps, or 5% of the assumed distribution of 1 million lamps per year. This assumption should overestimate the number of lamps likely to be stored in any warehouse. The EDE to a firefighter using respiratory protection during the fire would be 3×10^{-5} mSv (0.003 mrem). The dose to an individual not using respiratory protection during cleanup after the fire would be 2×10^{-4} mSv (0.02 mrem). For a fire involving lamps containing 2 g of thorium, the EDE to a firefighter would be 0.007 mSv (0.7 mrem).

3.8.5 Summary

This assessment has considered doses to the public from use and disposal of germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting that contain up to 2 g of thorium. Results of the assessment are presented in Table 3.8.1 and are summarized as follows:

- Doses to individuals during distribution and transport, routine use, and disposal in landfills or incinerators appear to be low. The low individual doses during routine use are due primarily to the assumptions that individuals would not be located close to lamps in a sports arena or factory and, for a sunlamp in a home, that individuals would use the lamp for only short periods of time during the year and would not be located close to the lamp for significant periods of time when it is not in use. However, the collective dose could be the highest during routine use, due primarily to the assumptions about the number of individuals who would be exposed in sports arenas and factories. The estimated collective dose from disposal in landfills would be reduced by more than two orders of magnitude if the dose to future on-site residents for 1000 years after facility closure were not taken into account.
- All estimates of individual and collective doses for routine exposure scenarios, except doses to individuals during routine use, are based on an assumption that 1 million lamps per year would be distributed, transported, used, disposed, or recycled. However, the assumed number could be a considerable overestimate. If the actual number of lamps containing thorium were known, the dose estimates for these scenarios could be adjusted by the appropriate amount.

This assessment also considered doses to individuals resulting from accidental breakage of a lamp and a fire in a home, a small storage room, and a large warehouse. Potential doses from breakage of a single lamp appear to be low, due to the likelihood that only a small fraction of the thorium in a lamp would be inhaled or ingested. Potential doses from a fire in a home, small storage room, or large warehouse also appear to be high, due to the amounts of thorium that are assumed to be involved in each case and the assumption that individuals fighting a fire in a warehouse would not use respiratory protection. In addition, the dose estimates for a warehouse fire are likely to be conservative, because it seems unlikely that 5% of the total annual distribution of lamps would be stored in a single warehouse.

Exposure Scenario	Individual Annual Effective Dose Equivalent (mrem)°	Collective Effective Dose Equivalent ^b (person-rem) ^c
Distribution and transport ^b	0.007 ^d	0.02
Routine use		
Sports arena	<0.001	0.4 ^e
Factory	<0.001	••••
Sunlamp in home	<0.001	
Disposal ^b		
Landfills	0.001 ^f	2 ^g
Incinerators	0.01 ^h	0.002
Recycling	0.008'	NA ^m
Accidents or misuse ⁱ		NA
Lamp breakage	0.03	
Home fire	0.03	
Storage room fire	0.1 ^k	
Warehouse fire	0.02'	

 Table 3.8.1 Summary of Potential Radiation Doses From Use of Thorium-Containing

 Germicidal Lamps, Sunlamps, and Lamps for Outdoor or Industrial Lighting ^a

See following page for footnotes.

Footnotes to Table 3.8.1

^a Dose estimates are based on assumption that all lamps contain 8 mg of thorium for the average lamp. Actual amounts of thorium in products are unknown. Dose estimates for distribution and transport, routine use, and accidents or misuse assume that thorium has been aged for 20 years after chemical separation. Dose estimates for disposal assume that all decay products of ²³²Th are in activity equilibrium.

^b Dose estimates are based on assumption that 1 million lamps containing thorium are distributed, used, disposed, or recycled per year.

^c 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

^d Dose estimate applies to truck drivers; dose estimates for UPS terminal employees, retail employees, or warehouse employees are somewhat less, and dose estimate for customers in retail stores is considerably less (see Section 3.8.4.1 and Table 3.6.1).

^e Dose estimate is based on assumed admixture of lamps in sports arenas, factories, and homes described in Section 3.8.4.2.4.

^f Dose estimate applies to waste collectors at landfills; dose estimates are considerably less for workers at landfills, off-site members of the public, and future on-site residents (see Section 3.8.4.3.1).

⁹ Dose estimate applies to future on-site residents at landfills for 1,000 years after disposal due to 1 year's disposals; if dose to future on-site residents is not taken into account, estimate annual collective dose to waste collectors and workers at landfills is 8×10^{-5} person-Sv (0.008 person-rem).

^h Dose estimate applies to waste collectors at incinerators; dose estimates are considerably less for workers at incinerators and off-site members of the public (see Section 3.8.4.3.1). ⁱ Dose estimate applies to workers at smelters; dose estimate is considerably less for off-site members of the public (see Section 3.8.4.3.3).

¹ Dose estimates apply to single occurrence of accident or misuse scenario; number of lamps assumed to be involved in each scenario is described in Section 3.8.4.4.2.

^k Dose estimate applies during firefighting without respiratory protection; dose estimate is considerably less during cleanup after fire (see Section 3.8.4.4.2).

¹ Dose estimate applies during cleanup after fire; dose estimate is somewhat less during firefighting using respiratory protection during fire (see Section 3.8.4.4.2). ^m Not applicable.

3.9 Rare Earth Products Containing Less Than 0.25% by Weight of Source Material

3.9.1 Introduction

In 10 CFR 40.13 (c)(1)(vi), any person is exempt from the requirements for a license, to the extent that the person receives, possesses, uses, or transfers rare earth metals and compounds, mixtures, and products containing not more than 0.25% by weight of thorium, uranium, or any combination of these. This exemption was first established on March 20, 1947 (12 FR 1855). The exemption was deleted on January 14, 1961 (26 FR 284), but reinstated on April 11, 1961 (26 FR 3063).

The original *Federal Register* notice contained no information on analyses of radiological impacts on the public from use of the exempted rare earth materials. In the 1961 notice reinstating the exemption, the Atomic Energy Commission referred to the radioactivity exposure data furnished by the American Potash and Chemical Corporation and concluded that the possession and use of source material in these materials can be conducted without an unreasonable hazard to life or property. However, no quantitative estimates of radiological impacts were given.

This exemption refers only to the weight percent of any combination of thorium and uranium. However, thorium is the primary radioactive element typically found in rare earth products. Naturally occurring and processed thorium normally consists primarily of ²³²Th (nearly 100% by mass and 50% by activity) and ²²⁸Th (50% by activity). If it is assumed that 0.25% by weight of a rare earth product is thorium and ²³²Th and ²²⁸Th are in equilibrium, the activity concentration of ²³²Th and ²²⁸Th would be about 10 becquerel (Bq)/g (275 picocurie (pCi)/g) of each. This concentration refers only to the amount of thorium in the material of interest, but the thorium decay products also must be considered when estimating doses.

3.9.2 Description of Items

The rare earths are defined as a group of 17 elements composed of scandium, yttrium, and the lanthanides (Hedrick, 1993). Scandium, atomic number 21, is the lightest rare earth. Yttrium, atomic number 39, is chemically similar to the lanthanides and commonly occurs in the same minerals. The lanthanides comprise a group of 15 elements with atomic numbers 57 through 71. Due largely to differing ionic radii, the rare earth elements are broadly classified into two groups: (1) the light rare earth elements (LREE) or cerium subgroup, composed of the first seven lanthanides (atomic numbers 57 through 63), and (2) the heavy rare earth elements (HREE) or yttrium subgroup, composed of the remaining lanthanides (atomic numbers 64 through 71) and yttrium (atomic number 39). Yttrium is classed with the heavy subgroup because of its occurrence, ionic radius, and other similar properties.

The rare earths are constituents of more than 100 minerals, but only a few are recovered for commercial production. Monazite, bastnasite, xenotime, and rare earth-bearing clays are the major sources of the world's rare earth supply (Hedrick, 1993). The following subsections briefly describe the primary sources of rare earth products, monazite and bastnasite, as well as, key products derived from or containing rare earths.

3.9.2.1 Monazite

Monazite, a rare earth phosphate, is one of the most abundant rare earth minerals. It is classified as an LREE and is usually enriched in cerium (Hedrick, 1993). Heavy-mineral sands are the source of most monazite mined in the world. Monazite is mined either as a primary product or as a byproduct of titanium, zirconium, or tin minerals. Monazite also was recovered from tailings at Jackson, NJ, which were reprocessed primarily for their zircon and ilmenite content.

From a radiological perspective, the mineral monazite is of particular importance because it typically contains 5 to 7% thorium and 0.1 to 0.3% uranium (Hewson and Terry, 1995). In the United States, the thorium content in monazite is about 4% (Phone call, J. Raiser, RGC Minerals and Sands, Inc., Green Cove Springs, FL, November 1994). Assuming a 5% thorium content, the ²³²Th and ²²⁸Th activity concentration in ore is about 204 Bq/g (5500 pCi/g). Assuming a 0.2% uranium content, the natural uranium (²³⁸U, ²³⁵U, and ²³⁴U) activity concentration is about 50 Bq/g (1360 pCi/g).

The major monazite-producing countries are Australia, Brazil, China, India, Malaysia, Thailand, and the United States (Hedrick, 1994). Worldwide demand for thorium-bearing rare earth ores is low (Hedrick, 1997). Its popularity has diminished because of the potential waste problem and possible radiological hazards. Domestic mine production of monazite ceased at the end of 1994, and monazite was not imported into the United States from 1992 to 1994 (Hedrick, 1997). In 1995 and 1996, monazite was imported into the United States from Australia and Malaysia (Hedrick, 1997).

3.9.2.2 Bastnasite

Bastnasite, a light lanthanide (Ln) fluoride carbonate, occurs in an unusual type of magmaderived deposit in which the Ln elements have been enhanced (Kilbourn, 1992). The largest recognized deposit, which is mixed with monazite and iron ores, occurs in a complex mineralization at Bayan Obo in Inner Mongolia, China. The only other current commercial bastnasite source is Molycorp's Mountain Pass, CA, deposit, where the average Ln oxide content of the ore is about 9% (Kilbourn, 1992). The U.S. deposit is the only resource mined solely for its content of cerium and other lanthanides.

The thorium content in bastnasite can range from about 0.02 to 0.1% by weight and the uranium content is about 0.002% by weight (Phone call, W. Almas, Manager of Environmental and Regulatory Affairs, Molycorp, Inc., Mountain Pass, CA, November 1994). Therefore, the ²³²Th and ²²⁸Th concentration in ore ranges from about 0.8 to 4 Bq/g (22 to 110 pCi/g) and the natural uranium (²³⁸U, ²³⁵U, and ²³⁴U) activity concentration is about 0.5 Bq/g (14 pCi/g). Depending on the purity of the particular rare earth product, the thorium and uranium content can vary from trace amounts up to about 0.25% by weight (primarily as thorium phosphate) (Phone call, S. Trout, Molycorp, Inc., Fairfield, NJ, November 1996).

3.9.2.3 Rare Earth-Derived Products

A wide variety of rare earth products are available for trade and industry, including concentrates, individual and mixed compounds, and pure and alloyed metals (Hedrick, 1993). Rare earths are used in hundreds of applications, ranging from 19th-century traditional

applications to high-technology uses. The applications encompass a wide range of products from mixed rare earth compounds and alloys to ultrahigh-purity individual metals and compounds. Quantities vary by application and range from tonnage quantities to milligrams. Table 3.9.1 provides a summary of the products that contain a rare earth element. The approximate distribution of rare earths by use, based on information supplied by primary processors and some consumers, is summarized in Table 3.9.2.

Individual rare earth oxides and other compounds, except promethium, are produced from bastnasite, monazite, and xenotime ore (Hedrick, 1993). Purities from 96% up to 99.9999% are available for most oxides. Compounds for nearly all of the rare earths are available as oxides, acetates, carbonates, chlorides, fluorides, nitrates, oxalates, and sulfates. High-purity rare earth metals are marketed in the forms of sponges, lumps, ingots, crystals, rods, wires, chips, powders, sheets, foils, plates, sputtering plates, and custom cast and machined shapes. Alloys such as mischmetal (mixtures of rare earth elements in metallic form), rare earth silicide, ferrocerium, and other rare earth alloys are available in a variety of ingot shapes and sizes. Rare earth magnet alloys are marketed in ingot form and crushed ribbon, or may be purchased as mixed oxides for powder metallurgical processes.

Cerium is the most abundant member of the series of lanthanides. It is characterized chemically by having two stable valence states, Ce⁴⁺ and Ce³⁺, and this property underlies several technological uses (Kilbourn, 1992). Bastnasite and monazite are the two major mineral sources for cerium. Of the worldwide production of 64,000 metric tons (MT) of lanthanides, only a small fraction is produced as relatively pure individual Ln derivatives, including cerium (Kilbourn, 1992). The majority of the lanthanides are used as concentrates and like products. The various cerium-containing materials available commercially are summarized in Table 3.9.3. Cerium oxide slurried in water is the most efficient glass polishing agent, especially for glass compositions produced commercially in large volume (Kilbourn, 1992). Other major uses of cerium compounds include the decolorization of glass, production of radiation-resistant glass (television glass faceplates and hot cell viewing windows), and production of ultraviolet absorption glass (such as medical glassware and selected types of automobile glass) (Kilbourn, 1992).

The scandium industry, for the most part, is separate and distinct from the lanthanide and yttrium industry (Hedrick, 1993). Most scandium is produced as a byproduct of other ores. In the United States, scandium has been recovered as a byproduct during processing of uranium at a copper mine, from tailings from a mined-out fluorite mine, and from a tungsten byproduct from a molybdenum operation (Hedrick, 1993). Tailings and concentrates from these sites were shipped to processors for production of high-purity scandium oxide and other compounds. Most scandium is sold in oxide form. A small group of companies produces scandium metal, primarily for lighting and research purposes.

Yttrium is often classed with the rare earth elements because it invariably occurs with them in nature and has similar properties (Hedrick, 1993). Xenotime concentrate with a minimum 25% yttrium oxide content is a major source of yttrium (Hedrick, 1993). Except for minor amounts of yttrium contained in domestically produced bastnasite concentrates, essentially all purified yttrium was derived from imported compounds (Hedrick, 1995).

3.9.3 Summary of Previous Analyses and Assessments

When this exemption was initially established and then reinstated, little information was provided on the radiological impacts on the public from use of the materials. The 1961 *Federal Register* notice reinstating the exemption refers to the radioactivity exposure data furnished by the American Potash and Chemical Corporation (APCC), but no quantitative estimates of radiological impacts are given. However, the radioactivity exposure data for selected rare earth processing workers at APCC's West Chicago, Illinois, facility were obtained and are summarized in this section.

Other than the information available on ophthalmic glass, there is no known information available assessing the potential radiological impact on members of the public (industrial workers and other members of the public) from the use of rare earth products and materials. Most of the exposure data and studies have evaluated workers processing mineral sands. For example, a number of studies were conducted in Australia to evaluate the radiological impacts associated with the mineral sands industry. In these studies, the processing of mineral sands to extract monazite and other minerals, which can contain between 5 to 7% by weight of source material, would be licensed operations. However, this information can be extrapolated, where applicable, to similar operations in which bulk rare earth products containing 0.25% by weight of source material may be handled. Air concentrations, bioassay results, and doses to industrial workers involved with the processing of mineral sands and subsequent extraction of heavy minerals, such as monazite, are summarized. Also summarized are the radiological impacts from the processing and handling of rare earth products derived from bastnasite ore, evaluated in the Molycorp Mountain Pass facility characterization study. The potential on-site and off-site radiological impacts from disposal of rare earth processing wastes were estimated by the EPA (RAE-9232/1-2).

3.9.3.1 Routine Exposures

This section presents a brief summary of the exposure data applicable to industrial workers and off-site members of the public impacted by the routine handling and processing of monaziteand bastnasite-derived rare earth products, respectively. The processing and handling of monazite-derived rare earth products are usually licensed operations; however, the exposure data are useful when evaluating the handling of rare earth products in similar nonlicensed operations.

3.9.3.1.1 Industrial Worker

3.9.3.1.1.1 Monazite-Derived Rare Earth Products

In 1961, APCC's West Chicago, Illinois, plant was the largest refiner in the United States of thorium, rare earth chemicals, and yttrium. Lindsay Chemical Company, which previously owned the West Chicago plant, was acquired by APCC in 1958. The rare earth feedstock for the West Chicago plant was rare earth sodium sulfate, which contained 42% rare earth oxide (REO). Most of the production was in the form of rare earth and cerium products, such as rare earth chloride, rare earth oxide, rare earth fluoride, cerium hydrate, and various rare earth- and cerium-based glass polishing compounds (McCoy, 1961).

The radiation protection program at the APCC plant included film badges and air concentration measurements. All operating personnel were required to wear badges. Many of these employees transferred between the thorium (licensed) and rare earth (nonlicensed) sections of the plant. Table 3.9.4 summarizes the annual doses from direct irradiation to personnel working with crude rare earth products and purified rare earths in ion exchange columns. These doses were associated with activities that involved handling, preparation, and packaging of rare earth materials, such as rare earth and cerium oxide, cerium hydrate, rare earth sodium sulfate, rare earth chloride, and rare earth fluoride. The main effort of the air sampling program was directed toward operations involving handling of thorium materials. However, there was a record of selected air concentration results associated with the rare earth processing plant (McCoy, 1961). The air concentration data were presented as a percent of the maximum permissible level (% ML). The average % ML and derived air concentrations (DACs) for these selected rare earth processing activities are summarized in Table 3.9.5.

In the mineral sands industry, work tasks involving dry particulate product handling pose a particular risk because very high localized dust concentrations may be generated in the breathing zone of workers. These tasks include bagging operations, manual cleaning methods (e.g., shoveling, sweeping, and brushing), and certain maintenance activities. According to Hewson and Fardy (1993), long-term workers may have been exposed to thorium ore dust at significant levels of intake, i.e., on the order of the annual limit on intake. Based on this possibility, bioassay studies were performed to complement estimates of radiation doses derived from air sampling measurements. Table 3.9.6 provides a summary of the number of personal air samples taken from designated industry workers (those with the potential to receive an annual effective dose equivalent (EDE) greater than 5 millisieverts (mSv) (500 mrem)) during the time period of 1986 to 1992, together with monazite production figures and an estimate of the industry wide average alpha activity airborne concentration. According to Hewson and Terry (1995), monazite preferentially concentrates in dust by a factor of between 10 and 30.

Average alpha activity levels have decreased substantially since 1986, when the average daily intake of alpha activity was estimated to be 8.5 Bq (235 pCi/g) (assuming 10 m³ of air breathed per work shift). Through a combination of engineering practices and improved work and management practices, the average daily intake is now estimated to be 1.6 Bq (44 pCi) (Hewson and Fardy, 1993). Analysis of the data also showed that the daily variability in the alpha activity concentration is described by a geometric standard deviation (gsd) in the range of 2.0 to 2.5 for workers and supervisory staff and a gsd in the range of 3.0 to 3.5 for workers involved in maintenance activities in the dry plant (Hewson and Fardy, 1993).

Serum and urine samples were collected from both unexposed workers and from 34 mineral sands workers selected for the study (Hewson and Fardy, 1993). For just over half (56%) of the mineral sands workers, the concentration of thorium in urine was above the upper end of the range observed for unexposed workers. Thirty-one of the 34 mineral sands workers recorded a concentration of thorium in urine well above the geometric mean of the background concentration.

The conclusions of this bioassay study indicated that the absorption of thorium is considerably less than that predicted by the assessment methods. The discrepancies between measured and expected bioassay values are due to protective effects of respirators, and also may be due

to inappropriate model parameters such as lung solubility and clearance of large particles to blood (Hewson and Fardy, 1993).

Marshman and Hewson (1994) compiled individual and collective radiation doses associated with the Western Australian mineral sands industry between 1986 to 1993 for designated and nondesignated employees. Designated employees are those workers directly involved in the operation and maintenance of the processing plants. Nondesignated employees include workers involved in mining, transport, administrative, and technical services. For the designated employees, the 1992 annual mean external dose, as measured using thermoluminescent dosimeters (TLDs) was 1.5 mSv (150 mrem). The 1992 annual mean internal dose for designated employees was estimated to be 6.3 mSv (630 mrem). As stated in Marshman and Hewson (1994), respiratory equipment use was not assumed. Also, the authors cautioned that the internal dose estimates have significant uncertainty and should be interpreted with caution. Based on 212 designated workers, the 1992 annual collective EDE was estimated as 1.7 person-Sv (170 person-rem).

Reductions in airborne radioactivity occurred across the industry from 1986 to 1993. The mean airborne radioactivity concentrations decreased from 0.85 Bq/m³ (23 pCi/m³) in 1986 to 0.18 Bq/m³ (5 pCi/m³) in 1992. These mean airborne radioactivity concentrations were obtained from personal air samples of workers involved in the operation and maintenance of five mineral sands processing facilities.

Hewson and Hartley (1990) reported levels of ²²²Rn and ²²⁰Rn in air in various mineral sands mining and processing stages. The data were consistent with natural background levels, due to the relatively low uranium content and the low emanation rate of radon. However, in Hewson (1993), it is stated that routine monitoring for radon progeny may be necessary for monazite processing, since grinding and cracking of the mineral matrix will increase the emanation rate of radon. According to Hewson (1993), the realistic apportionment of dose between various exposure pathways was 50% due to external irradiation and 50% due to internal radiation (comprising 30% from long-lived alpha emitters and 20% from ²²²Rn and ²²⁰Rn progeny).

3.9.3.1.1.2 Bastnasite-Derived Rare Earth Products

Radiation dosimetry data were collected at the Molycorp's Mountain Pass California facility during a characterization study (Bernhardt, 1996). TLDs were worn by plant workers and placed in selected areas of the facility. The personnel TLD results for selected plant workers are summarized in Table 3.9.7, and the area TLD results are summarized in Table 3.9.8. Radiation surveys for surface contamination and exposure rate measurements were also conducted in selected nonproduction plant areas. The highest average exposure rate measurement was 18 nanocoulomb (nC)/kg-h (70 microroentgen (μ R)/h) (including background), observed on the bastnasite warehouse floor. Typically, the average exposure rate measurements were between 3 and 10 nC/kg-h (13 to 40 μ R/h). The total surface contamination activity ranged from zero (0) (chemical plant control room) to 440 disintegrations per minute (dpm)/100 cm² (maintenance lunch room floor). The smearable contamination ranged from zero (0) (mill lunch area) to 48 dpm/100 cm² (bastnasite warehouse floor).

The radionuclide air concentration data available for the Molycorp Mountain Pass facility was for the lead and iron filter cake re-introduction processing project. The lead and iron filter cake is a waste product that contains greater than 0.05% by weight of uranium. The lead and iron filter

cake was added into the rare earth processing stream. During this re-introduction project, the average monthly gross alpha air concentrations in the chemical processing and warehouse areas were about 0.01 Bq/m³ and 0.002 Bq/m³ ($3\times10^{-13} \,\mu$ Ci/cc and $4\times10^{-14} \,\mu$ Ci/cc), respectively. The gross alpha air concentrations did not exceed the action level of 0.06 Bq/m³ ($1.5\times10^{-12} \,\mu$ Ci/cc), which is about one-tenth of the DAC for natural uranium 0.7 Bq/m³ ($2\times10^{-11} \,\mu$ Ci/cc).

A radiation survey was conducted by Molycorp on a truck load containing three drums of bastnasite concentrate and 14 pallets of cerium concentrate (Zapolski, 1985). Inside the loaded trailer, the exposure rate readings next to the drums of bastnasite concentrate and between the pallets of cerium concentrate were about 50 nC/kg-h (0.2 mR/h) and 100 nC/kg-h (0.4 mR/h), respectively. Distance estimates between the source and the measurement points were not provided. Exposure rate measurements taken in the truck cab and outside of the loaded trailer ranged from 3 to 5 nC/kg-h (0.01 to 0.02 mR/h) (including background).

3.9.3.1.2 Members of the Public

Exposures to members of the public (other than industrial workers) result from the processing of rare earths and from the commercial use of rare earth products. Environmental TLDs have been used to assess off-site direct irradiation doses from rare earth processing facility operations. Results of the environmental TLD programs are summarized in this subsection. The extent of off-site contamination resulting from past thorium and rare earth processing waste operations are briefly highlighted. Other than the information available on ophthalmic glass, there is no known reported information available for assessing the potential radiological impact on the public from the use of rare earth products.

Environmental TLDs were placed in several work and nonwork areas in and around the Molycorp Mountain Pass Facility (Bernhardt, 1996). The primary purpose of the environmental TLD program was to evaluate the potential impacts from the Molycorp Mountain Pass lead/iron filtercake project. However, the results of the environmental TLD program also include direct irradiation levels associated with nonlicensed operations. Results of this survey are presented in Table 3.9.9. The estimated annual EDE from direct irradiation at the plant boundary was 1.6 mSv (160 mrem). According to Bernhardt (1996), this annual EDE is essentially at ambient background radiation levels.

The Texas Department of Health, Bureau of Radiation Control (BRC), performs routine environmental monitoring at sites throughout Texas, based on either the potential for environmental release of radioactive materials or at sites that were contaminated from operations conducted when controls were less stringent. Rhone-Poulenc in Freeport, TX is a company that processes rare earth hydroxides. The company is licensed to possess and store these materials prior to disposal. At one time, there were many drums containing radioactive waste stored on-site, including waste from rare earth processing operations. Off-site shipment of the waste products was still ongoing in 1992. Ambient gamma levels were monitored using TLDs, and sediment, soil, vegetation, and surface water samples were collected around the Rhone-Poulenc facility (BRC, 1992). The annual dose equivalent measured by the TLDs at the fence line ranged from about 0.2 to 15 mSv (20 mrem to 1500 mrem). At two stations, the annual radiation levels exceeded 5 mSv (500 mrem). However, occupancy at these stations was low enough that no member of the public would likely receive an annual dose equivalent in

excess of 5 mSv (500 mrem). No off-site contamination was detected at this facility (BRC, 1992).

A legacy of on-site and off-site contamination has resulted from past thorium and rare earth extraction and processing operations. One of the sites still undergoing remediation is APCC's West Chicago, Illinois, facility. Wastes consisting of a tailings pile, five sedimentation ponds, and a sludge pile of material dredged from sedimentation ponds are located in a residential community, in some cases only 140 meters from occupied homes (Jenson, 1980). Currently, there are four off-site Superfund remediation activities associated with the West Chicago facility (Phone call, D. Seely, Project Manager, U.S. Environmental Protection Agency, Chicago, IL, December 1996). The thorium and rare earth processing operations of the Maywood Chemical Works (formerly the Stephan Company) also resulted in both on-site and off-site contamination (Foley et al., 1989; Ring et al., 1989).

As mentioned in Section 3.2, the manufacture of ophthalmic glass frequently uses mixtures of rare earths and zirconium oxides containing natural uranium and thorium. Buckley et al. (NUREG/CR–1775) performed an analysis of radiological impacts on the public from eyeglasses containing thorium. Analyses of doses from wearing of eyeglasses containing 0.05% by weight thorium considered exposure to the eye, primarily from irradiation by alpha particles, and whole-body exposures from photons. Dose estimates reported by Buckley et al. (NUREG/CR–1775) and the National Council on Radiation Protection and Measurements (NCRP 95) assumed that an individual wears eyeglasses for 16 h/day and that 55% of the population wears glasses. These dose estimates are summarized as follows:

- The annual dose equivalent to the cornea from alpha particles is about 0.04 Sv (4 rem). Assuming a tissue weighting factor of ≤10⁻⁴, the annual individual EDE from exposure to the cornea was estimated to be ≤0.004 mSv (0.4 mrem).
- The annual dose equivalent to the whole body of the individual from photons is about 2×10^{-4} mSv (0.02 mrem).

NCRP 95 estimated that 50 million people in the United States wear eyeglasses with glass. Assuming an annual individual EDE of 0.004 mSv (0.4 mrem) from alpha particles and a wholebody dose equivalent from photons of 2×10^{-4} mSv (0.02 mrem), the annual collective EDE to the U.S. population would be about 200 person-Sv (20,000 person-rem).

3.9.3.2 Disposal

Very little is known about the disposal of rare earth products. One source of information is an inspection report (New York State, 1994) that identified elevated radiation levels associated with slag from the production of additives rejected by two steel companies. Most of the available information is on the disposal of rare earth processing wastes. For example, the EPA (RAE–9232/1–2) conducted a generic dose assessment that evaluated the potential on-site and off-site impacts from disposal of monazite-derived rare earth processing wastes. In addition, a draft environmental impact report (ENSR, 1996) was prepared that describes the ²³²Th and ²³⁸U activity concentrations in bastnasite-derived rare earth processing wastes. In the draft environmental impact report, annual waste production estimates are given for the Molycorp Mountain Pass facility. However, the initial processing of monazite would be licensed, and the initial processing of bastnasite would likely fall under 10 CFR 40.13 (a). Disposal of monazite

initial processing of bastnasite would likely fall under 10 CFR 40.13 (a). Disposal of monazite and bastnasite processing wastes would be similarly regulated. However, this information is potentially useful for evaluating the EDEs attributed to the disposal of bulk rare earth products.

SKW Alloys of Niagara Falls, NY, produced additives for steel manufacturing (New York State, 1994). On April 26, 1994, a load of additives was rejected by a steel company because radiation alarms were set off. On April 28, 1994, a second load was rejected from another plant. These shipments were returned to SKW. Inspection of the slag from the first rejected load showed elevated radiation levels (maximum reading of 77 nC/kg-h (0.3 mR/h) was measured). SKW suspected that the problem derived from a previously prepared additive that contained cerium, so radiation surveys were conducted. The highest radiation levels measured in a room containing slag ranged from 1.7×10^{-4} to 1.9×10^{-4} mSv/h (0.17 to 0.19 mrem/h). Samples of slag, metal, and cerium product were collected and analyzed. The ²²⁸Ra and ²²⁶Ra activity concentrations from the slag were greater than that found in the cerium product or in the metal additives, as shown in Table 3.9.10. The slag also contained lower concentrations of ²²⁴Ra as compared to ²²⁸Ra, indicating that the thorium decay chain was not in equilibrium. Thorium was not analyzed in any of the samples. Off-site samples contained only background levels of ²²⁸Ra, ²²⁴Ra, and ²²⁶Ra radioactive materials.

Mineral processing facility waste is generally disposed in tailing ponds or used to construct dams, dikes, or embankments. Small amounts of waste have been used off-site for backfill, aggregate production, or road building (EPA, RAE–9232/1–2). The EPA (RAE–9232/1–2) conducted an assessment of a generic REO mine and mill site located in a hypothetical site in North Carolina, where ores associated with REO are composed of 50% monazite sands and have elevated levels of thorium and uranium. The model site consists of commingled waste rock, overburden, and a tailings pile of 1 hectare. The tailings have no further use other than reprocessing for their mineral content. The assessment assumes 200,000 MT of tailings, overburden, and wastes and that a 1-meter thick cover is placed over the waste when the site is closed. At the generic site, the ²³²Th, ²²⁸Th and ²²⁸Ra activity concentrations were 74 Bq/g (2000 pCi/g) and the ²³⁸U, ²³⁴U, ²³⁰Th, and ²²⁶Ra activity concentrations were 33 Bq/g (900 pCi/g). The ²¹⁰Pb, and ²¹⁰Po activity concentrations were 23 Bq/g (630 pCi/g) and the ²³⁵U, ²³¹Pa, and ²²⁷Ac activity concentrations were 1.7 Bq/g (45 pCi/g). This source term was considered to be representative of rare earth processing waste streams (EPA, RAE–9232/1–2).

Annual individual EDEs are estimated for individuals residing on the disposal site. Annual individual EDEs are also estimated for off-site individuals residing 100 meters from a disposal site. The exposure pathways evaluated for the on-site individuals residing on the disposal site are direct irradiation and dust inhalation. Ingestion of food grown on-site was not considered since the rare earth wastes generally do not have the textural and nutritional properties, and water-retention capabilities to support vegetation (EPA, RAE–9232/1–2). A number of potential exposure pathways were evaluated for the individual residing 100 meters from a disposal site. The estimated annual individual EDEs for these on-site and off-site receptors are summarized in Table 3.9.11.

Annual collective dose equivalents from disposal of rare earth processing waste estimated by the EPA (1993) are summarized in Table 3.9.12. The population density for atmospheric pathways was assumed to be 136 persons per square mile and the population using river water as a drinking water supply was assumed to be 235,000 (EPA, RAE–9232/1–2).

Molycorp commissioned a study to inventory and analyze various wastes, feedstock, and products associated with the Mountain Pass California facility (ENSR, 1996). Table 3.9.13 shows the average concentrations of ²³²Th and ²³⁸U in the feedstock bastnasite ore, in bastnasite concentrate, and in some of the wastes produced during rare earth processing. The lead and iron filter cake contained source material in excess of the 0.05% by weight Nuclear Regulatory Commission criterion (10 CFR 40.13(a)). Molycorp obtained an addendum to its license for the possession and use of radioactive sealed sources to cover the management and possession of uranium and thorium contained within the lead and iron filter cake (ENSR, 1996). Molycorp is required to feed stabilized lead and iron filter cake to the chemical process for the lanthanide recovery or dispose of stabilized material at an approved disposal facility. Molycorp generates small volumes of solvent extraction (SX) crud from the solvent extraction cells used for the separation and recovery of lanthanides. SX crud is an insoluble organic complex containing significant concentrations of both uranium and thorium, as shown in Table 3.9.13. Prior to 1984, Molycorp deposited lead and iron residues in three small surface impoundments. Molycorp is required to submit a plan for closure of these lead ponds. As a result of process changes instituted by Molycorp, radionuclide concentrations within the lead sulfide concentrate product are being held below 0.05% by weight for uranium and thorium combined (ENSR, 1996). The radiological impacts associated with the disposal of these rare earth processing wastes were not provided in ENSR (1996).

3.9.4 Present Exemption Analysis

In the present exemption analysis, annual individual and collective EDEs are estimated for truck drivers and industrial workers handling bastnasite and cerium concentrate. In addition, annual individual and collective EDEs are estimated for the use of some consumer products that contain rare earth products. These consumer products include pink ophthalmic glass, television glass faceplates, and specialty automotive window glass. The generic disposal methodology is used to estimate the individual and collective EDEs associated with the landfill disposal of bastnasite-derived rare earth products. The generic accident methodology was used to estimate EDEs associated with a warehouse fire containing cerium concentrate and a fire during transport of cerium concentrate.

3.9.4.1 Distribution and Transport

The results of the radiation survey described in Section 3.9.3.1.1.2 (Zapolski, 1985) are used in the present assessment to estimate individual and collective annual EDEs to truck drivers transporting bastnasite and cerium concentrate. The measured absorbed doses in the Molycorp bastnasite warehouse area (Bernhardt, 1996) are used to estimate the annual EDEs associated with the storage and distribution of bastnasite and cerium concentrate products.

Bastnasite and cerium concentrates are assumed to be transported on pallets from California to a product user located in the northeast. The direct irradiation exposure rate measurements taken in the cab of a truck transporting 14 pallets (1350 kg per pallet) of cerium concentrate (averaging about 0.25% by weight of thorium phosphate) and four drums of a bastnasite concentrate (0.1% by weight of thorium phosphate) were 3 to 5 nC/kg-h (0.01 to 0.02 mR/h), including background (Zapolski, 1985). If the natural radiation background is about 3 nC/kg-h (0.01 mR/h), the exposure rate from shipment is assumed to be 3 nC/kg-h (0.01 mR/h). To transport this shipment across the United States, it is assumed the truck driver is exposed for

about 50 hours, with an estimated EDE of 0.005 mSv (0.5 mrem). If this driver makes 30 similar trips across the United States in 1 year, the estimated annual EDE is 0.2 mSv (20 mrem). Industry sources indicate that approximately four shipments of product are transported weekly to the eastern United States, resulting in about 200 annual shipments. Assuming all of these shipments are in a truck, the estimated annual collective EDE is approximately 0.001 person-Sv (0.1 person-rem). These doses represent those near the maximum allowed concentration which may also approximate the average use concentrations for this product.

To estimate the annual EDE to warehouse workers involved with the storage and distribution of bastnasite and cerium concentrate, the measurements reported by Bernhardt (1996) are used in the current assessment. According to Bernhardt (1996), the 40-hour absorbed dose was estimated to be 0.02 milligray (mGy) (2 mrad) in the bastnasite bagging and warehouse area (Table 3.9.8). It is assumed that the area TLDs were located in the main operational area. Bastnasite concentrate contains about 0.1% by weight of thorium phosphate. If workers spend about 500 h/yr in the warehouse, the estimated annual EDE is 0.3 mSv (30 mrem).

Extrapolation of the 40-hour absorbed dose rate attributed to bastnasite concentrate (0.1% by weight of thorium phosphate) to the storage of cerium concentrate (0.25% by weight of thorium phosphate) results in an estimated 40-hour absorbed rate of 0.05 Gy (5 mrad). If a warehouse worker spends about 500 h/yr in the warehouse, the estimated annual EDE is 0.6 mSv (60 mrem).

Gross alpha airborne concentrations were sampled in warehouses A and B during the lead and iron filter cake reintroduction project (Bernhardt, 1996). The measured average gross alpha air concentrations were about 0.002 Bq/m³ (4×10⁻¹⁴ μ Ci/cc). Warehouses A and B are used to store rare earth products (separate from the bagging operations) and lead and iron filter cake that contained greater than 0.05% by weight of uranium. However, similar airborne alpha activity concentrations were also measured within the guardhouse, which is considered to be a background location. If it is assumed that the alpha activity is due to ²³²Th in equilibrium with its progeny and that workers spend 500 h/yr in the warehouses that may contain bastnasite and cerium concentrate, the estimated annual EDE due to dust inhalation is 0.46 mSv (46 mrem).

Therefore, the annual individual EDEs due to direct irradiation and dust inhalation to warehouse workers handling bastnasite and cerium concentrates are estimated to be 0.7 and 1 mSv (70 mrem and 100 mrem), respectively. It is unknown how many warehouse workers are actually involved in rare earth product warehouse operations. If it is assumed that there are two workers who spend 500 h/yr in a warehouse and there are about 100 facilities handling bastnasite and cerium concentrates, the annual collective EDEs are assumed to be 0.1 person-Sv and 0.2 person-Sv (10 person-rem and 20 person-rem), respectively.

3.9.4.2 Routine Operations

In this subsection, the annual individual and collective EDEs from routine exposures to industrial workers and other members of the public are estimated. For the industrial worker scenario, the bagging operations associated with a bastnasite concentrate containing 0.1% by weight of source material and a cerium concentrate containing 0.25% by weight of source material, as thorium, are evaluated. This job activity is used to represent the upper bound of potential industrial worker exposures, especially for workers handling bulk rare earth product.

Similar types of exposures may be possible for maintenance personnel. In addition, annual individual and collective EDEs are calculated for the viewing of television with glass faceplates and for the use of specialty automotive window glass. Individual and collective EDEs from pink ophthalmic lenses are based on previous dose analyses.

3.9.4.2.1 Industrial Workers

Industrial workers involved in bulk rare earth product handling are considered to represent the upper bound of likely exposures. Similar types of exposures are possible for maintenance personnel. In this assessment, the bagging operations associated with bastnasite and cerium concentrates are evaluated. The two primary exposure pathways associated with the handling of bulk rare earth materials are dust inhalation and direct irradiation.

For the purposes of this assessment, the ²³²Th activity concentration of 3 Bg/g (85 pCi/g), as shown in Table 3.9.13, is used for the bastnasite concentrate and the ²³²Th activity concentration of about 7.7 Bq/g (213 pCi/g) is used for the cerium concentrate. These activity concentrations correlate to the 0.1% and 0.25% weight as thorium phosphate found in the bastnasite and cerium concentrate products, as compared to the higher activity concentrations associated with the weight percent source material as thorium (1% by weight in bastnasite concentrate 4 Bq/g (110 pCi/g)) and 0.25% by weight in cerium concentrate 10 Bq/g (275 pCi/g), respectively. As of 1995, the Threshold Limit Value-Time Weighted Average (TLV-TWA) for respirable particulate (particulate not otherwise classified) was changed from 5 mg/m³ to 3 mg/m³, which is lower than the dust loading value of 5 mg/m³ measured by Hewson and Terry (1995). Assuming that the respirable dust loading was 3 mg/m³ and that ²³²Th is in equilibrium with its progeny, the estimated annual EDE (exposure duration of 2000 hours, no respiratory equipment) is about 10 mSv and 30 mSv (1000 mrem and 3000 mrem), respectively. If a filter-type dust respirator with an assigned protection factor (APF) of 10 is worn (National Institute for Occupational Safety and Health (NIOSH), 1990), the annual EDEs would be reduced to about 1 and 3 mSv (100 mrem and 300 mrem), respectively.

As shown in Table 3.9.7, the average 168-hour external absorbed dose associated with bastnasite and cerium bagging operations was about 0.03 mSv (3 mrem) and 0.07 mSv (7 mrem), respectively (Bernhardt, 1996). Assuming a 12-month worker exposure, the annual individual EDE from direct irradiation in the bastnasite and cerium bagging areas would be about 0.4 mSv (40 mrem) and 0.8 mSv (80 mrem), respectively.

Therefore, if a worker handled bastnasite concentrate for approximately 2000 h/yr under the given conditions, the annual EDE from dust inhalation (without using respiratory equipment) and direct irradiation would be about 10 mSv (1 rem). If a worker handled a cerium concentrate product for about 2000 h/yr under the given conditions, the annual EDE from dust inhalation (without using respiratory equipment) and direct irradiation would be about 30 mSv (3 rem). It is unknown how many U.S. workers are performing similar operations; however, if 100 workers are conducting bastnasite and cerium packaging (and like activities), the annual collective EDEs would be about 1 person-Sv and 3 person-Sv (100 person-rem and 300 person-rem), respectively.

3.9.4.2.2 Other Members of the Public

Some of the numerous products containing rare earths that are likely to be used by the public include pink ophthalmic glass, television faceplates, special application automotive window glass, and nickel metal hydride batteries. Certain types of ophthalmic glass lenses may contain cerium, such as glass lenses that have a pink tint (Phone call, W. Price, Product Assurance, Corning, Inc., Corning, NY, November 1996). This product is manufactured in Brazil and currently is not used much in the United States. Since cerium is used in glass to attenuate ultraviolet radiation and prevents discoloration, it is used in television faceplates and specialty automotive glass. Nickel metal hydride batteries may contain lanthanum, cerium, neodymium, and praseodymium. However, only high-purity rare earth elements, containing very low concentrations of thorium, are used in the manufacturing of batteries (Phone call, L. Houston, Chemist, EverReady Energizer, Inc., Gainesville, FL, November 1996). For the purposes of the current assessment, exposures from the use of pink ophthalmic glass, television faceplates, and automobile windshields containing cerium are evaluated.

3.9.4.2.2.1 Ophthalmic Lenses

Regardless of the origin of the radioactive material, whether it is in the zirconium oxide or cerium oxide, the ophthalmic glass industry has a radiation standard to which it adheres voluntarily. The evaluations conducted by Buckley et al. (NUREG/CR-1775) and the NCRP (NCRP 95), which are described in Section 3.2, incorporated the radiation limits of this standard and are therefore considered to be applicable to this assessment. The individual annual dose equivalent to the cornea from alpha particles was estimated to be 0.04 Sv (4 rem) and the annual whole-body dose equivalent from photons was estimated to be 2×10⁻⁴ mSv (0.02 mrem) (NUREG/CR-1775; NCRP 95). In NCRP 95, a tissue weighting factor of ≤10⁻⁴ was used to derive the EDE from the alpha particle dose equivalent to the cornea. In this assessment, the annual EDE due to ophthalmic glass evewear is based only on the photon whole-body dose equivalent. These doses assume 0.05% thorium in the glass, the estimated individual EDE for this section would be five times greater for the maximum allowed thorium content of 0.25%. As exact average content is not known, the maximum content allowed will be used for pink lenses. In Section 3.2, it was established that ten million people wore corrective eyewear that contained glass. If it is assumed that 1% of the glass evewear population used pink ophthalmic glass (about 100,000 people), the hypothetical annual collective EDE due to photon irradiation is 0.1 person-Sv (10 person-rem).

3.9.4.2.2.2 Television Faceplates

Television glass faceplates contain about 0.2 to 0.3% cerium oxide (Phone call, D. Johnson, Consumer Electronics Manufacturers Association, Electronics Industry Association (EIA), Arlington, VA, December 1996). Television faceplates are subjected to bombardment by high-energy electrons, particularly with the high tube voltages needed for color displays. This bombardment over time tends to cause discoloration, or browning, of the glass due to the creation of color centers. This unwanted effect is suppressed by the addition of cerium oxide to the glass (Kilbourn, 1992). If 0.3% cerium oxide is used in television faceplates and there is 0.25% by weight of thorium in the cerium product, then the ²³² Th activity concentration is about 0.03 Bq/g (0.8 pCi/g) in the television faceplate.

MicroShield (Computer Codes, Grove Engineering, 1996) was used to estimate the EDE rate to television viewers from gamma radiation. Assuming the dimensions of a 68.6-cm (27-inch) television faceplate are 53.34 cm \times 43.2 cm \times 0.3 cm and there is an average distance of 1.5 meters between the television faceplate and the viewer, the estimated EDE rate is 6×10^{-9} mSv/h (6×10^{-7} mrem/h). If a television viewer spends about 3 h/day watching television (EPA/600/P–95/002Fa), the annual EDE is calculated to be about less than 1×10^{-5} mSv (<0.001 mrem). In 1995, approximately 23 million color televisions were sold (EIA, 1996). If 250 million televisions were viewed for 3 hours per day at the given geometric dimensions, the annual collective dose equivalent is estimated to be 2 person-Sv (200 person-rem).

It is unclear how much, if any, cerium oxide is present in computer monitor faceplates. It is assumed that these faceplates contain 0.2 to 0.3% cerium oxide as an upper bound estimate. Further, if 13 million personal computers were sold in 1997 and if they are assumed to last for 5 years, taking into account built-in obsolescence, there would be about 65 million monitors in use at any time. Taking into account the amount of time an average person might spend in front of a computer at home or at work (e.g., 2000 hours) and the closer distance to a monitor compared with a television set, about 20 inches, the hypothetical individual annual EDE would be 3×10^{-5} mSv (0.003 mrem). The maximum annual EDE for this product would be proportional to the total time spent but it is unreasonable to estimate above twice this value. The hypothetical collective EDE would be 65 million monitors times the individual annual EDE or 2 person-Sv (200 person-rem).

3.9.4.2.2.3 Automotive Window Glass

Not all automotive glass contains cerium. Cerium is used in specialty automotive glass primarily to attenuate UV radiation and reduce solar heating (Phone call, M. Purvis, Libbey-Owens-Ford Company, Toledo, OH, November 1996). The market for this type of glass is primarily in higher priced or luxury automobiles. It was widely used in 1988 and 1989, but the use of cerium in automotive window glass has been reduced since then due to economic considerations. It is possible that up to about 1.5% of the automotive glass composition may be cerium (Phone call, P. Higby, Libbey-Owens-Ford Company, Toledo, OH, November 1996). If there is 0.25% by weight of thorium in the cerium product and 1.5% of the glass composition is cerium, then the ²³² Th activity concentration is about 0.15 Bq/g (4 pCi/g) in the automotive glass.

MicroShield (Computer Codes, Grove Engineering, 1996) was used to estimate the potential EDE rate to motorists from gamma radiation from the glass. Assuming that the windshield glass dimensions are about 157 cm \times 83.8 cm \times 0.64 cm and that the distance between the driver and windshield is 0.6 meter, the EDE rate is calculated to be about 1.4×10^{-6} mSv/h (1.4×10^{-4} mrem/h). If the side window glass dimensions were 83.8 cm \times 61 cm \times 0.64 cm and the distance between the driver and the distance between the driver and the side window was 0.3 meter, the EDE rate is estimated to be 3×10^{-6} mSv/h (3×10^{-4} mrem/h). If an individual spends 80 minutes per day in an automobile (EPA/600/P–95/002Fa), the annual EDE is estimated to be 0.002 mSv (0.2 mrem). It is unknown how many vehicles contain this type of glass. Assuming there are 800,000 vehicles with this type of window glass, the annual collective dose equivalent is estimated to be approximately 1 person-Sv (100 person-rem). This dose is based on the maximum allowed thorium content as no date is available regarding industry practices.

A maximum annual EDE would be to a commercial driver and is 0.007 mSv (0.7 mrem) based on 2000 h/yr driving, e.g., taxi driver. Assuming 200,000 vehicles of this type, the annual collective dose equivalent is estimated also to be about 1 person-Sv (100 person-rem).

3.9.4.3 Disposal

To evaluate the disposal of a rare earth product containing 0.25% by weight of source material, the generic disposal methodology, as described in Appendix A.2, is used to estimate the annual individual and collective EDEs from landfill disposal. The quantity disposed and the possible composition of these products when disposed is unknown. For the purposes of this assessment, it is assumed that 900 MT (9×10^5 kg) of cerium concentrate (0.25% by weight of source material as thorium) is disposed annually in the United States. As cerium concentrate contains about the maximum allowed concentration of 0.25% by weight, average and maximum EDEs will be about the same value. Assuming there are about 3,500 sanitary landfills in the United States, this equates to about 270 kg of product disposed annually in each landfill. Summarized in Table 3.9.14 are the annual individual and collective EDEs from disposal of cerium concentrate in municipal landfills.

Cerium may be used in certain metal and alloy additives. During the production of additives, radium appears to concentrate in the slag. As noted in Section 3.9.3.2, the measured dose rates ranged from 1.7×10^{-4} to 1.9×10^{-4} mSv/h (0.17 to 0.19 mrem/h) in a room where slag was stored. Assuming that a slag worker is exposed 24 hours in a year (Hill, et al., see references 1995), the annual individual EDE is estimated to be 0.05 mSv (5 mrem). Background radionuclide concentrations were indicated in off-site soil samples.

3.9.4.4 Accidents

In this evaluation, the warehouse and transport accident scenarios were selected to represent upper bound doses associated with a product that contains 0.25% by weight of source material. It is assumed that 1000 bags of cerium concentrate (45-kg bags) are stored in a warehouse or are being transported. If 0.25% by weight source of material is thorium, the ²³²Th activity concentration is 10 Bq/g (275 pCi/g). It is difficult to imagine a fire causing a significant fraction of cerium concentrate to become airborne; however, fire fighting activities could disperse the material and clean up workers could encounter contamination. Using the generic accident methodology, as described in Appendix A.1, the EDE to a firefighter combating a warehouse fire is about 0.05 mSv (5 mrem) and the EDE to a firefighter combating a transportation fire is about 0.3 mSv (30 mrem). The estimated EDEs to spill cleanup workers are 0.2 mSv (20 mrem) for the warehouse fire and 2 mSv (200 mrem) for the transportation fire. No respiratory equipment is assumed for the cleanup workers, unlike for the firefighters. However, some type of respirator would be worn by cleanup personnel. If dust-type respirators with an APF of 10 were used, the estimated EDEs for spill cleanup workers at a warehouse fire and transportation fire would be 0.1 mSv (10 mrem) and 0.2 mSv (20 mrem), respectively.

3.9.5 Summary

Numerous products may contain rare earth materials. Depending on the rare earth material used, negligible amounts up to 0.25% by weight of source material may be present. Selected rare earth products and byproducts were used to represent the upper bound of doses that could

be attributed to rare earth products. The annual and collective EDEs estimated in the current assessment are summarized in Table 3.9.15.

Based on this analysis, the following general conclusions about the radiological impacts on industrial workers and other members of the public were obtained:

- Upper bound doses to industrial workers involved in routine handling operations, especially during bagging operations, may be significant. This is due primarily to the dust inhalation pathway. However, based on current information, average doses to other industrial workers handling bulk rare earth products are likely to be lower than those estimated for the bagging operation (and like activities).
- The primary exposure pathways for industrial workers appear to be dust inhalation and direct irradiation. Radon emanation (²²²Rn and ²²⁰Rn) from rare earth products appear to be low except during certain processing operations and in certain waste streams. There was very little known information associated with radon exposures.
- It is unknown how many U.S. industrial workers (and exposure durations) may be involved with the handling of bulk rare earth products that contain 0.25% by weight of source material and whether respiratory protection is routinely used.
- The annual individual EDEs to the public from the glass used in television faceplates and automotive glass is primarily from direct irradiation.
- Ophthalmic glass is subject to Optical Industry Association alpha particle emission rate standards. These standards are applicable to alpha particle emission rates regardless of source origin (e.g., zirconium oxide or cerium oxide). In addition, pink ophthalmic glass is used very little in U.S. eyewear.

Other observations made during completion of this radiological assessment include:

- The primary rare earth producers and many users, such as the battery and glass manufacturers, appear to be well aware and/or have taken steps to reduce the thorium content of their products. Lower grade rare earth products, such as bastnasite or cerium concentrates, are more likely to contain thorium, whereas higher grade products contain small or trace amounts of thorium.
- Additional information on certain areas is necessary to accurately assess the potential radiological impacts associated with this exemption. These areas are (1) rare earth product use and corresponding thorium content, (2) radionuclide airborne activity concentrations both within the work place setting and at off-site receptor locations, and (3) current disposal practices.

Element Product Cerium Glass, metal, and gemstone polishing Radiation shielding glass Glass colorization and decolorization **Discolorization stabilizer** Photochromic glass opacifier Ceramic pigment Carbon arc lighting Incandescent lamp mantle Trichromatic fluorescent lighting phosphor (yellow green) Pyrophoric alloys, including lighter flints Cast iron alloys, aluminum alloys, magnesium alloys, superalloys Automotive catalytic converter monolith Heat stabilizer in alumina catalyst Fluid cracking catalyst Hydrolysis of phosphoric acid esters Oxidation of naphthalene Polymerization catalyst for olefins Corrosion inhibitor Getter in azide manufacture, getter in vacuum tubes Developing agent for aniline black Heavy metal soaps for lubricating greases and flatting agents Waterproofing and mildew proofing Dying and printing textiles Ceramic capacitors Semiconductors Cathodes Desulfurizing catalysts High-strength glass Catalyst for ammonia production Illuminant in photography Radiation dosimeter Diluent in plutonium nuclear fuels Fluorinating agent Oxidant for organic compounds Antiknock fuel additive Oxidizer in self-cleaning ovens Pharmaceutical for sea sickness and morning sickness Dysprosium Permanent magnet addition (NdFeB) Nuclear reactor control (DvP) Measurement of neutron flux Atomic weight determination High-temperature sensing phosphor

Table 3.9.1 Products Derived From Rare Earth Elements *

Table 3.9.1 Products Made From Rare Earth Elements^a (continued)

...

Element	Product			
Erbium	Fiber-optic amplifier Glass colorizing of crystal and glassware (light pink) Additive for colorization and nonreflective coating for sunglasses Infrared-absorbing glass Colorization of simulant gemstones Lasers Burnable poison in nuclear reactor Chemical catalyst for n-hexane Permanent magnet additive			
Europium	Cathode-ray tube phosphor (reddish orange) Trichromatic fluorescent phosphor (reddish orange and blue) X-ray intensifying screen phosphor (violet) Metal halide lamp phosphor (reddish orange) High-and low-temperature sensing phosphors Nuclear reactor neutron absorber Nuclear reactor control rods Ceramic sensor Organic shift reagent in nuclear magnetic spectroscopy Dopant in laser crystals Tagging agent in postage stamp glue			
Gadolinium	Trichromatic fluorescent lighting phosphor (yellow green) X-ray intensifying screen phosphors (yellow green) Nuclear control rods Nuclear reprocessing Neutron-absorbing paint High refractive index paint Synthetic garnet for computer memory (GGG) Scavenger for oxygen in titanium production Dielectric ceramics Laser crystal Cryogenic refrigeration Pharmaceutical radionuclide in scanning devices to detect osteoporosis (¹⁵³ Gd) Magneto-optical recording			
Holmium	Dopant in laser crystal Getter in vacuum tubes Refractories			

Table 3.9.1 Products Made From Rare Earth Elements^a (continued)

•••.

.

Element	Product			
Lanthanum	Petroleum fluid cracking catalyst Glass additive to increase refractive index and decrease dispersion (eyeglasses, cameras, binoculars, rifle scopes, and telescope lenses)			
	Nickel hydride rechargeable batteries Hydrogen storage alloys			
	Optical and laser glass			
	Solid oxide fuel cell cathode and interconnect			
	Trichromatic fluorescent lighting phosphor (green) X-ray intensifying screen phosphor (violet, blue, yellow-green and blue green)			
	Low-temperature sensing phosphor			
	Phosphor lamp coating			
	Calcium lighting			
	Lighting electrode			
	Tungsten welding electrode			
	Cast iron alloys, steel alloys, superalloys			
	Aromatization of n-hexane			
	Fluoride determination			
	Conductive ceramic neating elements (chromites)			
	Electro-ontical caramics			
	Dimerization of methane to ethylene			
	Antiseptic			
	Electronic thermo-emissive cathode (boride)			
	Reducing agent to convert nitrogen oxides to nitrogen			
	Low-temperature superconductor			
	Cryogenic refrigerant using hydrogen (LaNi ₅)			
	Superconductors (LaAs, LaSb, LaP)			
Lutetium	High-temperature sensing phosphor			
	Nuclear technology			
Neodymium	Permanent magnets (NdFeB)			
	Glass colorization (violet)			
	Dopant in laser crystals and glass			
	Ceramic capacitors			
	Ceramic glaze (pinkish violet)			
	Magnesium alloys			
	vvelder's protective glass			
	Carbon arc lighting electrodes			
	Fiber-optic temperature sensor			

3.9.1 Products Made From Rare Earth Elements^a (continued)

···.

Element	Product		
Praseodymium	Ceramic pigment (yellow) Glass colorization (light green) Refractory ceramic heating element Laser crystal Aluminum alloys Magnesium alloys Permanent magnets Oxidative dimerization of methane to ethylene Cryogenic refrigeration (PrNi ₅)		
Promethium	Fluorescent lighting starter Coatings for self-luminous watch dials Beta particle source for thickness gauges Miniature atomic batteries Semiconductor battery X-ray source (¹⁴⁷ Pm)		
Samarium	Permanent magnets (SmCo) Dopant in laser crystals Neutron absorber Nuclear reactor control rod Catalyst in dehydrogenation of ethyl alcohol Catalyst in oxidative dimerization of methane to ethylene Infrared absorbing glass		
Scandium	Metal halide lamps to improve color appearance and produce high output Laser crystal (GSGG) Aluminum alloys Dual-anode tube in X-ray spectrometers Semiconductors Tracer material Collimators in neutron lenses		
Terbium	Trichromatic fluorescent lighting phosphor (yellow green) X-ray intensifying screen phosphors (yellow green, violet, blue) Magnetostrictive alloys for transducers (TbFeNi) Dopant in laser crystals		
Thulium	X-ray source (¹⁷⁰ Tm) Dopant in laser crystal Ferrites		

11

Element	Product
Ytterbium	Portable X-ray source for blood irradiation Dialectic ceramics Carbon roads for industrial lighting Dopant in synthetic garnet Dopant in laser crystal Additive in special glass
Yttrium	High-temperature superconductor (YBaCuO) Incandescent lamp mantle Phosphor for cathode-ray tubes (reddish orange) Trichromatic fluorescent lighting phosphor (reddish orange) Metal halide lamp phosphor (reddish orange) High-temperature sensing phosphor Oxygen sensor ceramic Laser crystals (YAG, YLF) Gemstone simulants (CS and YAG) Microwave ferrite (YIG) Cast iron alloys and superalloys Magnesium castings Dispersion strengthened nickel and titanium alloys Aromatization of n-hexane Prostheses Stabilizer for silicon nitride and zirconia ceramics Microwave filter Acetylene lights Deoxidizer for vanadium and other nonferrous metals Solid oxide fuel cell electrolyte and anode Refractory crucibles and coatings High-temperature furnace electrodes

Table 3.9.1 Products Made From Rare Earth Elements^a (continued)

^a Hedrick, 1993.

.

...

Distribution	Percentage of Use
Catalysts in petroleum, chemical, and pollution control	34
Metallurgical uses as iron and steel additives and as alloys	27
Ceramics, glass polishing compounds, and glass additives	26
Miscellaneous uses in phosphors, electronics, permanent magnets, lighting, and research	13

Table 3.9.2 Distribution of Rare Earth Materials *

^a Hedrick, 1993.

•••

Nature of Ce Content in Material	Example	Commercial Application
Major component of mixed- lanthanide composition	Rare earth chloride, mischmetal	FCC catalysts, ^b Fe metallurgy
Minor component of mixed- lanthanide composition	Lanthanum concentrate, La-Ln chloride	FCC catalysts ^a
Dominant element in oxide-type composition	Cerium concentrate	Glass polishing, glass decolorizing
Relatively pure compound (>90%)	Oxide, nitrate, metal	Auto-emission catalysts
Highly pure compound (>99%)	Oxide, salts	Luminescence, catalysts

•••

-

^a Kilbourn, 1992. ^b Fluid Catalytic Cracking (FCC).

Operator Activity	Year	Annual Dose-Direct Irradiation (Beta/Gamma) (mrem) ^ь
Rare earth hydrate operator	1960 1959	455 210
Fluoride dryer-grinder	1960 1959	660 205
Cerium helper	1960 1959	760 500
Grinder operator	1960 1959	590 225
Bastnasite operator	1960 1959	535 125
Cerium operator	1960 1959	930 600
Cerium sifter	1960 1959	135 60
Rare earth fluoride operator	1960 1959	265 455
Rare earth hydrate dryer	1960 1959	580 160
lon exchange operator	1960 1959	200 605
Ion exchange operator	1960 1959	660 210
Ion exchange operator	1960 1959	815 465
Ion exchange operator	1960 1959	1,155 665
Ion exchange operator	1960 1959	785 870

Table 3.9.4 Annual Exposures to American Potash and Chemical CorporationPersonnel Working With Rare Earth Products ^a

__

^a McCoy, 1961. ^b 1 mrem = 0.01 mSv.

Rare Earth Material	Activities in Areas Sampled	% Weight of Thorium Content	% ML	Thorium Activity Concentration ^b (μCi/mL) ^c
<u>Glass polishes</u> bastnasite, cerium oxides, other products	Polishing materials are prepared, calcined, sifted and screened, and packed	<0.05–0.2	14	7×10 ⁻¹¹
Cerium hydrate	Drying in manually loaded and unloaded tray dryers and wet separation of cerium from rare earth mixture	0.18	7	4×10 ⁻¹¹
Rare earth oxide	Closed room with continuous rotary calciners and where material is screened and packed	0.22	24	1×10 ⁻¹⁰
Purified rare earths	lon exchange	<0.05%	11	6×10 ⁻¹¹
Rare earth sodium sulfate	Filter cake is dried in rotary dryer and handled in bulk	0.11	0	0
Rare earth chloride	NA ^d	<0.05	3.5	2×10 ⁻¹¹
Rare earth fluoride	Preparation, drying, pulverizing, and packing	0.22	11.5	6×10 ⁻¹¹

Table 3.9.5 Air Concentrations of Thorium in Rare Earth Processing Plant ^a

^a McCoy, 1961. ^b Estimated air concentration from % ML (maximum permissible level) as described in McCoy (1961). ^c 1 μ Ci/mL = 0.037 Bq/mL. ^d No applicable information.

Reporting Year	Monazite Production (kT/yr)	Number of Designated Workers	Number of PAS Measurements ^c	AA Index ^ь (Bq/m³)⁴
1986	10.9	266	485	0.85
1987	11.2	287	1,045	0.60
1988	10.6	301	1,509	0.51
1989	12.0	331	2,048	0.49
1990	8.5	287	1,458	0.27
1991°	6.9	194	1,204	0.16

Table 3.9.6 Airborne Radioactivity Levels Across All Mineral Sands Industry *

^a Hewson, 1993.

...

^b A representative industry AA index value for each site has been obtained by averaging the alpha activity concentration across each dry plant work category (i.e., operational, maintenance, and supervisory categories).

^c The number of personal air sampling (PAS) measurements relates to the number of samples used to derive the alpha activity (AA) index.

^d 1 pCi/m³ = 0.037 Bq/m³.

^e The AA index for 1991 also reflects significantly curtailed production capacity at most of the mineral sands sites.
Location ^a	Period of Evaluation ^a	Exposureª (mrem/168-h) ^ь	Estimated Average Annual Effective Dose Equivalent (mrem) ^b
Roaster operator	2/1–2/29 3/1–3/31 4/1–4/30	2 1 8	44
Bastnasite packaging	2/1–2/29 3/1–3/31 4/1–4/30	3 0 6	36
Cerium dryer operator	2/1-2/29 3/1-3/31 4/1-4/30	4 4 8	64
Cerium packaging	2/1–2/29 3/1–3/31 4/1– 4/30	 3 11	84
Ce-96 operator	2/1–2/29 3/1–3/31 4/1–4/30	4 0 10	56
Ce-96 helper	2/1–2/29 3/1–3/31 4/1–4/30	2 6 10	72

Table 3.9.7 1992 Molycorp Personnel TLD Results

^a Bernhardt, 1996. ^b 1 mrem = 0.01 mSv.

Location	Period of Analyses	Absorbed Dose Rate (µrad/h)	Absorbed Dose for 40 Hours (mrad)	Estimated Annual Dose Equivalent ^b (mrem)
Cerium bagging	1/9–2/12/92 2/11–4/15/92	40 40	2 2	80 80
Solvent extraction area SX-1 raff cell SX-1 sump pump SX-1 feed tanks	1/9–2/12/92	550 100 420	22 4 17	1,100 200 840
SX–1 raff cell SX–1 sump pump SX–1 feed tanks	2/11-4/15/92	570 150 510	23 6 21	1,140 300 510
Bastnasite bagging and warehouse	1/9–2/12/92 2/11–4/15/92	 60	2	 120

Table 3.9.8 Molycorp Plant Area TLD Results *

^a Bernhardt, 1996. Absorbed dose rate and 40-hour absorbed dose data rounded when applicable.

^b Annual exposure duration is estimated to be 2,000 hours and assumes that 0.01 mGy (1 mrad) is equal to 0.01 mSv (1 mrem).

	Estimated Monthly Dose Equivalent				
Date	Administration Office (mrem) ^b	Guardhouse Background (mrem) ^b	Fenceline Near School (mrem) ^b		
10/95-1/96	7	8	8		
1/96-4/96	19	24	lost		
7/96–10/96	17	19	17		
Average	15	17	13		
Annual average ^c	180	200	160		

Table 3.9.9 Environmental TLD Measurements Associated With Molycorp Lead and Iron Filter Cake Project ^a

^a Bernhardt, 1996. Estimated monthly dose equivalents are based on full-time occupancy and include background radiation levels as illustrated by estimated monthly dose equivalent at the guardhouse.

^b 1 mrem = 0.01 mSv. ^c Estimated annual effective dose equivalent (mrem).

	Radium Concentration (pCi/g) ^b		tration
Sample	²²⁸ Ra	²²⁴ Ra	²²⁶ Ra
Cerium oxide product	113	110	11
Metal additive (9-11% cerium)	21	38	1
Slag pile from production of metal additive	104	12	14
Metal additive from first rejected load	9	7	1
Slag from first rejected load	310	29	38
Metal additive from second rejected load	1	2	
Slag from second rejected load	592	59	63
Off-site soil	0.6	0.6	0.8

Table 3.9.10 Radium Concentrations in Slag, Metal Additives, and Cerium Oxide Product ^a

^a New York State, 1994. ^b 1 pCi/g = 0.037 Bq/g.

Exposure Scenario	Annual Effective Dose Equivalent (mrem) ^b
ON-SIT	E INDIVIDUAL
Direct gamma	0.01
Dust Innalation	70
	70
OFF-SIT	
Direct gamma	0.004
Dust inhalation	5×10⁻⁵
Well water	3
Food contaminated by well water	5×10 ⁻⁴
Food contaminated by dust deposition	6×10 ⁻⁷
Total	3

Table 3.9.11 Individual Effective Dose Equivalents From Storage or Disposalof Rare Earth Byproducts (Monazite Based) a

^a EPA, RAE–9232/1–2. ^b 1 mrem = 0.01 mSv.

...

Table 3.9.12 Annual Collective Effective Dose Equivalent From Storage or Disposal of Rare Earth Byproducts (Monazite Based) ^a

Exposure Scenario	Annual Collective Effective Dose Equivalent ^b (person-rem) ^c
Exposure to resuspended particulates	0.03
River water contaminated by groundwater	0.02
River water contaminated by surface runoff	3
Total	3

^a EPA, RAE–9232/1–2.
^b The collective effective dose equivalent is due to 1 year of intake.
^c 1 person-rem = 0.01 person-Sv.

Material	²³² Th (pCi/g)⁵	²³⁸ U (pCi/g) ^b	Annual Amount Produced or Disposed ^c (MT)
Bastnasite ore	25	7	408,150
Bastnasite concentrate	85	4	20,090
Stabilized lead iron filter cake	32	1,150	NA°
Pond lead iron residue	42	1,540	NA°
Lead sulfide concentrate	5	74	590°
SX crud	15	760	113 ^d

Table 3.9.13 Average Concentration of Radionuclides in Selected Materials ^a

^a ENSR, 1996.

^b 1 pCi/g = 0.037 Bq/g.
^c Stabilized lead iron residue and pond lead residue are no longer produced as a result of operations. ^d Estimate based on the past 3 years of production of SX crud.

Exposure Scenario	Individual Annual Effective Dose Equivalent (mrem)°	Collective Effective Dose Equivalent ^b (person-rem) ^c
······································	WASTE COLLECTOR	
Direct gamma Dust inhalation Soil ingestion Total	0.5 0.1 0.06 0.7	2
<u> </u>	ON-SITE WORKER	
Direct gamma Dust inhalation Soil ingestion Total	0.06 0.02 0.01 0.09	2
	ON-SITE INDIVIDUAL ^d	
Direct gamma Inhalation ^e Soil ingestion Total	<0.001 <0.001 <0.001 <0.001	420
	OFF-SITE INDIVIDUAL	
Dust inhalation	<0.001	<0.001
Well water ^d	<0.001	

Table 3.9.14 Summary of Annual Effective Dose Equivalent Due to Landfill Disposal of Cerium Concentrate (0.25% by Weight of Source Material as Thorium) ^a

^a Refer to the generic disposal methodology in Appendix A.2.

^b To estimate the annual collective effective dose equivalent (EDE), the following populations were considered: For the waste collector, there is assumed to be 1 collector for each of 3,500 landfills. The landfill worker population is 17,500 (5 workers per landfill multiplied by 3,500 landfills); the on-site population is 34,000; the off-site population associated with atmospheric releases is 1 million people within an 80-km radius of each landfill; and the off-site population associated with ground water emissions is 700,000.

^c 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

^d For the on-site individual, the annual individual and collective EDE is estimated by assuming the disposal of 270 kg of cerium concentrate per year per landfill, which is accumulated for 30 years until landfill closure and residents begin to live on-site 30 years after landfill closure. For the off-site individual EDE it is assumed that 270 kg of cerium concentrate is disposed in 1 year per landfill and the travel time to the groundwater well 100 meters from the landfill boundary is about 1.5×10^5 years.

^e Annual individual and collective EDE for inhalation exposure pathway takes into account ²²⁰Rn and its short-lived decay products during indoor residence.

Exposure Scenario	Individual Annual Effective Dose Equivalent (mrem) ^a	Collective Effective Dose Equivalent (person-rem)ª
	DISTRIBUTION AND TRANSPORT	
Driver Cerium and bastnasite concentrates	20	0.1
<u>Warehouse worker</u> Bastnasite concentrate Cerium concentrate	70 100	10 20
	ROUTINE OPERATIONS	
Industrial workers Bastnasite concentrate [°] Cerium concentrate [°]	1000 3000	100 300
Other members of the public ^d Pink ophthalmic glass (photon irradiation)	0.1	10
Television faceplates Personal computer faceplates	<0.001 0.003 0.006	200 200
	DISPOSAI °	6
Landfill		
Waste collector On-site worker On-site individual	0.7 0.09 <0.001	2 2 400
-Dust inhalation -Well water	<0.001 <0.001	<0.001
Slag worker	5	NA ⁹
	ACCIDENTS	
Warehouse fire Firefighter Cleanup personnel ¹	5 2	NA NA
<u>Transportation fire</u> Firefighter Cleanup personnel ^f	30 20	

Table 3.9.15Summary of Doses to Industrial Workers and Other Members of the PublicFrom the Handling and Use of Rare Earth Products Containing Source Material

See following page for footnotes.

--..

Footnotes to Table 3.9.15

^a 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

^b The truck driver is assumed to make 30 trips across the United States per year and there are 200 shipments annually. For the warehouse worker, it is assumed that each worker spends 500 h/yr in the warehouse, there are two workers per warehouse, and there are about 100 facilities in the United States storing bastnasite or cerium concentrates.

^c For the industrial worker, the annual effective dose equivalent (EDE) assumes a respirable dust concentration of 3 mg/m³, no dust respiratory protection, and 2,000-hour exposure duration. For the industrial worker, the annual collective EDEs assume a worker population of 100.

^d The hypothetical populations used to estimate annual collective EDE for the public are as follows: 100,000 wearers of pink ophthalmic glass, 250 million color televisions, and 1 million drivers with solar-controlled glass (800,000 public and 200,000 commercial).

^e For the landfill operators, a worker population of 17,500 is assumed. For the off-site population used to estimate annual collective EDEs there are 500,000 people who live within 80 km of each municipal landfill (atmospheric releases), and 700,000 people drink from off-site wells.

^f Respiratory protection is also assumed for warehouse and transportation spill cleanup personnel. An assigned protection factor of 10 (APF=10) is assumed. ⁹ Not applicable.

3.10 Thorium in Personnel Neutron Dosimeters

3.10.1 Introduction

In 10 CFR 40.13(c)(1)(vii), any person who receives, possesses, uses, transfers, or acquires personnel neutron dosimeters containing thorium is exempted from licensing requirements for source material, provided each dosimeter does not contain more than 50 milligrams (mg) of thorium. This exemption was proposed on June 24, 1976 (41 FR 26032), and issued as a final rule on February 3, 1977 (42 FR 6579).

The first *Federal Register* notice cited above includes estimates obtained by the Nuclear Regulatory Commission (NRC) of the radiation dose to individuals. A complete description of this product as originally envisioned and a more detailed discussion of potential radiation doses to individuals resulting from its use are given in the environmental impact statement for this exemption (NUREG–0137).

In the 1980s, the personnel neutron dosimeters of choice used detectors made of CR39 (allyl diglycol carbonate) plastic for fast neutrons and so-called neutron albedo dosimeters for thermal and intermediate-energy neutrons (Griffith, 1988). The neutron albedo dosimeters use thermoluminescent detectors made of nonradioactive materials such as lithium fluoride (LiF), and the CR39 plastic detectors are damaged by neutron-recoil hydrogen ions (protons) produced in radiator foils made of nonradioactive plastic materials (Luszik-Bhadra et al., 1990).

This product is not believed to be currently manufactured or in use. Past distribution, if any, is not known.

3.10.2 Description of Items

The personnel neutron dosimeters containing thorium are simple in construction and have no moving parts. A dosimeter consists basically of three components: (1) a radiator with a thorium-containing foil or coating; (2) a detector that is a suitable medium for recording the damage tracks due to the reaction products from the radiator; and (3) a case that encloses the detector and radiator. The case is designed for attachment to the clothing of the dosimeter wearer.

In the dosimeter design used as a basis for the assessment (NUREG–0137), the radiator consisted of a 5-mg/cm^2 -thick (0.0008-cm–thick) film of thorium fluoride (ThF₄) coated, by vapor deposition, onto an aluminum substrate. This substrate was a 0.2-cm-thick disk with a radius of 1.8 cm, and each coating contained 50 mg of thorium that was assumed to have aged 20 years since purification. A single detector of polycarbonate plastic, about 5.0 cm in diameter by 0.0001 cm thick, contacted the radiator in each dosimeter. Both the detector and radiator were enclosed in an aluminum case with a wall thickness of 0.2 cm, and the case was designed so that the portion covering a detector could be removed during detector replacement.

When fast neutrons with energies greater than approximately 1.5 MeV impinge on a radiator of thorium, fission reactions occur that are characterized by the splitting of a thorium nucleus into at least two very energetic nuclei (fission fragments) and by the release of a relatively large amount of energy (about 200 MeV). These very energetic fission fragments will produce

damage tracks in the detector that can be counted after appropriate chemical etching of the detectors (Tommasino and Harrison, 1985). The number of tracks in the detector can be counted visually, by automatic optical scanning, or by electric spark counting techniques.

The personnel neutron dosimeters that contain thorium would be issued to and worn by persons who work in or visit facilities where they may be exposed to fast neutrons (NUREG-0137). These facilities might include nuclear power plants, research reactors, particle accelerators, military installations, nuclear research laboratories, and other establishments containing neutron-producing equipment or neutron-emitting materials.

3.10.3 Summary of Previous Analyses and Assessments

A variety of routine, accident, and misuse exposure scenarios have been considered previously by the NRC (NUREG–0137). Scenarios for routine aspects of the life cycle for personnel dosimeters include:

- Exposures to radiation workers and members of the public during distribution and transport.
- Exposures to radiation workers while wearing dosimeters or changing detectors used in dosimeters.
- Exposures to a wearer's family or members of the public in the vicinity of a dosimeter wearer.
- Exposures to disposal workers at landfills.

The exposure scenarios involving accidents or misuse of personnel dosimeters that have been considered include:

- Release of thorium in a fire.
- Release of thorium to water following accidental dumping of dosimeters into a reservoir used to supply drinking water.

For exposure scenarios involving routine aspects of the life cycle of the neutron dosimeters, it was assumed that ingestion or inhalation of thorium would not occur normally and external irradiation would be the only significant exposure pathway. However, the previous assessment used an older dosimetry system (International Commission on Radiation Protection (ICRP) 2) and simply reported radiation doses to a number of critical organs, including the "total body." Because the radiation doses were from external irradiation involving high-energy photons, it is possible to equate the "total body" doses to effective dose equivalent (EDE) (ICRP 26). Thus, the dose estimates for routine exposure scenarios obtained by the NRC (NUREG–0137), based on a thorium content at the exemption limit of 50 mg, an annual distribution of 75,000 dosimeters, and a useful lifetime of 4 years for the dosimeters, are summarized as follows:

- Distribution and transportation workers would receive annual EDEs in the range of 2×10⁻⁶ to 0.002 millisievert (mSv) (2×10⁻⁴ to 0.2 mrem), and the annual collective EDE would be 2×10⁻⁵ person-Sv (2×10⁻³ person-rem).
- Radiation workers, including wearers of dosimeters and personnel changing detectors, would receive annual EDEs of 2×10⁻⁶ to 0.013 mSv (2×10⁻⁴ to 1.3 mrem), and the annual collective EDE would be 2 person-Sv (200 person-rem).
- Members of families of dosimeter wearers and other members of the public would receive an annual EDE of 0.008 mSv (0.8 mrem) or less, and the annual collective EDE would be 0.008 person-Sv (0.8 person-rem).
- Disposal workers at a landfill, handling an estimated 75 dosimeters per year, would receive annual EDEs of about 2×10⁻⁶ mSv (2×10⁻⁶ mrem), and the annual collective EDE would be about 2×10⁻⁶ person-Sv (2×10⁻⁴ person-rem).

Dose estimates obtained by the NRC (NUREG-0137) for the accident and misuse scenarios are summarized as follows:

- If a fire involving a shipping carton of 200 dosimeters is assumed to discharge all of the thorium uniformly in a moderately sized storage room, an individual breathing the contaminated air for 1 hour could receive radiation doses of 5 mSv (0.5 rem) to the total body, 200 mSv (20 rem) to the bone, and 3 mSv (3 rem) to the lungs. However, actual doses for the postulated accident probably would be only small fractions of these values because of such factors as ventilation of the room, accumulation of the dispersed thorium near the storage carton for the dosimeters, rapid deposition of airborne thorium onto the floor, and use of respiratory protection equipment by the individual entering the room. More importantly, it is highly improbable that a fire would be sufficiently hot to vaporize all of the thorium.
- If a shipping carton containing 200 neutron dosimeters is dumped into a reservoir and all of the thorium is dispersed uniformly in the water, the radiation doses to the individuals drinking the water would be about 7×10⁻⁶ mSv (7×10⁻⁴ mrem) to the total body and 6×10⁻⁵ mSv (0.006 mrem) to the bone. Because such an accident is highly unlikely, it is eliminated from further consideration in this report.

3.10.4 Present Exemption Analysis

This hypothetical assessment is based on the previous NRC assumptions (NUREG–0137) of a useful lifetime of 4 years for the dosimeters, based on normal wear and tear on the case of the dosimeters, and an annual distribution of 75,000 dosimeters, with each dosimeter containing the maximum allowed 50 mg of natural thorium. Thus, the total amount of natural thorium distributed annually in the 75,000 dosimeters would be 3.75 kg. Collective doses were not determined as this product is not believed to be currently manufactured or in use.

3.10.4.1 Distribution and Transport

Shipments would contain the number of dosimeters ordered by a customer, probably between 20 and 1000 dosimeters. Hence, the NRC (NUREG–0137) assumed a typical shipment might contain 200 dosimeters, and 375 such shipments per year would be required for an assumed annual distribution of 75,000 dosimeters.

For this assessment, it is assumed that the dosimeters are shipped primarily by a parcel-delivery system, and a local parcel-delivery driver picks up the dosimeters and takes them to a local terminal for shipment to customers. It is further assumed that semi-trucks are used to transport the dosimeters between local terminals, and the dosimeters pass through an average of four regional terminals before reaching their final destination. Radiation doses to workers at both local and regional terminals are assumed to be the same as those estimated for workers at a large warehouse (see Appendix A.3).

Based on the above assumptions and the generic methodology of Appendix A.3, the individual receiving the largest radiation dose is the local driver who is assumed to pick up 1 or 2 shipments each day from the same supplier for a total of 375 shipments during a year. The annual EDE to this individual is estimated to be 0.005 mSv (0.5 mrem). Individual doses to other drivers, terminal workers, and members of the public along the truck routes would be less. The maximum EDE would be five times this value to accommodate the maximum number of dosimeters in a shipment.

3.10.4.2 Routine Use

Persons touching or wearing a dosimeter could be exposed to both beta particles and photons from decay of the natural thorium contained in the dosimeters (NUREG–0137). Beta particles will penetrate the aluminum case (approximately 26% of the beta particles have energies greater than 1.26 MeV), and the beta-particle dose rate was calculated to be 6.5×10^{-4} mSv/h (0.065 mrem/h) at the center of the front surface to the aluminum case of a dosimeter (NUREG–0137). Photons from decay of natural thorium account for most of the external dose from the dosimeters, and the photon dose rate at 1 meter from the dosimeter was calculated to be 5.5×10^{-8} mSv/h (5.5×10^{-6} mrem/h) (NUREG–0137). The photon attenuation was neglected in this calculation. The natural thorium was assumed to be 20 years old so that the decay chain was about 87% of equilibrium.

3.10.4.2.1 Dosimeter Wearers

The NRC (NUREG–0137) considered individual doses from wearing or touching dosimeters for three different groups: (1) dosimeter wearers, (2) members of the wearer's family, and (3) other persons who might be near dosimeter wearers. Results of these previous NRC considerations can be summarized as follows.

Wearers are assumed to carry two dosimeters attached to their clothing over the chest for 2,000 h/yr and to touch a dosimeter with their hands for 1 min/day (4 h/yr). Such a person, could receive an annual dose equivalent to the hands of 0.004 mSv (0.4 mrem) from both beta particles and photons. The annual EDE to dosimeter wearers from whole-body irradiation by photons could be 0.009 mSv (0.9 mrem).

Members of the wearers' families (four other members assumed) might be exposed for 12 h/day and 365 days/yr (4380 h/yr) to a dosimeter stored at an average distance of 6 meters from each member. Each family member might also handle a dosimeter for a total of 0.7 h/yr, and thereby receive an annual dose equivalent to the hands of 7×10^{-4} mSv (0.07 mrem) from both beta particles and photons. The annual EDE to a family member from whole-body irradiation by photons could be less than 1×10^{-5} mSv (<0.001mrem).

Other persons who might be near dosimeter wearers (e.g., in restaurants, public buildings, and transportation vehicles) could be exposed to the photons from the dosimeters. If such a person should be exposed daily to a single dosimeter for 1 hour and if the dosimeter and person were separated at an average of 3 meters, the person could receive an annual EDE from whole-body irradiation by photons of less than 1×10^{-5} mSv (<0.001mrem).

3.10.4.2.2 Detector Changers

The NRC (NUREG–0137) assumed that each customer assigned a trained employee (e.g., a health physics technician) the task of replacing periodically the detectors in used dosimeters. The detector changers were assumed to handle 400 dosimeters per month for 5 seconds with the case closed and 3 seconds with the case open. This monthly activity would require the use of 375 detector changers to service the 150,000 dosimeter wearers. The changer also is assumed to be exposed at 4.5 meters from an average of 200 dosimeters for 2000 h/yr. Such exposures could result in an annual dose equivalent to the hands of 0.2 mSv (20 mrem) from both beta particles and photons. The annual EDE to a detector changer from whole-body irradiation by photons could be 8×10^{-4} mSv (0.08 mrem).

Because opening of the dosimeter cases is assumed during detector replacement, any thorium that might be detached from the radiators could be released to work areas and either ingested or inhaled (NUREG–0137). The likelihood of such releases is considered small, because (1) ThF₄ is a relatively stable compound that should not oxidize or undergo destructive chemical changes during normal conditions of use, (2) radiators should not be subjected to destructive mechanical stresses in dosimeter cases, and (3) any significant detachment of ThF₄ from a radiator should be visible to and recognized by a trained worker who would initiate procedures to prevent the spread of any detached thorium. Hence, potential releases of thorium during detector replacement are considered unimportant if trained workers are used for detector replacement, and the potential radiation dose from ingestion of detached thorium on a radiator is shown to be negligibly small in Section 3.10.4.4 of this report.

3.10.4.3 Disposal

To estimate potential individual and collective doses to the public from disposal of neutron dosimeters containing thorium in landfills and by incineration, the generic disposal methodology in Appendix A.2 was used. The amount of natural thorium in 1 year's distribution of 75,000 personnel neutron dosimeters is 3.75 kg, and it is in a form that is not readily dispersible in air or readily accessible to infiltrating groundwater at landfills. Thus, reduction by a factor of 10 was assumed in the following dose-to-source ratios for inhalation and ingestion in Appendix A.2: (1) waste collectors at both landfills and incinerators, (2) workers at landfills, (3) off-site members of the public exposed to airborne releases during landfill operations and releases to groundwater following disposal in landfills, and (4) future on-site residents at landfills.

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

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

3.10.4.4 Accidents and Misuse

In the case of misuse, exposure to a person who finds and vandalizes a personnel neutron dosimeter containing 50 mg of natural thorium is considered. It is assumed that the person deliberately destroys the dosimeter out of curiosity and handles the thorium radiator foil in the dosimeter for 30 minutes before discarding both the radiator foil and other dosimeter parts. For the skin dose from beta particles from the radiator foil, a dose rate of about 0.054 mSv/h (5.4 mrem/h) was calculated using VARSKIN MOD2 (Computer Codes, Durham, 1992). Thus, the dose equivalent to the hands from touching the active surface of the thorium radiator for 30 minutes could be 0.03 mSv (3 mrem). For potential ingestion of material due to touching of the active surface of the thorium radiator, the generic accident methodology developed in Appendix A.1 for spills of radioactive materials in the form of a powder was used. It was assumed, first, that 10% of the loose material on the radiator is deposited on the body and, second, that 0.1% of this deposited material is ingested before it is removed from the body by washing. Thus, the estimated EDE from ingestion of material from the natural thorium radiator could be 3×10^{-5} mSv (0.003 mrem).

In the case of accidents, the following scenarios are considered: (1) a residential fire involving a single dosimeter, and (2) a transportation fire involving an typical shipment of 200 dosimeters. A release fraction of 0.1% is assumed for the 50 mg of natural thorium contained in each dosimeter. Based on these assumptions and the generic accident methodology in Appendix A.1, the estimates of individual dose from fires involving dosimeters containing thorium are summarized as follows:

- For a person trying to escape from a residential fire or a neighbor trying to rescue a
 person from a residential fire, the individual dose from a single dosimeter containing
 50 mg of natural thorium could be 4×10⁻⁵ mSv (0.004 mrem).
- For a firefighter wearing a respirator at a residential fire, the individual dose from a single dosimeter containing 50 mg of natural thorium could be less than 1×10⁻⁵ mSv (<0.001 mrem). The individual dose could be less than 1×10⁻⁵ mSv (<0.001 mrem) to a worker who is involved in the cleanup following the fire and who does not wear a respirator.
- For a firefighter wearing a respirator at a transportation fire, the individual dose from 200 dosimeters containing a total of 10 g of natural thorium could be 3×10⁻⁵ mSv (0.003 mrem). The individual dose could be 2×10⁻⁴ mSv (0.02 mrem) to a worker who is involved in the cleanup following the fire and who does not wear a respirator.

3.10.5 Summary

Table 3.10.1 presents the results of this hypothetical assessment of radiation doses to the public from the distribution, routine use, and disposal of personnel neutron dosimeters containing thorium. These results are based on an assumed 1 year's distribution of 75,000 neutron dosimeters with an exemption limit of 50 mg of natural thorium each. For routine use of the dosimeters, including distribution and disposal, the annual EDE to the most highly exposed individual (wearer of a dosimeter) is estimated to be 0.009 mSv (0.9 mrem). For accidents and misuse, the individual EDEs were estimated to be 2×10^{-4} mSv (0.02 mrem) or less, and these results suggest there is very little potential for significant exposure to members of the public from accidents or misuse involving the thorium-containing foils in the personnel neutron dosimeters.

An annual distribution of 75,000 personnel neutron dosimeters was assumed in the previous assessment by the NRC (NUREG–0137) and also used as a basis for this assessment. No demand has developed, and there is no known distribution or use.

Exposure Pathway	Individual Annual Effective Dose Equivalent (mrem)ª
Distribution and transport	0.5 ^b
Routine use Dosimeter wearers Detector changers	0.9 ^c 0.08 ^d
<u>Disposal</u> Landfills Incinerators	0.001 ^e 0.004 ^f
<u>Accidents or misuse</u> Fire Vandalism of dosimeter unit	0.02 ^g 0.003 ^h

Table 3.10.1 Potential Radiation Doses From Personnel Neutron Dosimeters Containing Thorium

a 1 mrem = 0.01 mSv.

^b Dose estimate applies to local parcel-delivery driver; dose estimates are considerably less for terminal workers, long-haul semi-truck drivers, and members of the public along truck routes (see Section 3.10.4.1).

^c Dose estimate applies to dosimeter wearer; dose estimates are considerably less for members of wearers' family and other casually exposed members of the public (see Section 3.10.4.2.1).

^d Dose estimate applies to whole-body irradiation of detector changer; dose estimate for hands of detector changer is 0.2 mrem/yr (20 rem/yr) (see Section 3.10.4.2.2).

^e Dose estimate applies to waste collector at landfill; dose estimates are less for workers at landfills, off-site members of the public, and future on-site residents at landfills (see Section 3.10.4.3).

^f Dose estimate applies to waste collector at incinerators; dose estimates are less for workers at incinerators and off-site members of the public (see Section 3.10.4.3).

⁹ Dose estimate applies to worker during cleanup following transportation fire; dose estimates are less for firefighters and others involved in transportation or residential fires (see Section 3.10.4.4).

^h Dose estimate applies to ingestion of natural thorium by a person who finds and destroys a personnel neutron dosimeter containing a thorium radiator foil; dose estimate for beta-particle dose to hands from touching active surface of the radiator foil is 0.03 mSv (3 mrem) (see Section 3.10.4.4).

3.11 Glazed Ceramic Tableware

3.11.1 Introduction

In 10 CFR 40.13(c)(2)(i), glazed ceramic tableware is exempted from licensing requirements for source material, provided that the glaze does not contain more than 20% by weight of source material. A general exemption for ceramic products was included in the original 10 CFR 40 published on March 20, 1947 (12 FR 1855). The present exemption for glazed ceramic tableware was proposed on September 7, 1960 (25 FR 8619), and issued as a final rule on January 14, 1961 (26 FR 284).

Neither the original 1947 *Federal Register* notice nor the later notices from 1960 and 1961 cited above provide information on radiological impacts on the public from use of glazed ceramic tableware. The later notices stated that the exemption would not result in an unreasonable hazard to life or property but do not present a supporting dose analysis.

Potential radiological impacts on members of the public from use of glazed ceramic tableware are associated with beta-particle irradiation during handling tableware, beta-particle and gamma-ray irradiation while near tableware, and ingestion of uranium leached into foodstuffs that contacted tableware. Older data on and analyses of exposure and dose rates near glazed ceramic objects and leaching of uranium from glazes by water and acidic solutions have been summarized by Buckley et al. (NUREG/CR–1775) and the National Council on Radiation Protection and Measurements (NCRP 95). These summarises and more recent data are presented in Section 3.11.3.

3.11.2 Description of Items

An item of glazed ceramic tableware is a piece of ceramic formed into a plate, dish, bowl, cup, or saucer and coated with a layer of glass-like material called glaze. The purposes of the glaze are to seal the ceramic piece and to provide color to the item. Uranium oxide (U_3O_8) or sodium uranate (Na_2UO_2) has been added to glazes to produce a variety of attractive colors (e.g., red, orange, yellow, blue, and chocolate brown). Although such glazes may contain as much as 20% by weight of uranium, the highest percentage reported is 14%. (Ceramic glazes normally contain uranium, not thorium.)

Glazes are applied to the surface of a fired ceramic piece by brushing on or dipping into liquefied glazing material. The coated piece is then fired at a high temperature (e.g., 1430°C) to set the glaze. The resulting product is glass-like in appearance but is not as resistant as glass to environmental stresses. The quantities of uranium-containing glazes applied to various objects are not well documented, but Buckley et al. (NUREG/CR–1775) estimated a plate to contain 4.5 g of uranium, and Piesch et al. (1986) reported a red-glaze thickness of 0.2 mm.

Uranium-containing glazes were applied to fine china and earthenware, items used in homes, as overglazes, which are less resistant to wear and leaching than are the color-containing underglazes used on commercial tableware. The use of uranium-containing glaze on porcelain enamel and in pottery products was rare.

Domestic production of glazed ceramic tableware containing uranium apparently ceased about 20 years ago. The number of pieces of such tableware still in use is unknown. Buckley et al. (NUREG/CR–1775) estimated production of at least 2 million pieces of dinnerware between 1959 and 1969, approximately 200,000 pieces per year. Although possibly still used for dining purposes, pieces of such tableware are now considered collectible items (Landa and Councell, 1992) and likely would be displayed as such. The use of uranium in ceramic-tableware glazes ceased primarily because of the availability of cheaper substitutes, such as mixtures of antimony and iron, that yield acceptable glazes. Other factors that may have influenced the use of uranium in glazes are: (1) publicity regarding the presence of radioactive materials in the tableware probably affected sales adversely, and (2) as revealed by a general review of commercial consumer-product literature, manufacturers are striving to remove from their products all materials that are perceived to be hazardous. The above considerations, however, do not preclude the possibility that uranium is present in glazes on imported products.

3.11.3 Summary of Previous Analyses and Assessments

Previous studies were concerned with measurement of exposure or dose rates near glazed ceramic objects and determination of rates at which uranium leaches into various liquids. Dose assessments largely were restricted to doses from handling tableware and from ingesting acidic liquids that were in contact with the glazes.

3.11.3.1 External Exposures

Menczer (1965), using film badges, measured absorbed dose rates of between 5 and 200 microgray (μ Gy)/h (0.5 and 20 mrad/h) at the surfaces of various items of glazed ceramic tableware due to gross beta and gamma radiation. He estimated that a person who handles tableware for 1.5 h/day could receive a dose equivalent to the hands in the range of 0.02 to 0.1 sievert (Sv)/yr (2 to 10 rem/yr). The assumed handling time, 1.5 h/day, likely is an upper limit value for most persons.

Simpson and Shuman (NUREG/CP-0001) reported exposure rates from a "typical place setting" of uranium-containing dishes to be 4 microcoulomb (μ C)/kg (15 milliroentgen (mR)) (130 μ Gy or 13 mrad) per hour at contact or 0.8 μ C/kg (3 mR) (30 μ Gy or 3 mrad) per hour at 2.5 cm. They estimated that a person seated before this place setting could receive 0.8 μ C/kg (3 mR) (30 μ Gy or 3 mrad) per hour to the hands and arms and approximately 0.08 μ C/kg (0.3 mR) (3 μ Gy or 0.3 mrad) per hour to the torso. They also report measured exposure rates of between 1 and 2 μ C/kg (5 and 7 mR) (40 and 60 μ Gy or 4 and 6 mrad) per hour at contact with 0.2 μ C/kg (0.7 mR) (6 μ Gy or 0.6 mrad) per hour at 25 cm from Fiestaware[®].

Simpson and Shuman (NUREG/CP–0001) also used a beta-particle point-source formulation to calculate an absorbed dose rate of 0.024 Gy/yr (2.4 rad/yr) from a 45.6 day/yr (3 h/day) exposure at 7.6 cm from a glazed ceramic surface in which all uranium daughters are in radioactive equilibrium. This estimate likely is high by a factor of about 2 because all uranium daughters would not achieve radioactive equilibrium during the useful life of a piece of tableware. Also, a 3 h/day exposure at 7.6 cm is unrealistically conservative.

Buckley et al. (NUREG/CR-1775) cite gamma-ray exposure-rate measurements taken near various ceramic products by the Bureau of Radiological Health. For plates, exposure rates

were between 0.1 and 4 μ C/kg (0.5 and 15 mR) (4 and 130 μ Gy or 0.4 and 13 mrad) per hour at the surface and between 0.5 and 77 nC/kg (0.002 and 0.3 mR) (or 0.02 and 3 μ Gy or 0.002 and 0.3 mrad) per hour at 1 meter.

Piesch et al. (1986) measured a dose rate of 0.32 mSv (32 mrem) per hour at the surface of a teacup covered with a red glaze. Assuming daily use of the teacup (an actual exposure time was not given), they estimated an individual could receive 4 mSv (400 mrem) to the lips and 12 mSv (1200 mrem) to the fingers.

McCormick (1992) counted a variety of Fiestaware[®] items (plates, bowls, and pitchers) in an ion chamber. Reported exposure rates ranged between 0.5 and 3 μ C/kg (2 and 10 mR) (20 and 90 μ Gy or 2 and 9 mrad) per hour.

3.11.3.2 Internal Exposures

Kendig and Schmidt (1972) contacted red-glazed plates with 4% acetic acid solutions for time periods between 24 and 60 hours. Uranium contents in the solutions were 8.6, 1.8, 8.2, and 8.6 ppm (or mg/L) after 24 hours of contact for two bowls, a saucer, and a plate, respectively. After 60 hours of contact, uranium contents in solution were 55 and 41 ppm for a saucer and plate, respectively.

Landa and Councell (1992) contacted uranium-bearing glazes on two objects (a plate and a gravy boat) with 20 mL of three different solutions: (1) water, (2) 4% acetic acid, and (3) 1 molar nitric acid. After 24 hours of contact, uranium concentrations in the solutions ranged from 3.9 to 10.6 μ g/L in water; 470 to 31,800 μ g/L in acetic acid; and 96,100 to 304,000 μ g/L in nitric acid.

Landa and Councell (1992) noted the possibility of a memory effect on the leaching process. (The amount of uranium leached from an item per period of contact will decrease as the number of contact periods increases.) Such an effect would be expected as the amount of uranium available for leaching decreases after each leaching operation. For example, 20 mL of 4% acetic acid at 31,800 μ g/L contains approximately 640 μ g of uranium. If the uranium content in 20 mL of acetic acid were removed from an item each day for 1 year, approximately 0.23 g of uranium would be removed from the item. If the uranium content of 0.5 L of acetic acid (e.g., 8.5 fluid ounces of orange juice) were removed from an item each day for 1 year, approximately 5.8 g of uranium would be removed. This exceeds or is a substantial portion of the total uranium content of many pieces of tableware (see Section 3.11.2 and Table 3.11.1).

The above studies report neither the uranium concentrations in, nor the surface areas of, the glazes that were in contact with the leaching solutions. Normalizing the above results with respect to uranium concentration and contacted surface area could yield a better measurement for comparing leach rates from a variety of objects (i.e., micrograms of uranium per liter of leachate, per hour of contact, per square centimeter of glaze).

3.11.4 Present Exemption Analysis

3.11.4.1 General Information

Most pieces of glazed ceramic tableware are distributed in sets of four, six, or eight place settings. Other pieces that may be available separately include specialty items such as a platter, a gravy dish, and serving bowls. Individual pieces may be distributed as replacement items; however, the potential radiation doses associated with distribution of individual pieces are a small subset of the doses associated with distribution of place settings. Therefore, distribution of individual pieces is not evaluated in this section.

As noted in Section 3.11.2, the availability of tableware coated with uranium-bearing glazes cannot be quantified because domestic manufacture of such products ceased at least 20 years ago and because import of such tableware, if it occurs at all, could not be verified. The present analysis conservatively assumes an annual distribution similar to that reported in Buckley et al. (NUREG/CR–1775), 216,000 pieces of tableware. This tableware is assumed to be divided into 6000 sets, each of which contains 36 pieces in six place settings. Each place setting consists of a 25-cm- (10-inch-) diameter plate, a 20-cm- (8-inch-) diameter dish, a 25-cm- (10-inch-) diameter bowl, a 15-cm- (6-inch-) diameter bowl, a cup, and a 10-cm- (4-inch-) diameter saucer. Analyses in this section include: (1) individual and collective doses due to distribution of the tableware; (2) individual doses due to home use of a set of tableware; (3) individual and collective doses due to display of a place setting as a collectible; (4) individual and collective effective dose equivalents (EDEs) due to disposal; and (5) individual doses due to an unintended use of a plate as a radiography source.

Measurements of several pieces of tableware indicate that representative thicknesses of ceramic are between 0.64 and 1.3 cm for plates, dishes, saucers, and bottoms of bowls and cups and approximately 0.64 cm for the walls of cups and bowls. Piesch et al. (1986) report a glaze thickness of about 0.02 cm. A glaze thickness of 0.025 cm is used in the following analyses.

Geometries and materials used to represent the various pieces and arrays of tableware used in this assessment are listed in Table 3.11.1. Each item is represented by two cylinders of glaze separated by 0.64 cm of ceramic. The thicknesses of the glaze cylinders were obtained by dividing the total volume of glaze on an item by twice the top surface area of the cylinder. Arrays of tableware are represented by cylinders of a composite material, which is a homogeneous mixture of glaze and ceramic. Array representations were derived in a way that kept the exposed (projected) surface area constant while preserving the total mass of ceramic, glaze, and uranium in the items that make up the arrays (see Table 3.11.1).

The glaze is modeled as a material having the composition of Pyrex[®] glass plus 20% by weight of uranium, a density of 2.7 g/cm³, and an atomic number of 11.1. The ceramic is represented by a material with the composition of concrete and a density of 1.2 g/cm³. The composite material is modeled as a material having a density of 1.3 g/cm³ and an atomic number of 9.7.

3.11.4.1.1 External Exposures

External exposures to glazed ceramic tableware involve exposures to both beta particles and gamma rays. Beta-particle exposure is especially important during handling of tableware.

Absorbed dose rates in tissue due to beta-particle irradiation were calculated using VARSKIN MOD 2 (Computer Codes, Durham, 1992). Dose rates were calculated at a depth of 7 mg/cm² in tissue from contact with and from exposures at selected distances from a 25-cm-diameter glazed ceramic disk. Dose rates, which vary between 240 μ Gy (24 mrad) (0.24 mSv or 24 mrem) per hour at contact and 0 μ Gy (0 mrad) per hour at 760 cm, are listed in Table 3.11.2. Use of the 25-cm disk as a surrogate for all tableware geometries maximizes the calculated doses, except doses from contact with the disk. (Dose rates from a 25-cm diameter disk are between two and three times higher than dose rates from a 15-cm diameter disk.)

The contribution of beta-particle irradiation to the EDE rate at a selected distance from a piece of glazed tableware is obtained by multiplying the calculated dose rate by the skin weighting factor (0.01) and the fraction of skin under irradiation. For handling a plate, the dose rate to the palms and fingers is 0.24 mSv/h (24 mrem/h). The skin of the palms and fingers constitutes about 1% of the total skin area of the body. Therefore, the contribution of beta-particle irradiation to the EDE rate is 2.4×10^{-5} mSv/h (0.0024 mrem/h). For exposures at less than 90 cm, one-fourth of the total skin area is assumed to be irradiated; for the remaining exposure distances, one-half of the total skin area is assumed to be irradiated. Resulting dose rates are presented in Table 3.11.2.

Dose rates due to gamma and bremsstrahlung irradiation were calculated using CONDOS II (Computer Codes, O'Donnell et al., 1981). Tables 3.11.3 and 3.11.4 are lists of calculated EDE rates at selected distances from the items described in Table 3.11.1.

3.11.4.1.2 Internal Exposures

Reported uranium concentrations for various contact durations (see Section 3.11.3.2) in water are 3.9 and 10.6 μ g/L/24-h of contact. Reported concentrations in 4% acetic acid are 470, 10,000, and 31,800 μ g/L/24-h; 55,000 μ g/L/50-h; and 105,000 μ g/L/60-h. Normalizing these reported concentrations with respect to their reported leach durations yields effective concentration rates of 0.2 and 0.4 μ g/L/h in water and 20, 420, 1325, 1100 and 1750 μ g/L/h in 4% acetic acid.

In the following assessment, maximum 24-hour contact concentrations, 10.6 μ g/L in water and 31,800 μ g/L in 4% acetic acid solutions, are used for liquids that contact glaze for 24 hours or more (e.g., liquids not consumed in one sitting). For liquids in contact with glaze for less than 24 hours, the normalized concentrations corresponding to the above are used, 0.4 μ g/L/h in water and 1325 μ g/L/h in acetic acid.

Based on EPA (EPA/600/P-95/002Fa) studies, a typical daily intake of liquids is approximately 2 L. This intake may include 1.4 L of water and water-based beverages, 0.11 L of fruit juices, 0.046 L of soups, and 0.55 L of food-derived liquids. These liquids are designated as water-like and acid-like as follows: water and water-based beverages, 100% (1.4 L) water-like; fruit juices, 100% (0.11 L) acid-like; soups, 75% (0.035 L) water-like and 25% (0.012 L) acid-like; and food-derived liquids, 75% (0.41 L) water-like and 25% (0.14 L) acid-like. Thus a typical

individual could ingest approximately 1.8 L/day (670 L/yr) of water-like liquids and 0.26 L/day (95 L/yr) of acid-like liquids.

The above liquids normally are consumed out of glasses, not out of ceramic tableware, and most contact times are much less than 24 hours. However, the following assumptions were used to bound the analysis:

- Water and water-based liquids are 100% (1.4 L/day) water-like; 75% (1.1 L/day) are consumed from glassware and 25% (0.35 L/day) from ceramic tableware; and 90% of the liquids are in contact with the container for less than 30 minutes and 10% for 24 hours or more. Thus, 0.79 L/day are in contact with glassware and 0.315 L/day are in contact with ceramic ware for 30 minutes, and 0.035 L/day are in contact with ceramic ware for 24 hours.
- Fruit juices are 100% (0.11 L/day) acid-like; 90% (0.099 L/day) are consumed from glassware and 10% (0.011 L/day) from ceramic tableware; and 100% of the liquids are in contact with the container for less than 30 minutes. Thus, 0.099 L/day are in contact with glassware and 0.011 L/day are in contact with ceramic ware for 30 minutes.
- Soups are 75% (0.035 L/day) water-like and 25% (0.012 L/day) acid-like; 100% are consumed from ceramic tableware; and 90% are in contact with the container for 30 minutes and 10% for 24 hours or more. Thus, 0.032 L/day of water-like and 0.0109 L/day of acid-like soups are in contact with ceramic ware for 30 minutes, and 0.0035 L/day of water-like and 0.0012 L/day of acid-like soups are in contact with ceramic ware for 24 hours.
- Food-derived liquids are 75% (0.41 L/day) water-like and 25% (0.14 L/day) acid-like; 100% are consumed from ceramic tableware; and 90% are in contact with the container for 30 minutes and 10% for 24 hours or more. Thus, 0.37 L/day of water-like and 0.126 L/day of acid-like food-derived liquids are in contact with ceramic ware for 30 minutes, and 0.041 L/day of water-like and 0.014 L/day of acid-like food-derived liquids are in contact with ceramic ware for 24 hours.

Thus, considering only liquids consumed from ceramic ware, an individual may consume about 0.72 L/day (260 L/yr) of water-like and 0.15 L/day (54 L/yr) of acid-like liquids that were in contact with glaze for 30 minutes plus 0.08 L/day (29 L/yr) of water-like and 0.015 L/day (5.5 L/yr) of acid-like liquids that were in contact with glaze for 24 hours or more. Based on these consumption rates and the leaching factors discussed above, an individual could ingest approximately 0.21 g of uranium during 1 year.¹¹ Thus, given an ingestion dose factor of 1.9×10^{-6} mSv/µg (1.9×10^{-4} mrem/µg), this individual could receive an annual EDE of about 0.4 mSv (40 mrem) from ingestion of uranium leached from glazed ceramic tableware.

The derivation of the leaching factors was made without correlation to the uranium concentration of the glaze. However, because of other assumptions regarding contact time and usage, the results are considered conservative for a maximum exposed individual. Unless

¹¹ (54 L of 30-min acidic × 1325 μ g/L/h × 0.5 h) + (260 L of 30-min water-like × 0.4 μ g/L/h × 0.5 h) + (5.5 L of 24-h acidic × 31,800 μ g/L) + (29 L of 24-h water-like × 11 μ g/L) = 2.1×10⁵ μ g.

glazed ceramic tableware is used as primary dinnerware, any actual dose would be substantially less.

3.11.4.2 Distribution

As noted above, domestic production of tableware coated with uranium-containing glazes has ceased and import of such tableware, if occurring, cannot be quantified. However, based on past production, this assessment assumes an annual distribution of 216,000 pieces of tableware in 6,000 cartons, each of which contains 36 pieces (six 6-piece place settings) of tableware. Each of 2 manufacturers is assumed to ship 3000 sets of tableware (cartons), of which 1000 are distributed through 10 wholesalers, 1000 through 20 large retailers, and 1000 through 50 small retailers.

The following model is used in this assessment. (See Table 3.11.5 for a list of the model steps, the number of facilities involved in each step, and the number of cartons handled at each facility.) During the year, all cartons (3000) from each of two manufacturers are loaded into a small express-delivery truck and transported to a parcel-delivery center. At each center, (1) 100 cartons are loaded into a large local-delivery truck for transport to a wholesaler; (2) 50 cartons are loaded into each of two large local-delivery trucks for transport to two large retail stores; (3) 20 cartons are loaded into each of five small express-delivery trucks for transport to five small retail stores; and (4) 900 cartons are loaded into each of three large regional-delivery trucks for transport to three intermediate truck terminals. At each intermediate truck terminal, (1) 100 cartons are loaded into one large local-delivery trucks for transport to a wholesaler;

(2) 50 cartons are loaded into each of two large local-delivery trucks for transport to two large retail stores; (3) 20 cartons are loaded into each of five small express-delivery trucks for transport to five small retail stores; and (4) 300 cartons are loaded into each of two large regional-delivery trucks for transport to two final regional truck terminals. At each final truck terminal, (1) 100 cartons are loaded into one large local-delivery trucks for transport to a wholesaler; (2) 50 cartons are loaded into each of two large local-delivery trucks for transport to two large retail stores; and (3) 20 cartons are loaded into each of five small express-delivery trucks for transport to two large retail stores; and (3) 20 cartons are loaded into each of five small express-delivery trucks for transport to five small retail stores.

The carton source representation given in Table 3.11.1 was used in conjunction with the exposure conditions and calculation methods given in Appendix A.3.3 to calculate individual and collective EDEs for each step in the model. Results of the calculations are presented in Table 3.11.5. The highest calculated individual annual EDE was approximately 0.08 mSv (8 mrem), to a worker in the initial parcel warehouse. The total collective EDE for distribution was about 0.5 person-Sv (50 person-rem), almost entirely due to exposure at retail establishments. Because of the number of cartons carried in each truck, the average truck driver exposure conditions were used in the calculations (see Appendix A.3.3.). These dose estimates are based on calculations using the maximum allowed concentration of 20% by weight. Using an average 14% by weight, the workers dose would be 0.06 mSv (6 mrem) and the collective EDE would be 0.3 person-Sv (30 person-rem).

3.11.4.3 Routine Use

3.11.4.3.1 Use as Dinnerware

Members of a household can be exposed to the uranium contained in glazed ceramic tableware both externally and internally. External exposures can occur while dining; moving tableware before and after dining, washing and storing tableware, and merely being in the home, especially in the kitchen where arrays of tableware are stored in cabinets. Internal exposure can occur via ingestion of foodstuffs in contact with the uranium-containing glaze. The following model, which is described partially in Table 3.11.6, the dose rates given in Tables 3.11.2 through 3.11.4, and the uranium concentration rates given in Section 3.11.4.1.2 are used to estimate EDEs and dose equivalents to skin. External exposure durations were based on studies of homemakers' activities (Steidl and Bratton, 1968; EPA/600/P–95/002Fa); exposure distances are based on observation. Liquid intakes are derived from the Exposure Factors Handbook (EPA/600/P–95/002Fa).

A person designated as a homemaker in Table 3.11.6 is assumed to wash all dishes and to spend the most time at home and in the kitchen. These activities could result in an EDE of about 0.07 mSv/yr (7 mrem/yr). Other family members could receive EDEs similar to the home maker. From Table 3.11.6, these persons could receive an EDE of about 0.06 mSv/yr (6 mrem/yr) from external exposures. However, the doses from external exposures are small with respect to the 0.4 mSv/yr (40 mrem/yr), that could be received from intakes of liquids that were in contact with uranium-bearing glazes (see Section 3.11.4.1.2).

3.11.4.3.2 On Display as Collectibles

Collectibles may be displayed in many ways, both in homes and in public facilities, such as museums and stores that sell collectibles. Display in a public facility is the largest contributor to collective dose.

Due to space limitations in a home display cabinet, one six-piece place setting is assumed to be on permanent display in a glass-enclosed cabinet that eliminates the beta dose component except during handling of tableware. It is assumed that an individual (1) handles individual pieces (25-cm-plate is assumed) from a distance of 30 cm for 6.1 h/yr, (2) views the display from a distance of 91 cm for an additional 10 h/yr, and (3) is in other rooms at average distances of 610 and 762 cm for 200 and 6,000 h/yr, respectively. This individual could receive an EDE of 6×10^{-4} mSv/yr (0.06 mrem/yr). This person also could receive an annual dose equivalent to the skin of the hands of about 1.4 mSv (140 mrem). Collective EDEs to a family of four, with only one individual handling the items, could be 1.7×10^{-6} person-Sv/yr (1.7×10^{-4} person-rem/yr) and 3.4×10^{-5} person-Sv (3.4×10^{-3} person-rem) over 20 years. If a place setting from all 6000 sets of tableware is put on display in 6000 homes, the 20-year collective EDE could be 0.2 person-Sv (20 person-rem).

An open display of six place settings in a public facility (e.g., museum, shop, etc.) is assumed to be viewed by 500 persons/day (0.18 million/yr). Each person is assumed to spend 5 minutes viewing the display from 91 cm away, 30 minutes at an average distance of 460 cm from the display, and 1 hour at an average distance of 760 cm from the display. Each person could receive an EDE of less than 1×10^{-5} mSv (<0.001 mrem). For purposes of estimating collective doses, the calculated value of 1.8×10^{-6} mSv (1.8×10^{-4} mrem) based a 14% by weight glaze has

been used rather than the less than value. Collective EDEs per display could be 3×10^{-4} person-Sv/yr (0.03 person-rem/yr), 0.006 person-Sv (0.6 person-rem) over 20 years. If 6,000 such displays were available, the 20-year collective EDE could be 40 person-Sv (4,000 person-rem).

3.11.4.4 Accidents and Misuse

3.11.4.4.1 Accidents

....

It is inconceivable that an accident involving ceramic tableware could have radiological consequences greater than those from routine distribution and use. Even in the event of a fire, a uranium-bearing glaze is unlikely to become airborne, because temperatures used to set glazes, approximately 1430°C, exceed temperatures normally achieved during a vehicular or structure fire. Thus, any radiological consequences of an accident involving glazed ceramic tableware would be due to handling the residue of the accident. Such handling is unlikely to result in an EDE greater than the 0.08 mSv (8 mrem) that could be received by a parcel-delivery terminal worker who handles all 6000 cartons in 1 year.

3.11.4.4.2 Misuse

Some pieces of glazed ceramic tableware have been used in a manner unintended by the exemption—as crude radiography sources. In such instances, a person could handle a plate for 1 minute (hands-on contact and body at 30 cm), be an average of 91 cm from the plate for 5 minutes while placing and removing the object to be radiographed on the plate and covering and uncovering both plate and object with undeveloped film, and be 610 cm from the plate for 8 hours. Such exposures could result in reception of about 0.004 mSv (0.4 mrem) to the skin of the hand and an EDE of less than 1×10^{-5} mSv (<0.001 mrem).

3.11.4.5 Disposal

Under normal circumstances, glazed ceramic tableware would be disposed of as ordinary, non-combustible household trash. The following assessment assumes discard of 1.8 Mg of uranium in 6000 sets of tableware in 1 year based on the average uranium mass per set presented in Table 3.11.1. This assumption omits consideration of random discard of individual pieces of tableware prior to the year of discard. Individual doses will be maximized if all pieces are disposed of at once, and collective doses will be affected little, if at all.

Using the assumptions of the generic disposal methodology (see Appendix A.2) for disposal of 2.5 Mg of natural uranium with 20% being incinerated, the highest calculated individual EDE is 0.004 mSv (0.4 mrem), to a waste collector at a municipal incinerator. No other individual is estimated to receive an EDE greater than 0.001 mSv (0.1 mrem). The total collective EDE to all workers and potentially exposed members of the public could be about 0.08 person-Sv (8 person-rem).

3.11.5 Summary

This assessment has considered potential doses to the public from use of uranium as a coloring agent in glazes applied to ceramic tableware. All direct exposure calculations were

based on the maximum concentration (20% by weight) of uranium allowed in the glazes and an assumption that the tableware is still manufactured. However, there is no evidence that this is occurring. The highest concentration reported in actual pieces of tableware is 14% by weight; the collective dose for dinnerware is based on weight percent. Internal doses were estimated based on the measured concentration of uranium in various liquids simulating different type foods. Results of the assessment are summarized in Table 3.11.7.

The highest potential EDEs, approximately 0.04 mSv/yr (40 mrem/yr), are to persons who might drink liquids that were in contact with and had leached uranium from the glazes. Although such leaching does occur, the amount of leaching assumed in this assessment likely is excessive for a 20-year useful life. No measurements were made of the uranium content of the glaze (relative to the limit of 20% by weight). However, because of other assumptions regarding usage, the estimated doses are considered conservative. Unless glazed ceramic tableware is used as primary dinnerware, any actual dose would be substantially less.

Calculated doses due to distant external exposures to beta particles also are overstated due to the exclusive use of a 25-cm-diameter disk to represent all tableware geometries. Using smaller source diameters could reduce external dose estimates due to dining by a factor of 3 or more.

Since currently domestically manufactured as decorative or ornamental objects only, the most likely exposure to an individual would be external from display. Therefore, the average individual annual EDE is estimated to be 6×10^{-4} mSv (0.06 mrem) for the 20% by weight limit and 4×10^{-4} mSv (0.04 mrem) at an average 14% by weight.

	Representation					
	Cylinder			Separator		
Item	Radius (cm)	Depth (cm)	Material	Uranium Mass ^a (g)	Thickness (cm)	Material
10" plate	13	0.026	Glaze	14	0.64	Ceramic
8" dish	10	0.027	Glaze	9.3	0.64	Ceramic
4" saucer	5.1	0.028	Glaze	2.5	0.64	Ceramic
3.5" cup	4.4	0.11	Glaze	7.7	0.64	Ceramic
10" bowl	13	0.046	Glaze	25	0.64	Ceramic
6" bowl	7.6	0.059	Glaze	12	0.64	Ceramic
6, 10" plates	5.8	20	Composite	87	0	None
6, 8"dishes	5.2	16	Composite	56	0	None
6, 4" saucers	3.7	7.9	Composite	15	0	None
2 cups	5.0	4.7	Composite	15	0	None
6, 10" bowls	8.3	6.6	Composite	150	0	None
6, 6" bowls	6.4	13	Composite	70	0	None
Carton	12	23	Composite	424	0	None

Table 3.11.1 Source Representations of Ceramic Tableware

•••.

^a Based on 20% by weight glaze. Multiply by 0.7 for 14% by weight average.

Distance (cm)	Absorbed Dose Rate (mrad/h)ª	Fraction of Skin Exposed	Contribution to Effective Dose Equivalent (mrem/h) ^a
0	24	0.01	0.0024
30	0.84	0.25	0.0021
40	0.49	0.25	0.0012
91	0.089	0.50	0.00045
183	0.017	0.50	8.5×10⁻⁵
457	0.00040	0.50	2.0×10 ⁻⁶
610	0.000013	0.50	6.6×10⁻ ⁸
762	0	0.50	0

Table 3.11.2 Absorbed Beta Dose Rates at 7 mg/cm² in Tissue at Selected DistancesFrom the Surface of a 25-cm- (10-in-) Diameter Disk of 0.025-cm-Thick Ceramic GlazeContaining 20% by Weight of 20-Year-Old Natural Uranium

^a Assume 1 mrad/h = 1 mrem/h. 1 mrem/h= 0.01 mSv/h.

Distance	Dose Equivalent Rate (rem/h)ª							
in Air (cm)	Plate 13 cm (10")	Dish 10 cm (8")	Saucer 5.1 cm (4")	Cup 4.4 cm (3.5")	Bowl 13 cm (10")	Bowl 7.6 cm (6")		
	1.8×10⁻⁵	1.6×10⁻⁵	1.2×10⁻⁵	4.1×10⁻⁵	3.0×10⁻⁵	3.1×10⁻⁵		
30	6.5×10 ⁻⁷	4.4×10⁻ ⁷	1.2×10⁻ ⁷	3.7×10⁻ ⁷	1.1×10⁻ ⁶	5.5×10⁻ ⁷		
40	3.8×10⁻ ⁷	2.5×10⁻ ⁷	6.8×10 ⁻⁸	2.1×10 ⁻⁷	6.6×10⁻ ⁷	3.2×10⁻ ⁷		
91	7.7×10 ⁻⁸	5.1×10⁻ ⁸	1.3×10⁻ ⁸	4.1×10 ⁻⁸	1.3×10⁻ ⁷	6.2×10⁻ ⁸		
183	1.9×10⁻ ⁸	1.3×10 ⁻⁸	3.3×10⁻⁰	1.1×10⁻ ⁸	3.3×10⁻ ⁸	1.5×10⁻ ⁸		
457	3.0×10 ⁻⁹	2.0×10⁻⁰	5.2×10 ⁻¹⁰	1.6×10 ⁻⁹	5.3×10⁻ ⁹	2.5×10⁻⁰		
610	1.7×10 ⁻⁹	1.1×10 ⁻⁹	2.9×10 ⁻¹⁰	9.0×10 ⁻¹⁰	3.0×10⁻ ⁹	1.4×10⁻ ⁹		
762	1.1×10 ⁻⁹	7.1×10 ⁻¹⁰	1.9×10 ⁻¹⁰	5.7×10 ⁻¹⁰	1.9×10 ⁻⁹	8.7×10 ⁻¹⁰		

Table 3.11.3 Effective Dose Equivalent Rates Due to Photon Irradiation at SelectedDistances From Pieces of Ceramic Tableware Coated With GlazeContaining 20% by Weight of 20-Year-Old Natural Uranium

^a 1 rem/h = 0.01 Sv/h.

Distance			Dose E	quivalent (rem/h)ª	Rate		
in Air (cm)	6 Plates 13 cm (10")	6 Dishes 10 cm (8")	6 Saucers 5.1 cm (4")	3 Stacks 2 Cups	6 Bowls 13 cm (10")	6 Bowls 7.6 cm (6")	Carton
1	8.6×10⁻⁵	7.7×10⁻⁵	4.6×10 ⁻⁵	4.1×10⁻⁵	9.2×10 ⁻⁵	7.7×10⁻⁵	1.1×10 ⁻⁴
30	1.8×10⁻ ⁶	1.3×10⁻⁵	4.5×10⁻ ⁷	5.9×10⁻ ⁷	3.5×10⁻⁵	1.9×10⁻ ⁶	7.7×10 ⁻⁶
40	1.1×10⁻ ⁶	8.0×10⁻ ⁷	2.7×10⁻ ⁷	3.4×10 ⁻⁷	2.1×10⁻ ⁶	1.1×10 ⁻⁶	4.8×10 ⁻⁶
91	2.5×10 ⁻⁷	1.8×10⁻ ⁷	5.6×10 ⁻⁸	7.1×10⁻ ⁸	4.7×10⁻ ⁷	2.4×10⁻ ⁷	1.1×10 ⁻⁶
457	1.1×10⁻ ⁸	7.8×10 ⁻⁹	2.4×10 ⁻⁹	2.9×10⁻⁰	2.1×10 ⁻⁸	1.1×10 ⁻⁸	5.1×10 ⁻⁸
610	6.3×10⁻⁰	4.4×10⁻ ⁹	1.3×10⁻ ⁹	1.6×10 ⁻⁹	1.2×10 ⁻⁸	6.0×10 ⁻⁹	2.9×10 ⁻⁸
762	4.0×10 ⁻⁹	2.8×10 ⁻⁹	8.4×10 ⁻¹⁰	1.0×10 ⁻⁹	7.6×10 ⁻⁹	3.8×10 ⁻⁹	1.9×10 ⁻⁸

Table 3.11.4 Effective Dose Equivalent Rates Due to Photon Irradiation at SelectedDistances From Arrays of Ceramic Tableware Coated With GlazeContaining 20% by Weight of 20-Year-Old Natural Uranium

^a 1 rem/h = 0.01 Sv/h.

				and the second	
Step	Representation	Cartons per Facility	Number of Facilities	Highest Individual Effective Dose Equivalent (mrem) ^b	Collective Effective Dose Equivalent (person-rem) ^b
To parcel delivery center	Express delivery, small truck	3,000	2	2.3	0.0050
At parcel delivery center	Medium warehouse	3,000	2	7.5	0.087
To regional truck center 1	Regional delivery, large truck	900	6	0.19	0.0013
At regional truck center 1	Large warehouse	900	6	0.53	0.024
To regional truck center 2	Regional delivery, large truck	300	12	0.062	0.00084
At regional truck center 2	Large warehouse	300	12	0.18	0.016
To wholesaler	Local delivery, large truck	100	20	0.14	0.0031
At wholesaler	Large store	100	20	4.0	13
To large store	Local delivery, large truck	50	40	0.070	0.0031
At large store	Large store	50	40	2.0	13
To small store	Local delivery, small truck	20	100	0.034	0.00041
At small store	Small store	20	100	1.0	22
Total:					50

Table 3.11.5 Summary of the Distribution Model and Potential Individual andCollective Effective Dose Equivalents a

^a Based on 20% by weight limit. Multiply by 0.7 for doses using average 14% by weight. ^b 1 mrem = 0.01 mSv; 1 person-rem=0.01 person-Sv.

Person	Activity	Duration	Distance	Course	Effective Dose Equivalent
Feison	Activity	(10/91)	(ciii)	Source	(mrem/yr)-
Home maker	In house	5,214	760	All arrays in cabinet	0.1
	In kitchen	986	460	All arrays in cabinet	0.1
	Washing dishes	260 6.1	91 Hand contact	1 place setting 1 item (10-in plate)	0.8 0.02
	Dining	365	40	1 place setting	3.3
			180	1 place setting	0.2
			91	2 place settings	2.2
Total					7
Others	In house	4,780	760	All arrays in cabinet	0.1
	In kitchen	620	460	All arrays in cabinet	0.07
	Dining	365	40	1 place setting	3.3
			180	1 place setting	0.2
			91	2 place settings	2.2
Total					6

Table 3.11.6 Summary of Home Use Model and Potential IndividualEffective Dose Equivalents

^a 1 mrem/yr = 0.01 mSv/yr.

_

L___

•••

Exposure Scenario	Individual Annual Effective Dose Equivalent Rate (mrem) ^c	Total Collective Effective Dose Equivalent ^b (person-rem) ^c
Distribution	6	30
<u>Routine use</u> ⁴ As dinnerware On display	50° 0.06 [†]	4,000 ⁹
Disposal	0.4	8
Misuse	0.4	NA ^h

Table 3.11.7 Summary of Potential Radiation Doses From Uranium-Containing, Glazed Ceramic Tableware *

^a External dose estimates are based on the assumption that the glaze contains the maximum amount of uranium allowed under the exemption, 20% by weight. Collective doses are based on an annual distribution of 6,000 sets of tableware. The collective dose for dinnerware on display is based on 14% uranium, the maximum observed.

^b Refer to text for time period for collective dose calculations.

^c 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

^d Tableware is assumed to be used or on display for an average of 20 years.

^e Represents maximum exposed individual, typical usage would yield substantially lower doses. Refer to 3.11.4.1.2.

^f On display in the home.

⁹ Six thousand displays with six place settings each on display in a public place over 20 years. ^h Not applicable.