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Environmental Assessment of Consumer Products Containing Radioactive Material

Prepared by D. W. Buckley, R. Belanger, P. E. Martin,
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Science Applications, Inc.

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1. INTRODUCTION

1.1 OBJECTIVE

Since the issuance of the National Environmental Policy Act (NEPA) of 1969, all agencies of the Federal Government are required to prepare detailed environmental impact statements (EIS) on proposals for legislation and other major Federal actions which may significantly affect the quality of the human environment. The main objective of these studies is the careful consideration of environmental aspects of proposed actions. Licensing and regulatory policy and procedures for environmental protection as related to the Nuclear Regulatory Commission (NRC) are stated in the Code of Federal Regulations, Title 10, Part 51 (10CFR51).

At present, the NRC is reevaluating the adequacy of existing policy dealing with the distribution of consumer products containing radioactive material. As part of this reexamination, a generic environmental impact statement (GEIS) on consumer products will be prepared.

The objective of this report is the assessment of the impact of consumer products containing radioactive materials on people and the environment. An assessment of benefits and risks is made for presently distributed consumer products and possible alternatives. The work should prove to be an important source of information for the GEIS on consumer products containing radioactive material which will be written in the future.

For the purposes of this report, consumer products are considered to be those products, commodities, or materials which contain radioactive material and are available in the marketplace to the general public as "off-the-shelf" items and intended for personal and household use. The radioactive material can be byproduct material, source material, naturally occurring radionuclides, or accelerator-produced radionuclides. Products, such as building materials, and products intended solely for industrial and medical use are not considered to be consumer products for purposes of this report.

The work performed for this report is part of a study sponsored by the NRC. An earlier publication by the authors entitled "Environmental Assessment of Ionization Chamber Smoke Detectors Containing Am-241" was published in November, 1979 and reported the results of the first phase of the study. The present publication reports the results of the second phase of the study. A third phase is presently being undertaken to perform an assessment of the current criteria for NRC approval of the distribution of consumer products to the public, including an evaluation of existing regulatory policy. A third report, which will contain supporting data and recommendations for necessary modifications of NRC regulations dealing with consumer products, will be published upon completion of the third phase.

1.2 APPROACH

The approach followed in this report is to assess each important consumer product individually in separate sections. Each section provides the information necessary to arrive at a comprehensive assessment of each consumer product. First, a general overview or background information on the product is given, including product descriptions, distribution, life span uses, and other pertinent data.

Following the general overview, a discussion of the environmental impact of the product due to manufacture, distribution, use, and disposal is given. The particular benefits derived from the use of the product are discussed, as are the adverse impacts of product use, including both radiological and non-radiological aspects. The radiological impacts are expressed in terms of dose commitments and health effects, where appropriate, due to exposures received by the population of concern during the life span of the product. Included in the discussion of radiological impacts is the disposal of the product in landfills and by incineration, as well as the consequences of possible accidents or misuses of the product.

Following the discussion of environmental impacts, alternatives to the product are presented. The alternatives include both radioactive and non-radioactive products, although no radioactive alternatives appear to exist for some products. Having discussed both the benefits and impacts of the product and various alternatives, a cost-and-risk-benefit analysis is performed, where appropriate, to assess the product and various alternatives from both a cost- and risk-effectiveness viewpoint. The assessment for certain products is necessarily more qualitative than quantitative in nature. This is because the benefits of certain products are not easily quantified, owing largely to the fact the benefits are simply convenience-related and not safety-related.

The final section of each product assessment is a discussion of future trends or distribution of the product and similar products.

The order of the discussion of the individual consumer products is as follows:

Section 2	Radioluminous Timepieces
3	Static Eliminators
4	Dental Products
5	Incandescent Gas Mantles
6	Thoriated Tungsten Welding Rods
7	Optical Glass
8	Tableware and Glassware
9	Electron Tubes
10	Fluorescent Lamp Starters
11	Ionization Chamber Smoke Detectors (Summary of Previous Report)
12	Other Products

The detail of each product assessment varies depending on such factors as the amount of information available in the literature, communications with various organizations, and the relative importance of the product. For this reason, the assessment performed for a product such as radioluminous timepieces contains more detail than the assessment on tableware and glassware.

Four appendices are included with the report. Appendix A discusses the dose calculational methods used to assess the radiological impact of the products. Appendix B discusses radiological health effects and includes a listing of the health effect risk estimates used in this report. Appendix C discusses the regulations and safety criteria governing consumer products with radioactive material presently in effect in the U. S. The discussion covers naturally occurring and accelerator-produced radionuclides as well as byproduct and source material. Appendix D discusses the disposal of products, including disposal in landfills and via incineration. The particular analytical methods and models employed by the study are discussed.

1.3 DISCUSSION

The use of radioactive materials in products intended for use by consumers must be regulated in order to protect consumers from potential harm. It is obvious that products intended for ingestion, inhalation, or human application (e.g., foodstuffs, beverages, and cosmetics) should not contain radionuclides that are added intentionally. However, other products that achieve benefits ranging in nature from convenience to life-saving may contain intentionally-added radionuclides if the addition of the radionuclide permits the particular benefit to be realized. In some cases, alternatives that contain no radioactive materials may exist. Therefore, the question arises as to whether the use of radioactive material in consumer products is justified. To answer such a question, a thorough assessment based on a comprehensive set of data is required.

This report attempts to compile such a set of data on consumer products containing radioactive material. Table 1.1 is a listing of radiological data for radionuclides commonly found in consumer products. Many of these radionuclides are found in more than one type of product. As can be seen, the types of radionuclides vary considerably in terms of emitted radiations and radionuclide-dependent parameters, such as half-life.

1.4 SUMMARY

Table 1.2 is a list of the consumer products for which assessments have been made in this report. The specific details on each product are included in the appropriate section of the report. Much data has been generated pertaining to distribution, radionuclide content, and assessment of the product. In many instances, a historical overview of the particular product is given in order to demonstrate the manner in which the product came into use. Each assessment was made using all available literature and, for this reason, the references for each section should provide a thorough bibliography for the product of concern.

A brief summary for each product follows.

Table 1.1. Radiological Data for Nuclides of Interest.

Radionuclide	Half Life (Years)	Specific Activity (Ci/g)	Energy of Emission, MeV(%)			Other Radiations	Decay Product	Most Restrictive MPC (μCi/ml)	
			Alpha	Beta	Gamma			Air	Water
H-3	12.3	9.8E+3(1)		0.018(100)			He-3	2.0E-7	3.0E-3
C-14	5730	4.6		0.155(100)			N-14	1.0E-7	8.0E-4
Co-60	5.3	1.1E+3		0.31(99+) 1.5(0.12)	1.17(100) 1.33(100)		Ni-60	3.0E-10	3.0E-5
Ni-63	92	62		0.067(100)			Cu-63	2.0E-9	3.0E-5
Kr-85	10.8	400		0.67(99)	0.5(0.4)		Rb-85	3.0E-7	-
Cs-137	30	100		1.18(7) 0.51(93)	0.66(85) From Ba-137m		Ba-137	5.0E-10	2.0E-5
Pm-147	2.7	900		0.23(100)			Sm-147	2.0E-9	2.0E-4
Po-210	0.4	4.5E+3	5.3(100)		0.803(trace)		Pb-206	7.0E-12	7.0E-7
Ra-226	1620	1.0	4.78(95) 4.60(5)	0.09,0.17 (both e$^-$)	Rn x-rays 0.19(4) trace others to 0.6	Daughter radiations from Rn-222, Po-218, Pb-214, Bi-214, Po-214	Decay Chain	2.0E-12	3.0E-8
Th-232	1.4E+10	1.1E-7	4.01(76) 3.95(24)	0.04,0.06 (both e$^-$)	Ra L x-rays	Daughter radiations from Ra-228, Ac-228, Th-228, Ra-224, etc.	Decay Chain	1.0E-12	2.0E-6
U-238	4.5E+9	3.7E-7	4.20(75) 4.15(25)	0.03,0.04 (both e$^-$)	Th L x-rays	Daughter radiations from Th-234, Pa-234m, etc.	Decay Chain		
Am-241	458	3.3	5.49(85) 5.44(13)	0.02,0.04 0.05(all e$^-$)	Np L x-rays 0.06(36) trace others to 0.07	Daughter radiations from Pu-241	Np-237	2.0E-13	4.0E-6

(1) 9.8E+3 = 9.8x10^3

Table 1.2 Summary of Consumer Products with Radioactive Materials.

Consumer Product	Radionuclide	Radionuclide Content(uCi)	
		Average	Range
1. Radioluminous Timepieces			
Paint	H-3	1,850	100 - 25,000
	Pm-147	45	10 - 110
	Ra-226	0.2	0.002 - 4.5
Gas	H-3	195,000	160,000 - 200,000
2. Static Eliminators	Po-210	200 (Nominal)	30 - 200 (During useful
		500 (Nominal)	80 - 500 life)
3. Dental Products	U	2.E-5 (0.06 mg)	Up to 1.E-4 (Up to 0.3 mg)
4. Gas Mantles	Th	0.036 (325 mg)	0.028 - 0.044 (250 - 400 mg)
5. Welding Rods	Th	0.029 (260 mg)	0.005 - 0.031 (30 - 300 mg)
6. Optical Glass			
Ophthalmic Lens	Th	0.001 (9 mg)	Up to 0.002 (Up to 15 mg)
7. Tableware and Glassware	U	NC(1)	NC
8. Electron Tubes	H-3	21	NC
	C-14	47	NC
	Co-60	0.17	NC
	Ni-63	0.91	NC
	Kr-85	0.18	NC
	Cs-137	1.1	NC
	Pm-147	8.7	NC
9. Lamp Starters	Th	5.E-6 (0.05 mg)	NC
10. Ionization Chamber Smoke Detectors	Ni-63	NC	NC
	Ra-226	0.05	0.05 - 1.5
	Am-241	3.5	0.2 - 13.7

(1) Not calculated.

Radioluminous Timepieces

The principle of radioluminescence has been used since about 1910 for the purpose of illuminating the hands and dials of timepieces. Radium-226 (Ra-226) was used initially, but in recent years, tritium (H-3) and promethium-147 (Pm-147) have become the principal radionuclides used in radioluminescent products.

Until recently, these radionuclides were used with phosphorescent paints which were applied to the various timepiece components. An application which has become popular in recent years is the gaseous tritium light source (GTLS). These sources consist of small borosilicate glass tubes whose inside surfaces are coated with zinc sulfide. The tubes are filled with gaseous tritium and sealed. The main consumer use of the GTLSs has been as backlight sources for liquid crystal display (LCD) watches.

In 1977, the annual distribution of timepieces containing radioluminous paint was about seven million with H-3, about two million with Pm-147, and about 200,000 with Ra-226. In 1978, about two million watches containing GTLSs were distributed.

Analysis of the radiological impacts due to the life span of radioluminous timepieces from manufacture to disposal indicates that there is a small but positive health risk associated with the total number of radioluminous timepieces currently in distribution. When compared to the current cancer mortality rate, the radiological impact of radioluminous timepiece use appears to be very low, if not completely insignificant.

Static Eliminators

The use of alpha radiation to eliminate static electricity has been an established practice for many years. The use by consumers has essentially been limited to photographic and high fidelity needs. Polonium-210 (Po-210) is the only radionuclide used presently in consumer static elimination devices.

Two units are available presently, one containing a nominal 200 uCi of Po-210 and the other containing a nominal 500 uCi of Po-210. The devices include a brush along with the Po-210 source which is contained in ceramic microspheres resin-bonded to an aluminum backing.

Approximately 370,000 nominal 200 uCi units and 110,000 nominal 500 uCi units have been distributed since 1970. The maximum units distributed in one year were 50,000 and 20,000, respectively, for the 200 and 500 uCi units.

The resultant estimated health risk from manufacture, distribution, normal use, and disposal of existing consumer-oriented Po-210 static eliminators is found to be very small. This is attributable to the relatively short half-life of Po-210, the lack of a significant gamma exposure mechanism, and the integrity of the ceramic microsphere sources during normal use.

Dental Products

Since the early 1940's, small amounts of uranium oxide have been added to artifical porcelain teeth and porcelain powders used in the construction of dental prostheses in order to impart a fluorescent quality similar to the natural fluorescent appearance of normal teeth. The result has been a cosmetic benefit to the users of the dental prostheses.

In 1977, approximately 40 million porcelain teeth were distributed in the U. S. The average tooth weight was about 0.6 grams, of which 0.03 percent by weight is uranium. This is equivalent to about ten pounds of depleted uranium.

The risks to the individual who elects to wear dental porcelain prostheses are difficult to determine. The radiation dose rate to the basal mucosa from one porcelain tooth containing 300 ppm of uranium was found to be about one rem per year. It appears the probability of cancer induction from natural factors is much higher than from the potential dose from uranium in dental porcelain.

Incandescent Gas Mantles

The use of thorium in gas mantles began in 1885 when the Austrian Baron Carl Auer von Welsbach began investigating the properties of rare earth minerals. Since then incandescent lighting utilizing mantles has been used for home lighting, street lighting, automobile lighting, and for other outdoor lighting. Today, the mantles are used primarily for outdoor lighting, particularly camping, and some home use.

Incandescent mantles are used primarily in gas-fired lanterns and operate in a relatively simple manner. A vortex passes gas from a supply to the mantle. The mantle resembles a fine screen or wire mesh which surrounds the vortex. The gas flame produced from the vortex heats the mantle causing heat energy to transfer into light which is emitted from the mantle. The thorium in the mantle increases the emissivity of the mantle and the evenness of light produced. Typical mantles contain between 250 to 400 mg of thorium as oxide.

It is estimated that 25 million mantles are distributed annually. Approximately 82 percent are used in portable lanterns and 12 percent are used for residential lighting.

The health hazards or risks due to thorium mantles are of two types. The first is due to the radioactive nature of Th-232 and its progeny. The second is due to the use of beryllium in mantles and the potential for the development of beryllium disease. The largest impact due to the radioactive nature is from use by campers and their families. The mechanisms most responsible for exposure are mantle replacement by campers and radon-220 (Rn-220) emanation from lanterns used by campers and their families. Accidents with mantles were found to be potential health hazards, but were not found to be excessive.

Thoriated Tungsten Welding Rods

Several different welding processes are currently used by industry, including arc welding. Arc welding produces coalescence of two metal objects by heating the work metal with an electric arc or arcs. Arc welding equipment uses two kinds of electrodes, carbon or metal. One type of metal electrode is the tungsten electrode, which can be pure or alloyed with thorium oxide or zirconium oxide. The weight percent of thorium oxide in electrodes ranges from 0.35 to 2.2. The addition of thorium in tungsten electrodes increases electron emissivity, allows greater current carrying capacity, increases life, and offers greater resistance to contamination compared to pure tungsten.

Information on thoriated tungsten electrode distribution is not available from published references. It is estimated that 5.2 million electrodes are produced and used annually in the U. S. This corresponds to an annual use of about 3000 pounds of thorium oxide.

The radiological health implications of thoriated tungsten welding electrode usage are found to be low and within acceptable levels, considering the important benefits achieved through their use. Virtually all of the health risk associated with the electrodes is from the inhalation of thorium released during normal use.

Ophthalmic Lenses

The base material in ophthalmic glass is silica. Other materials are added to adjust the density and refractive index of the glass. Various oxides are used including zinc oxide. Thorium and uranium are natural contaminants of both silica and zinc. Manufactures of ophthalmic lenses have indicated that no thorium, uranium, or other radioactive materials are intentionally added to their products. An Ophthalmic Glass Radiological Standard has been published by the Optical Manufacturers Association to stablish a uniform maximum level for radiological emissions from ophthalmic glass.

Currently about 30 million ophthalmic lenses are distributed annually. This corresponds to about 55 percent of all ophthalmic lenses in distribution. The other 45 percent consists of plastic lenses. The 1977 population of people wearing eyeglasses was 96 million.

Alpha particles are emitted from glass ophthalmic lenses. The alpha dose to the corneal germinal layer of the eye of a wearer was estimated to be about four rem per year. However, the overall environmental impact due to glass ophthalmic lenses containing thorium and uranium is found not to be significant.

Glassware and Ceramic Tableware

Uranium compounds have been used for many decades to produce pigmented glazes for glassware and ceramic pottery and tableware. Examples of products containing uranium compounds are platters, pitchers, tumblers, vases, mugs, and bowls. Sodium uranyl carbonate was commonly used to produce fluorescent and irridescent glass in the past. Uranium oxides and sodium uranite were used in ceramic glazes, with concentrations ranging from one to 20 percent by weight.

Today, no manufacturer is known to use uranium as a glaze material for dinnerware, though the manufacture and distribution of decorative glassware containing U_3O_8 still takes place. Approximately 480,000 glassware pieces were distributed in 1978. As recently as 1972, drinking glasses containing uranium were distributed.

In the case of both decorative glassware and glazed ceramic products, some exposure of the public to ionizing radiation will result. In some instances, these exposures may be significant. Since the use of uranium in these products results in little or no benefit in terms of the health or well-being of the public, and since suitable substitutes for uranium exist, it is concluded that the exposures incurred as a result of product distribution and use, while low in most cases, are unnecessary and unwarranted.

Electron Tubes

Radioactive materials are currently in widespread use in many electronic applications. Faster and more reliable operation is achieved through preionization of the gases encased in electron tubes. Among the most common uses of electron tubes are voltage regulation, current surge protection, and as indicator lights. Indicator lights are provided in many appliances on the market, such as clothes washers and dryers, stereos, coffee makers, and pinball machines. Such lights typically contain tritium or Kr-85 with activities in the range of one to five uCi per unit. The other electron tube applications use radionuclides including Cs-137, Ni-63, Co-60, Pm-147, and Kr-85. The activites per unit are usually below one uCi.

Several hundreds of millions of indicator lights have been distributed in the last few years. Several million of the other electron tubes are distributed annually.

The level of risk associated with the normal use and disposal of electron tubes and with accidents is very low. Since some degree of benefit is gained by incorporation of radionuclides into electron tubes, and since this benefit is not achieved at a large increase in cost, these products are considered to be acceptable from both a cost- and risk-effectiveness point of view.

Fluorescent Lamp Starters

A fluorescent lamp starter is a device which functions to preheat the filaments of a fluorescent lamp. When fluorescent lighting was first designed and marketed, there were no fluorescent starters in their circuit and the functional life of these lamps was a few hundred hours. The design of circuitry, which includes some method of preheating the filaments of the fluorescent lamps, has managed to extend the functional life of a fluorescent lamp to 7000 hours or more. In the later 1940's and early 1950's, uranium was used in some of the more popular models of fluorescent lamp starters. In the mid 1950's, the industry switched to thorium in place uranium.

Ten to 15 years ago, the U. S. fluorescent lamp starter market amounted to an annual volume of 75 to 80 million units. In 1978, the annual U. S. sales volume for fluorescent lamp starters was 18.4 million. This drop in sales volume is the reflection of functional changes in fluorescent lamp circuitry. Today the consumer market is the main area of use for the 40 watt fluorescent lamp and it is this device that still requires the aid of starter circuitry.

No major environmental impacts were found for any manufacturing, distribution, use, or disposal operation related to thoriated fluorescent lamp starters.

Ionization Chamber Smoke Detectors

A detailed environmental assesment of ionization chamber smoke detectors was performed by the authors of this report and was published in an earlier document. Section 11 of this report summarizes the results of the detailed assessment.

Other Products

Other products containing radioactive materials not covered in the previous sections are identified in Section 12 and discussed briefly. These products include miscellaneous radioluminous products, military surplus commodities, educational products, photographic toner, and uranium in pottery products and glass enamel frits.

2. RADIOLUMINOUS TIMEPIECES

2.1 PRODUCT DESCRIPTION AND BACKGROUND

2.1.1 General Discussion

The principle of radioluminescence has been used for some time for the purpose of illuminating the hands and dials of timepieces. Since about 1910, radium-activated luminous paint has been used on watches, clocks, and various other types of instruments. In recent years, tritium (H-3) and promethium (Pm-147) have been the principal radionuclides used in radioluminescent products, although radium (Ra-226) is still used to some extent. A brief discussion of the radioluminescence process and its application to the timepiece industry is presented below.

2.1.2 Product Description

Radioluminous compounds are mixtures of radioactive materials and phosphors. Alpha or beta particles emitted by radioactive decay processes interact with the electrons of the phosphor molecule resulting in molecular excitation. Excess energy is then emitted as visible light as the molecule returns to its ground state. The most commonly-used phosphor is zinc sulfide to which small amounts of trace metals have been added. Radium-activated luminous compounds are usually produced by mixing radium sulfate and zinc sulfide with a varnish or other medium to produce a fairly thick paint. The paint is normally applied to the timepiece by hand, using a small pointed rod. If large numbers of the same type of dial (or other item) are to be luminized, the paint is often applied by machine (Duggan 67).

Tritium-activated paint consists of zinc sulfide coated with a tritiated polymer and mixed with an adhesive dissolved in an organic solvent. Xylene can be added to the mixture if thinning is desired. Methods of application of the luminous paint range from manual painting to automated rotary applicators (Moghissi 70).

Another application of tritium-activated luminescence which has become popular in recent years has been the use of gaseous tritium light sources (GTLS). Small borosilicate glass tubes, the inside surfaces of which are coated with zinc sulfide, are filled with gaseous tritium (HT or T_2) and then sealed. The glass tubes are produced in a wide variety of shapes and sizes depending on the particular application. Gaseous tritium light tubes are commonly used as backlight sources for liquid crystal display (LCD) watches. Typically, a GTLS used for backlighting an LCD watch will contain between 160-200 mCi of tritium gas in two tubes which are bonded to a metal tray with a shock-absorbent silicone adhesive. The tray is sealed to a transflector-display panel and the entire assembly is encapsulated in a plastic collar. This reduces the probability of breakage and would prevent a quick release of the tritium gas in case of tube breakage.

The tritium gas used in GTLS applications has a specific activity of up to 2.6 Ci/cm^3 at STP. In order to purify the gas, it is adsorbed on finely dispersed activated metallic uranium at room temperature. All contaminants will either react irreversibly with the uranium, or will not react and will be removed by the vacuum system used to evacuate the tube. The glass container is evacuated to 10^{-4} mm Hg while simultaneously being heated to 350°C. The purified tritium gas is fed into the glass container immediately after the heating and evacuation has been completed by desorbing the tritium from the uranium at 400°C. As a result of this procedure, tritiated water vapor (HTO or T_2O) is usually present at concentrations of less than two percent (NEA 75).

Another radionuclide currently being used in radioluminescent timepieces is Pm-147, a rare earth isotope which is abundantly produced in the fission process. Presently there is only one plant in the U. S. which performs luminizing operations involving Pm-147. The vast majority of promethium-activated luminous timepieces are imported from other countries. The major U. S. user of Pm-147 is a plant which manufactures special devices as well as timepiece components and, due to its specific market conditions, is an exceptionally well operated plant which is not representative of general dial painting operations (Moghissi 75). Thus, information concerning the radiological health aspects of Pm-147 radioluminescence applications is very limited.

2.1.3 Distribution

Tables 2.1 through 2.4 present data on the distribution of radioluminous timepieces in recent years. The data for tritium and Pm-147 timepieces were summarized from NRC licensing reports, while Ra-226 timepiece distribution was estimated on the basis of conversations with manufacturers and from data reported in the literature.

No timepiece manufacturers have been located in the U. S. that are presently manufacturing units incorporating Ra-226 as the source material. It has been estimated that ten million watches containing Ra-226 may still have been in use in 1977 (NCRP 77). A survey aimed at determining the population exposures from gamma-emitting luminescent material on clocks was made in 1974 (Butler 75). According to the survey, which only covered 48 households in East Tennessee, one out of every three households has one clock emitting penetrating radiation, a characteristic of Ra-226. The findings of this survey, however, have recently been challenged as being of questionable validity (Moghissi 77).

There are at least four companies still in existence which refinish timepieces with radium paint. Listed below are estimates of the number of radium dials being refinished and the average amount of Ra-226 being applied to each dial.

	Dials or Activity per			
	Dial	Week	Month	Year
Average number dials	-	108	468	5616
Average activity (uCi)	0.22	24	103	1236

It is interesting to compare the quantity of Ra-226 (0.2 uCi/dial) being employed in the refinishing with some previously published data on quantities present on originally manufactured units. In a paper by Klein, et al. (1970), the average radium content of military pocket watches was found to be 1.0 uCi. In 1975, a report by Moghissi and Carter indicated an estimated average activity per timepiece of 0.5 uCi (Moghissi 75). The estimate by Moghissi and Carter is in close agreement with what manufacturers have indicated during the current study in that the total activity per watch would be approximately twice the activity of a single dial if radioluminous hands are included.

Table 2.1. Total Distribution (Domestic and Import) of Luminescent Timepieces Containing Tritium.

Year	Item	No. Distributed	Total Activity (Ci)	Average Activity (mCi) Per Unit
Through 1969	-	-	61,400	-
1970	-	-	33,300	-
1971	Watches[1]	11,700,000	14,200	1.21
	Clocks	720,000	200	0.25
	Total		14,400	
1972	Watches	12,700,000	17,000	1.34
	Clocks	595,000	3,700	6.16
	Total		20,700	
1973	Watches	5,270,000	11,900	2.26
	Clocks	907,000	300	0.35
	Total		12,200	
1974	Watches	4,860,000	8,000	1.65
	Clocks	429,000	200	0.51
	Total		8,200	
1975	Watches	6,740,000	6,200	0.93
	Clocks	75,500	100	1.61
	Total		6,300	
1976	Watches	6,740,000	20,700	3.07
	Clocks	8,380	100	14.28
	Total		20,800	
1977	Watches	6,170,000	22,900	3.71
	Clocks	944,000	600	0.60
	Total		23,500	
Total (Through 1977)			201,000	

(1) Items reported as "timepieces" were recorded as watches.

Table 2.2. Distribution of Backlit (Tritium Light Tube) Watches.

Year	No. Watches Distributed	Total Activity (Ci)	Average Activity (mCi) per Watch
1976	12,300	2,000	160
1977	81,900	14,300	174
1978	2,020,000	395,000	196
Total	2,110,000	411,000	195

Table 2.3. Total Distribution (Domestic and Import) of Timepieces Containing Pm-147.

Year	Item	No. Distributed	Total Activity (mCi)	Average Activity (uCi) Per Unit
1970	Watches(1)	500,000	15,300	30.5
	Clocks	874,000	27,800	31.9
	Total		43,100	
1971	Watches	542,000	26,300	48.6
	Clocks	1,540,000	41,600	27.1
	Total		67,900	
1972	Watches	735,000	35,300	48.1
	Clocks	2,030,000	34,800	34.0
	Total		70,100	
1973	Watches	852,000	46,000	53.5
	Clocks	1,360,000	62,100	45.7
	Total		108,100	
1974	Watches	958,000	70,000	73.1
	Clocks	857,000	36,600	42.7
	Total		106,600	
1975	Watches	1,470,000	96,000	65.3
	Clocks	694,000	44,900	64.7
	Total		140,900	
1976	Watches	1,280,000	67,900	53.3
	Clocks	587,000	31,700	54.0
	Total		99,600	
1977	Watches	1,090,000	56,800	52.0
	Clocks	947,000	46,600	49.2
	Total		103,400	
Total (1970-1977)			740,000	--

(1) Items reported as "timepieces" were recorded as watches.

Table 2.4. Distribution of Timepieces Containing Ra-226.

Year	Item	No. Distributed	Total Activity (mCi)	Average Activity (uCi) Per Unit
1970(1)	Watches	Negligible	---	---
	Clocks	2,800,000	Unknown	Unknown
1971	Watches	Negligible	---	---
	Clocks	2,800,000	Unknown	Unknown
1972	Watches	Negligible	---	---
	Clocks	2,800,000	Unknown	Unknown
1973	Watches	Negligible	---	---
	Clocks	2,800,000	Unknown	Unknown
1974(2)	Watches	Negligible	---	---
	Clocks	1,500,000	Unknown	Unknown
1975	Watches	Negligible	---	---
	Clocks	500,000	75	0.15
1976	Watches	Negligible	---	---
	Clocks	230,000	34.5	0.15
1977	Watches	Negligible	---	---
	Clocks	230,000	34.5	0.15
1978	Watches	Negligible	---	---
	Clocks	60,000	9.0	0.15

(1) Data for 1970 to 1973 based on data of Moghissi, 1975.

(2) Data for 1974 to 1978 based on information primarily from final U. S. manufacturer of radium dials and hands.

At present, tritium and Pm-147 are the only radionuclides used in new watches for painting of dials and hands. Their use in clocks has also increased relative to radium as the industry is apparently phasing out radium usage. As previously mentioned, however, clocks with Ra-226 are still being distributed.

The use of gaseous tritium light tubes in watches has become common since 1976, when the gas tubes were first introduced. The growing popularity of LCD watches could result in an increase in their distribution over the next few years. However, recent events in the nuclear industry, together with an increase in digital watch imports, have caused a shift in the GTLS market. Digital watch-makers are currently rejecting the application of light tubes in their products, making the future of these devices in this application uncertain. These tubes are also used in a variety of other products, as discussed in Chapter 11.

2.2 ENVIRONMENTAL IMPACT

2.2.1 Benefits

The main benefit associated with the use of radioactive materials in timepieces is the availability, on a continuous basis, of a useful source of visible light. The useful life of the radioluminous product, which is taken here to mean the length of time the minimum luminosity remains above the level of effective visibility, is a function of the type of both the radionuclide and the phosphor. The most commonly used phosphor, zinc sulfide, is damaged by the absorption of ionizing radiation, resulting in a decreased ability to produce light.

The effective half-life of a radioluminous timepiece (T_t) is given by:

$$T_t = \frac{T_p T_r}{T_p + T_r}$$

where

T_p = half-life of phosphor (yr)

T_r = physical half-life of radionuclide (yr)

According to the International Atomic Energy Agency (IAEA) a value of ten years should be used for the half-life of zinc sulfide (IAEA 67). Thus, the half-lives for tritium, Pm-147, and Ra-226 timepieces, respectively, are:

$$\frac{(12.3\ \text{yr})(10\ \text{yr})}{(12.3\ \text{yr} + 10\ \text{yr})} = 5.5\ \text{yr}$$

$$\frac{(2.6\ \text{yr})(10\ \text{yr})}{(2.6\ \text{yr} + 10\ \text{yr})} = 2.1\ \text{yr}$$

$$\frac{(1600\ \text{yr})(10\ \text{yr})}{(1600\ \text{yr} + 10\ \text{yr})} = 10\ \text{yr}$$

For gaseous tritium light tubes, the loss of brightness is slower than for luminous paints, resulting in an effective half-life of up to eight years (NEA 73).

As Moghissi and Carter have pointed out, an estimation of the effective half-life of tritium timepieces must not disregard the fact that tritium is continuously diffusing from the timepieces (Moghissi 75). Fitzsimmons, et al. (1972), measured tritium release rates from watches of 4.6 to 83 nCi/day. If a mean value of 1.85 mCi of tritium total activity per watch is assumed, this would result in an average release half-life of 55 years. These results conflict with the data summarized by Moghissi, et al. (1978), which suggest a release half-life of about four years, and with those of McMillan (1978), who reported measured average release rates of 172 nCi/day. Krejci and Zeller (1978) have suggested the following release rates which are a function of the specific activity of the paint:

Specific Activity (mCi/g)	Release Rate (percent/yr)
100	4.6
250	5.0
420	5.4
650	8.2

For a watch of 1.85 mCi, an average release rate of about 100 uCi/yr would result, corresponding to a release half-life of 9.6 years.

On the basis of the above, a release half-life of ten years will be assumed. This results in a reduction of the effective half-life of the timepiece from 5.5 to 3.6 years.

Aside from the effective luminosity half-life, the useful life of the product itself must be considered. In 1967, an IAEA expert panel suggested that a period of ten years is representative of a typical timepiece lifespan (IAEA 67). Since that time, however, changes have taken place in the watch industry which tend to reduce that period. Recent technological advances have made available a wide variety of timepieces at very moderate prices, while repair costs have soared. This makes it more likely that a person will purchase a new watch instead of having a broken or malfunctioning watch repaired. For the purpose of risk-benefit analysis, the lesser of the values of eight years, or twice the effective luminosity half-life, will be used as the timepiece lifespan. These values are:

H-3 (radioluminous paint)	Seven years
H-3 (gaseous light tube)	Eight years
Pm-147	Four years
Ra-226	Eight years

The amount of a radionuclide required to produce a particular level of light must also be evaluated. The luminous intensity of a source is measured in units of candela (cd) where 1 cd is the luminous intensity, in the perpendicular direction, of a surface of 1/600,000 square meter of blackbody at the temperature of freezing platinum under a pressure of 101,325 newtrons per square meter (ASTM 76). A more convenient unit for use here is the microcandela (ucd), which is equal to 10^{-6} cd. According to Krejci and Zeller (1978), recent advances have resulted in luminous yields for tritium paints of up to 0.2 ucd/mCi. This is an increase by a factor of two over previously obtained yields. For tritium light tubes, the light yield has been reported as 0.18 ucd/mCi (NEA 73). Yields for Pm-147 and Ra-226 can be estimated from the relative luminous efficiency data given by the IAEA panel (1967) as 3 ucd/mCi and 500 ucd/mCi, respectively.

If one assumes the useful life and luminous yield estimates listed above, the quantities of tritium and Pm-147 required to produce an average luminosity equal to that produced by 1 uCi of Ra-226 can be calculated. These values are:

Tritium paint	7,140 uCi (0.1 ucd/mCi) or 3,570 uCi (0.2 ucd/mCi)
Tritium gas	3,500 uCi
Pm-147	240 uCi

In addition, radioluminosity in timepieces is beneficial in that it results in reduced energy consumption. This effect is greatest in electric clocks which utilize lamps for illumination. The widespread application of self-lighting timepieces would result in substantial cumulative savings in energy consumption and associated costs over a long period of time.

2.2.2 Radiological Impact

Around 1910, radium-activated paint was used on to illuminate the hands and dials of some of the more expensive Swiss and German watches. A few years later, a cheaper paint was developed in the U. S., and the dial painting industry flourished as a result of the demand for military instruments and watches during World War I. About one thousand people worked in the radium extraction and dial painting industries in northern New Jersey between the years of 1917 and 1924. Despite some previous indications of the dangers associated with radium use, it was not until the unequivocal radium intoxication deaths of five workers in 1925 that a full appreciation of the internal radiation hazard was developed. (see Moghissi 75 and Sharpe 79 for a review of this subject.) In recent years, radionuclides with more desirable radiological hygiene properties (most notably tritium and Pm-147) have for the most part replaced radium as luminous paint activators.

This section assesses the total radiological health impact resulting from the manufacture, use, and disposal of radioluminous timepieces. As a consequence of a number of shortcomings (such as product variability, lack of data, generalization of assessment methodology, etc.), this assessment cannot be expected to represent more than a general estimate of the health risk associated with these products. Nevertheless, the results of this assessment are useful in identifying the relative hazard represented by the products containing different radionuclides, and in helping to delineate the upper bound of the level of risk.

2.2.2.1 Occupational Exposure

Much has been written on the subject of occupational exposure in the dial painting industry, especially with respect to radium and tritium. The legacy of the radium dial painters has been thoroughly studied for many years. Indeed, the selection in 1941 of 0.1 ug radium as the maximum permissible skeletal burden was based on the 30 cases of radium intoxication, most of whom were dial painters, which had been studied at that time (Evans 67).

In recent years, attention has been focused on the tritium-related occupational hazard (Moghissi 70, Lambert 72). Seelentag (1973) has presented strong evidence that two people have died and several others have been seriously affected by the incorporation of tritium in the course of producing tritium luminous compounds in Europe. Unfortunately, very little data is available regarding occupational exposures to Pm-147 in the radioluminous industry. Since essentially all Pm-147 timepieces are currently imported, these data would be of limited usefulness in this report.

Tritium Luminous Compounds

Most of the occupational exposure in the tritium timepiece industry in this country is incurred in the course of dial painting and watch assembly, storage, and repair. Most of the tritium-activated luminous compounds which are used on the timepieces are imported. In some cases, the luminous compound ingredients are imported and then mixed in the U. S. Approximately two-thirds of all of the tritium luminous timepieces distributed in the U. S. from 1970 through 1977 were imported, either as painted components or as intact timepieces.

Table 2.5 summarizes occupational exposure data for various tritium timepiece activities. Only one U. S. dial painting plant is represented in this summary, and it is not known if these exposures are typical of other American dial painting operations. The best data currently available are those from seven Swiss dial-painting plants averaged over four years (Table 2.6). As can be seen, the total dose per curie processed seems to be related to the specific activity of the paint. This relationship, however, is ill-defined and uncertain since

Table 2.5. Summary of Occupational Exposures in Various Tritium Timepiece Plants.

Type of Activity	Average Individual Dose (mrem/yr)(1)	Total Dose (person-rem/yr)	Amount of H-3 Processed (Ci/yr)	Total Dose per Curie Processed (person-mrem/Ci)	Reference
Luminous Compound(2) Manufacture	1.7E+3	3.0E+2	3.7E+6	8.3E-2	Krejci, 1975
Dial Painting(3)	8.6E+2	7.1E+1	1.1E+4	6.3E+0	Krejci, 1975
Dial Painting(4)	2.0E+3	3.4E+1	1.8E+3	2.0E+1	Moghissi, 1975
Watch Assembly(5)	2.6E+1	8.6E-1	1.5E+2	5.7E+0	Bradley, 1971
Watch Storage(6)	1.4E+2	1.1E+0	3.0E+2	3.6E+0	Bradley, 1971
Watch Repair(5)	1.6E+1	1.1E+0	1.4E+2	7.4E+0	Bradley, 1971

(1) Assumes H = 0.1C where H is the annual dose equivalent in mrem/yr and C is the average urine concentration in mCi/ℓ (see Appendix A).

(2) Nine years of data from several luminous compound manufacturing plants in Switzerland.

(3) Data from seven Swiss plants averaged over four years.

(4) Data from one American firm averaged over one year.

(5) Data from two American firms averaged over one year.

(6) Data from three American firms averaged over one year.

Table 2.6 Occupational Exposures at Seven Swiss Tritium Dial Painting Plants(1)(2)

Plant	Average Specific Activity of Paint (mCi/g)	Processed H-3 (Ci/person-yr)	Average Annual Dose to Individuals (mrem/yr)(3)	Risk (mrem/Ci)
A	140	144	700	4.86
B	170	210	1,200	6.05
C	200	238	800	3.36
D	240	107	510	4.77
E	260	116	780	6.72
F	320	85	1,200	14.11
G	420	56	770	13.75

(1) From Krejci, 1975

(2) Data averaged over four years

(3) Assumes $H = 0.1C$ where H is the annual dose equivalent in mrem/yr and C is the average urine concentration in mCi/ℓ (see Appendix A).

there are many other variables which can influence the total dose. For example, the amount of tritium processed per person-year and the relative extent of implemented radiological controls would also influence the total exposure incurred. The average value for the total risk for the seven Swiss and the one American dial-painting operations is about 8 mrem per Ci processed. The data given in Table 2.7 are useful in revealing the distribution of doses to workers using tritiated luminous compounds and tritium gas in the United Kingdom.

Between one-half and two-thirds of the tritium timepieces are imported intact. The remainder are imported as components and assembled in the U. S. The doses to workers involved in timepiece assembly and storage operations must also be considered. This assessment utilizes the data of Bradley, et al. (1971), who reported exposure data for workers involved in timepiece assembly, storage, and repair operations in the New York City area. An estimation of the annual average occupational dose resulting from tritium self-luminous timepiece activities is presented in Table 2.8.

Gaseous Tritium Light Sources

The use of gaseous tritium for illuminating watches has certain advantages (from an exposure standpoint) over the use of tritium luminous compounds. Leakage from the tubes is typically less than 100 nCi/day, while leakage from intact watches is typically below 10 nCi/day (Ristagno 78). The incidence of tube breakage, both in the assembly plant and in purchased watches, is very low. One company has reported that less than 0.03 percent of the tubes handled in one year were broken during wristwatch assembly, and that 0.005 percent of the watches distributed in one year were returned due to tritium tube breakage (Ristagno 78).

Little data is available regarding occupational exposures as they relate specifically to gaseous tritium timepiece manufacturing. Some data has been provided by light tube manufacturers, but these data reflect exposures incurred as a result of activities involving miscellaneous radioluminous products as well as timepieces. Based on discussions with manufacturing representatives, an estimate of the total occupational dose associated with gaseous tritium timepiece manufacture and assembly was made, and these results are given in Table 2.9. It should be emphasized, however, that the estimates for light tube manufacture are probably overestimates since other types of products may have

Table 2.7 Distribution of Doses to Workers Using Tritiated Luminous Compounds and Tritium Gas in the United Kingdom.(1)

Year	No. of Workers Monitored	Percent of Workers Receiving the Following Annual Doses			
		<0.1 rem	0.1-1.5 rem	1.5-5 rem	>5 rem
1963	43	62.8	34.9	2.3	0
1964	56	51.8	44.6	3.6	0
1965	47	61.7	38.3	0	0
1966	42	19.0	71.4	11.9	0
1967	56	39.3	58.9	0	0
1968	56	41.7	58.9	0	0
1969	82	31.7	59.8	6.1	2.4(2)
Average	54.6	43.9	52.4	3.4	0.3

(1) From Lambert and Vennart, 1972

(2) This percentage represents accidental doses to two workers of seven and 29 rem as a result of leaks from flasks which contained several hundred curies of tritium gas.

Table 2.8 Average Annual Occupational Exposures in the U. S. Tritium Luminous Compound Timepiece Industry.

Type of Activity	Risk (mrem/Ci)	Amount Processed (Ci/yr)	Total Dose (person-rem)
Dial Painting	8.2E+0	5.8E+3	4.8E+1
Timepiece Assembly	5.7E+0	5.0E+3(1)	2.9E+1
Timepiece Storage	3.6E+0	5.0E+3(2)	1.8E+1
Repair or Refinishing	7.4E+0	1.5E+3(3)	1.1E+1
Total			1.1E+2

(1) Assumes that one-half of all imported timepieces are assembled in U. S.

(2) Assumes that one-half of all imported timepieces are stored for a significant period of time before distribution.

(3) Assumes that one timepiece in ten is repaired or refinished annually.

Table 2.9. Annual Occupational Exposure in the U. S. Gaseous Tritium Light Source Timepiece Industry

Type of Activity	Average Individual Dose (mrem/yr)	Total Dose (person-rem/yr)	Amount Processed (Ci/yr)	Total Risk (person-rem/Ci)
Light Tube Manufacture	7.5E+2(1)	1.9E+1	4.0E+5(2)	4.8E-5
Timepiece Assembly	4.5E+0(3)	9.0E-1	4.0E+5	2.3E-6
Total	-	2.0E+1	-	5.0E-5

(1) Exposure data from 1978-1979

(2) Quantity processed in 1978

(3) Exposure data from 1976

contributed to this dose. The data used for timepiece assembly are those reported by Ristagno (1978), and it is believed that there is much less uncertainty associated with these estimates.

While it is evident that the occupational exposures in the tritium gas timepiece industry are low, the potential exists for exposures of much greater magnitude since large quantities of tritium are stored and processed. Accidental exposures could occur which are of a magnitude comparable to, or greater than, the occupational dose limit (1.25 rems per calendar quarter). The magnitude of accidental exposures would be determined by the volume of gas released, the rate at which the HT or T_2 gas is oxidized to HTO, as well as occupancy, ventilation, and other factors. Because of the uncertainties involved in estimating the probabilities and magnitudes of different types of accidents, this type of estimation will not be attempted here.

In addition, gaseous tritium plants typically release more radioactivity in gaseous effluents than do other types of radioluminous processing plants. Once again, this is due primarily to the larger quantities of radioactive material processed. Indeed, excessive tritium releases to the environment constituted a major reason for one plant recently shutting down. Another plant reported that tritium releases are responsible for an annual dose to the surrounding population of about 10 person-rem compared to a total background population dose of 1.2×10^6 person-rem (SPL 80).

Pm-147

As previously mentioned, virtually all of the Pm-147 timepieces distributed in the U. S. are imported from overseas. In addition, nearly all of these timepieces are imported as assembled units, so very little assembly is performed in this country. Because of the lack of data regarding occupational exposure to Pm-147 in this country, this aspect will not be further considered.

Radium

The tragic history of the women radium dial painters is a classic case in the annals of occupational health. The water-based paint which was used in the early days of the American radium timepiece industry required brush application, whereas most radioluminous paints used elsewhere had an oil or varnish base and were applied with glass or metal rods. Characters were usually painted using very fine camel hair brushes which most workers pointed by drawing against the corners of their mouths. Since payment was usually by piecework, the more characters one painted, the more frequently she pointed her brush and, consequently, the more radium she ingested (Sharpe 79). According to Hoffman (1925), a worker who painted 250 dials in a day would swallow about 1.75 grams of paint daily, which represents a weekly dose of 15 to 125 ug of radioactive material. Cases of jaw necrosis (or "jaw rot" as it was known locally), osteosarcoma, and aplastic anemia resulted in excess numbers, and were recognized shortly thereafter as having been caused by radium intoxication.

Although the radium timepiece industry nearly collapsed by the end of the twenties, the use of this material for self-luminous devices has persisted. Today, at least four firms in this country are known to use radium-based paint for refinishing timepieces. No manufacturers have been identified as currently producing radium timepieces. As recently as 1976, however, at least two firms were still manufacturing timepiece dials and hands. Currently, about 1,200 uCi of Ra-226 per year is used in this country in the course of refinishing timepieces.

Duggan and Godfrey (1967) have performed thorough surveys of fourteen factories and nine workshops using radium in the United Kingdom. They measured facility surface contamination levels as well as radium body burdens of the workers. A summary of these results is given in Table 2.10. As can be seen, an average body burden of 2 nCi results from a yearly processing of 1 mCi. According to the ICRP, the fraction of ingested radium which reaches body fluids after entry into the gastrointenstinal tract is 0.2 to 0.3 (ICRP 59, ICRP 79). If it is assumed that all of the measured body burdens resulted from ingestion (as opposed to inhalation), the average intake of radium for these workers would have been up to 10 nCi per mCi processed.

As Moghissi and Carter have pointed out, the full impact of radium timepiece refinishing must take into consideration the internal hazard represented by the radon series daughter products, and the external radiation hazard from radium (Moghissi 75). They suggest using a dose conversion factor of 1,520 mrem/yr to the basal cells of segmented bronchioles for continuous exposure to a radon concentration of 1 pCi/l. Assuming an average exposure time of 2,000 hours, an individual would receive an annual lung dose of up to 350 mrem. Duggan and Godfrey (1967) reported that typical radon concentrations in the plants they

Table 2.10. Summary of Occupational Exposure Data for Radium in the United Kingdom(1)(2).

No. of Workers	46
Average Hours of Exposure per Week	24
Processed Radium (mCi/person-yr)	19
Average Body Burden (nCi)	14.9
Body Burden per mCi Processed per Year (nCi/mCi-yr)	2.0

(1) From Duggan and Godfrey, 1967.

(2) Data from 23 plants, 14 of which were classified as factories and 9 of which were classified as workshops.

studied were on the order of 10 pCi/l. This would result in an average individual lung dose of 3.5 rem, and a collective dose of about 185 rem per Ci processed. In actuality, the doses would probably be significantly lower since most workers are not exposed to these concentrations on a continuous basis. Since workers in these plants usually occupy themselves with many matters other than dial painting in the course of a day (e.g., coffee breaks, washing hands, other chores, etc.), the daily exposure period will surely be less than eight hours.

As indicated earlier, 1,200 uCi of radium are used annually in this country for refinishing timepieces. Assuming an average value of 19 mCi for the amount of radium one person can process in one year, it can be seen that less than one person working full-time would be required to process the amount used annually. This assessment will assume that one worker in each of the four plants currently operating is exposed continuously (40 hours per week, 50 weeks per year) to 300 uCi at an average distance of one meter. The total external exposure would be:

$$\frac{(0.83\ \frac{\text{R-m}^2}{\text{hr-Ci}})\ (4)(3\text{x}10^{-4}\ \text{Ci})(2000\ \text{hr/yr})}{(1\ \text{m})^2}$$

$$= 2\ \text{R}$$

Additional external exposure and possibly some radon or radon daughter product inhalation could result from other operations, such as the assembly or storage of refinished timepieces. These exposures, however, would probably be a small fraction of those resulting from refinishing operations. A summary of the collective organ and total body doses resulting from an annual processing of 1.2 mCi of radium is contained in Table 2.11.

2.2.2.2 Exposures During Distribution

Personnel exposures as a result of tritium and Pm-147 timepiece distribution, catalogue warehousing, and retail sales have been calculated by McDowell-Boyer and O'Donnell (1978a, 1978b). The exposures calculated by those authors are summarized in Table 2.12. Exposures from radium watch distribution were estimated by scaling the Pm-147 exposures calculated by McDowell-Boyer and

Table 2.11. Collective Dose Associated with Annual Processing of 1.2 mCi of Radium.

Organ	Dose or Dose Commitment (Rem)			
	Radium Ingestion(1)	Radon Inhalation	Direct Radiation(2)	Total
Total Body	5.3E+0	-	1.1E+0	6.4E+0
Liver	1.3E-4	-	1.0E+0	1.0E+0
Bone	7.2E+0	-	1.1E+0	8.3E+0
Red Marrow	7.2E+0(3)	-	1.1E+0	8.3E+0
Lungs	-	2.2E-1	1.1E+0	1.3E+0
G. I. Tract	7.9E-3	-	1.1E+0	1.0E+0
Gonads	4.1E-3(4)	-	1.2E+0	1.2E+0

(1) Fifty-year dose commitment resulting from one-year intake.

(2) Dose conversion factors for external radiation are averages of values for males and females using an effective energy of 0.8 MeV for Ra-226 in equilibrium with its daughters (Kathren 76).

(3) Assumed to be equal to dose commitment for bone.

(4) Assumes dose conversion factor of 0.34 rem/mCi (ICRP 79).

Table 2.12. Radiation Doses to Humans during Distribution of Radioluminous Timepieces.(1)

Exposed Group	Dose (Person-rem)			
	H-3 Paint	H-3 Gas	Pm-147	Ra-226
Parcel delivery system				
Truck drivers	3.7E-2	3.2E-3	4.7E-4	2.0E-4
Terminal Workers	2.6E-1	6.6E-3	2.4E-2	1.0E-2
Marketing system				
Catalogue sales workers	2.4E+0	4.4E-1	3.6E-2	1.5E-2
Large jewelers				
● Employees	1.4E+0	2.0E-1	3.6E-2	1.5E-2
● Customers	3.7E-1	4.3E-2	2.3E-2	9.6E-3
Small jewelers				
● Employees	1.7E+0	2.7E-1	2.4E-2	1.0E-2
● Customers	2.2E+0	3.5E-1	3.2E-2	1.3E-2
Department Stores				
● Employees	2.7E+1	4.4E+0	8.7E-1	3.7E-1
● Customers	1.9E+1	3.0E+0	5.9E-1	2.5E-1
Wholesalers	3.0E-1	4.0E-1	1.3E-2	5.5E-3
Total	5.4E+1	9.0E+0	1.6E+0	6.7E-1

(1) Doses for H-3 and Pm-147 are from McDowell-Boyer and O'Donnell, 1978a and 1978b; doses for Ra-226 are calculated using same scenario and assuming an annual distribution of 1.2 mCi.

O'Donnell by factors which take into account differences in external radiation levels, number of timepieces distributed, and average activity per timepiece. This is not entirely justifiable since radium timepieces are not marketed in the same manner as Pm-147 timepieces. This method, however, will suffice for the purposes of comparatively evaluating the different types of timepieces.

As can be seen from the results, the total doses to the parcel delivery, marketing, and shopping populations are relatively small for each of the timepiece types. Tritium luminous compound timepieces result in the largest doses, followed by GTLS, Pm-147, and radium timepieces, in that order. If the doses are calculated on a "per curie" or a "per timpiece" basis, however, the results are significantly different, as illustrated by Table 2.13.

2.2.2.3 Exposures To Users

This section estimates the doses to users of radioluminous watches and clocks, and to bystanders, under normal conditions.

Tritium Luminous Compounds

The estimation of doses to users of timepieces utilizing tritium luminous compounds has been made by many authors in the past. Fitzsimmons, et al. (1972) observed an average urine concentration of 4.1 nCi/l (range: 1.5 to 8.6 nCi/l) in eight persons wearing tritium luminous watches. This level of tritium in the urine corresponds to an annual dose to the average wearer of 0.33 mrem/year.

Moghissi and Carter (1975) have reported results of studies on human subjects exposed to tritium from radioluminous watches. Based on measured urine concentrations, the average dose was determined to be 0.03 mrem per mCi of tritium paint (range: 0.012 to 0.44 mrem/mCi). Moghissi and Carter assumed that the average activity of a tritium timepiece is 5 mCi and estimated that the annual radiation dose to the user is 0.15 mrem.

Based on the distribution data reported in Section 2.1.3, it appears that the average activity of tritium watches distributed in recent years is about 1.85 mCi. Using the dose conversion factor reported by Moghissi and Carter, this corresponds to an annual radiation dose to the user of 0.06 mrem. The collective dose associated with the average annual watch distribution (7.8×10^6 watches per

Table 2.13. Doses Due to Timepiece Distribution as a Function of Activity and Timepiece Number

Timepiece	Total Dose (Person-rem)		
	Actual Estimated	Per Curie	Per Timepiece
H-3 Paint	5.4E+1	4.2E-3	6.4E-6
H-3 Gas	9.0E+0	4.5E-5	9.0E-6
Pm-147	1.6E+0	1.8E-2	8.0E-7
Ra-226	6.7E-1	5.6E+2	2.8E-4

year) would be 468 person-rem. Since these watches have an estimated lifespan of seven years, the annual collective total body dose to users as a result of tritium luminous watches is about 3275 person-rem.

In addition to users, a number of other persons will also be exposed to tritium from watches. McDowell-Boyer and O'Donnell have performed a dose assessment in which "nonusers" were exposed, both in the office and in the home, to tritium air concentrations caused by watches worn or carried by their owners (McDowell-Boyer 78a). The exposure scenario chosen by these authors in their assessment is presented in Table 2.14, as are the results of their calculations. According to this scenario, the population dose to the nonusers exceeded the collective user dose by a considerable margin. If the nonuser exposure scenario of McDowell-Boyer and O'Donnell is applied to the present study, a nonuser population dose of about 2400 person-rem would result.

The population dose resulting from use of tritium luminous clocks was also calculated by McDowell-Boyer and O'Donnell. They assumed that 75 percent of the distributed clocks were located in bedrooms, 20 percent were used elsewhere in the home, and the remaining five percent were used in offices. It was further assumed that bedroom clocks exposed four persons for 2920 hours per year, other home clocks exposed four persons for 8760 hours per year, and the office clocks exposed four persons for 2000 hours per year. The resultant population dose calculated by these authors was 520 person-rem. If the method of McDowell-Boyer and O'Donnell is applied to the annual clock distribution assumed in this study, (5.3×10^5), and assuming a seven-year product lifespan, an annual population dose of 175 person-rem is calculated.

Gaseous Tritium Light Sources

The external dose rate from a gaseous tritium timepiece is due almost entirely to the production of bremmstrahlung within the device. This is evident from the fact that the maximum range of the 18 keV beta particle emitted in tritium decay is only 0.6 mg/cm^2. Measurements on thin-walled tubes (about 0.1 mm borosilicate glass) have shown dose rates at around 100 mrad/hr per Ci at the surface, while surface measurements of thicker tubes (1 mm glass with an outer coating of 2-3 mm plastic) show values of about 1 mrad/hr per Ci. The effective energy of the bremmstrahlung has been found to be in the range of from 8 to 14 keV (NEA 73). Bremmstrahlung dose rates in air at 5 cm from a variety of

Table 2.14. Assessment of Doses to Nonusers from Tritium Luminous Compount Watches(1).

Location or Activity (volume, cm^3)	Ventilation Rate (air changes/hr)	Exposure Duration (hr/yr)	No. Persons Exposed per Timepiece	Probability of Exposure	Population Dose (person-rem/yr)
Office (1.8×10^7)	1	2000	3	0.12	564
Home (3.4×10^8)	1	5840	3	0.88	584
					1148

tritium tube configurations have been measured by the British National Radiological Protection Board (Wilkins 79). These results are presented in Table 2.15, and indicate ranges of dose rates from 0.06 to 0.46 mrad/hr. Measurements on intact gaseous tritium timepieces have been reported as being less than 0.1 mrad/hr at 1 cm in air (NEA 73), and as not significantly above background radiation levels (Wilkins 79).

According to O'Brien (1978), the external dose factor for converting exposure to effective whole body dose is 0.03 rem/R for 20 keV photons. Since the attenuation of 10 keV photons in tissue is greater by a factor of 10 or so than that for 20 keV photons, it is evident that the effective whole body dose from tritium-produced bremmstrahlung will be negligible, and only the skin dose determination warrants assessment. Dose rates to the skin are calculated using the dose factor for 20 keV photons (0.4 rem/hr), since data are not available for 10 keV photons (O'Brien 78). It should be remembered, however, that the actual skin dose will be considerably lower due to the increased attenuation of the lower energy photons by the epidermis, clothing, etc.

If an annual distribution of 2×10^6 watches and a product lifespan of eight years are assumed, the skin dose to a user can be estimated using an average exposure rate value of 0.1 mR/hr. Due to the extremely large cross-section of steel for low energy photons, virtually none of the bremmstrahlung will escape the timepiece through the steel backing, even if the steel layer is very thin. (For example, the factor by which 15 keV photons are attenuated by 2 mm of steel is approximately 10^{40}.) Thus, only the external radiation emanating from the front of the watch needs to be considered. Assuming that the fraction of time a wearer is exposed to the watch front is 0.5, and that the average watch-to-skin distance is 20 cm, the skin dose to the wearer would be:

$$D_{sw} = \frac{(0.1\ \text{mR-cm}^2/\text{hr})(8760\ \text{hr/yr})(0.4\ \text{mrem/mR})(0.5)}{(20\ \text{cm})^2}$$

$$= 0.4\ \text{mrem/yr}$$

This assessment assumes that the watch is worn continuously (8760 hours) throughout the year. If an average usage time of 16 hours per day day (5840

Table 2.15. Bremmstrahlung Dose Rates in Air at 5 cm From Gaseous Tritium Light Sources(1).

Manufacturer	Nominal Activity (mCi)	No. of Samples	Range of Dose Rates in Air (mrad/hr)
A	50 unmounted	6	0.15 - 0.21
	50 mounted in sets of 4	2	0.15 - 0.16
	85 mounted in sets of 2	2	0.06
	80 unmounted	10	0.18 - 0.24
	100	10	0.33 - 0.45
B	60 unmounted (length 23.32mm)	18	0.27 - 0.35
	60 unmounted (length 15.29 mm)	9	0.09 - 0.14
C	200 mounted in sets of 1	5	0.24 - 0.37
D	46 (unmounted)	6	0.10 - 0.12
	100 (unmounted)	5	0.30 - 0.46
	58 (unmounted)	2	0.2

(1) From Wilkins, 1979

hours per year) is assumed for the entire watch-wearing population, the collective skin dose would be:

$$D_{sp} = \frac{(0.4\text{ mrem/yr})(16\times10^6\text{ persons})(5840\text{ hr/yr})(1\text{ rem})}{(8760\text{ hr/yr})(10^3\text{ mrem})}$$

$$= \text{or } 4.3\times10^3\text{ person-rem.}$$

Using a dose conversion factor of 0.1 mrem/mR (O'Brien 78), and assuming an average unobstructed watch-to-organ distance of 30 cm, the gonadal dose to a male wearer would be:

$$D_{tw} = \frac{(0.1\text{ mR-cm}^2\text{/hr})(8760\text{ hr/yr})(0.1\text{ mrem/mR})(0.5)}{(30\text{ cm})^2}$$

$$= 0.05\text{ mrem/yr.}$$

The collective gonadal dose to ten million male wearers (62.5 percent of the total wearers), assuming an average usage time of 5840 hours per year, would be:

$$D_{tp} = \frac{(0.05\text{ mrem/yr})(10^7\text{ males})(5840\text{ hr/yr})(\text{rem})}{(8760\text{ hr/yr})(10^3\text{ mrem})}$$

$$= 3.3\times10^2\text{ person-rem.}$$

Once again, it should be emphasized that these doses must be considered as upper level estimates since in reality much more attenuation would occur than is given credit for in the calculations. Doses to the ovaries of female wearers would be virtually zero since the ovaries are located at a tissue depth of several centimeters.

McDowell-Boyer and O'Donnell (1978b) have estimated the population dose to wearers of gaseous tritium watches, as well as to bystanders, as a result of tritium gas leakage. They assumed that tritium leaks from each watch at the rate of 50 nCi/day, that all leaked tritium is in the form of water vapor, and that the tritium disperses rapidly and uniformly throughout the volume of airspace.

The results of their calculations are presented in Table 2.16. As those authors are careful to point out, the assumptions used regarding the rate of tritium leakage from watches and the chemical form of the released tritium significantly affect the dose estimates.

Knapton and Comer (1971) estimate that from 6 to 52 percent of tritium in light tubes should be considered as HTO or T_2O. Their findings, however, are contested by Guthrie and Coates (1972), whose measurements have indicated that less than 0.1 percent of the gas should be considered HTO or T_2O. Wilkins and Dixon (1978) have determined the water-soluble fraction of many tubes by breaking the tubes under distilled water which had previously been flushed with a mixture of nitrogen (98 percent) and hydrogen (2 percent) followed by liquid scintillation counting of the solution. Their results indicated that between 1.5 and 4.2 percent (average 2.8 percent) of the released gas was retained as water soluble tritium. While there is no way to resolve these wide differences in reported values, it is certainly reasonable to assume that the fraction of released tritium which should be treated as water vapor is significantly less than unity.

According to Wardaszko, et al. (1978), the buildup of HTO from HT in an unventilated area by the autocatalysis mechanism (i.e., the oxidation of HT molecules stimulated by tritium beta particles) is described by:

$$dA_{HTO}/dt = kA_{HT}$$

where k is the HTO buildup factor, a constant parameter with a numerical value which is assumed to be 0.0018 Ci_{HTO}/Ci_{HT}-sec. These authors evaluated The doses resulting from HT and HTO subsequent to a rupture of a 1 Ci tritium light tube and concluded that, for a room volume of 50 m^3 and a ventilation rate of 0.25 air changes per hour, the dose due to HT is predominant for the first eight hours. For an infinite exposure period under these conditions, the ratio of the dose from HTO to that from HT is only about 1.4. This is surprising in light of the fact that the risk associated with exposure to HT is usually assumed to be insignificant in comparison to the HTO dose.

Table 2.16. Dose Estimates to Wearers of 10^7 Gaseous Tritium Light Source Watches and to Bystanders(1).

Exposed Group	Number of Exposed Persons	Dose to Individual, mrem/year	Population Dose Person-rem/year
Wearers(2)	$1.2x10^6$	$2.4x10^{-2}$	29
Wearers(3)	$3.8x10^6$	$3.2x10^{-3}$	12
Wearers(4)	$5.0x10^6$	$4.8x10^{-3}$	24
Subtotal			65
Bystanders(5)	$3.6x10^6$	$2.1x10^{-2}$	75
Bystanders(6)	$2.6x10^7$	$3.2x10^{-3}$	84
Distant Persons(7)	$1.7x10^8$	0 to $2.6x10^{-6}$	1
Subtotal			160
Total			225

(1) From McDowell-Boyer, 1978b; does not include dose from bremmstrahlung
(2) Assumes 2000 hr/yr in office and 5840 hr/yr in home
(3) Assumes 2000 hr/yr outdoors and 5840 hr/yr in home
(4) Assumes 8760 hr/yr in home
(5) Assumes 2000 hr/yr in office
(6) Assumes 5840 hr/yr in home
(7) Assumes 8760 hr/yr exposure to H-3 air concentrations of 0 to $9.7x10^{-12}$ uCi/cm^3

The HTO buildup constant used by Wardaszko, et al. describes HTO generation by autocatalysis only. Little is known regarding the HTO formation rate in air by isotope exchange. It is believed, however, that the exchange process is slow in comparison to the total exposure time, obviating the need for evaluation (Wardaszko 78). If this is the case, the dose estimates of McDowell-Boyer and O'Donnell are probably high by a considerable margin. For the purpose of dose estimation in the present study, the doses determined by McDowell-Boyer and O'Donnell will be scaled directly to the timepiece distribution assumed here (i.e., 1.6×10^7 timepieces). This results in a collective dose of 360 person-rem.

Promethium-147

Doses to users of watches and clocks containing Pm-147 are due almost entirely to bremmstrahlung production within the timepiece. McDowell-Boyer and O'Donnell (1978a) have estimated external doses resulting from Pm-147 timepiece use, and have arrived at a collective whole body dose of 345 person-rem from four million watches and six million clocks (Table 2.17). If it is assumed that the timepieces contain four ppm Pm-146, which emits 0.453 and 0.75 MeV gamma-rays, the collective dose is estimated to be about 360 person-rem. Since this report assumes a steady-state distribution of 3.7 million watches and 4.4 million clocks, the collective dose can be estimated at about 315 person-rem for isotopically pure Pm-147 timepieces, or about 325 person-rem for four ppm Pm-146. The collective doses to specific organs are listed in Table 2.18.

Radium-226

The major disadvantage associated with the use of radium-activated timepieces is the emission of comparatively high levels of penetrating radiation in the form of gamma-rays from Ra-226 and daughter product decays. Consequently, numerous studies have been undertaken in an effort to quantify the external doses to wearers of radium-activated watches, as well as to persons in homes containing radium clocks. For a summary of these investigations, the reader is referred to the reviews by Robinson (1968), Moghissi and Carter (1975), and Moghissi, et al. (1978).

Table 2.17. Collective Doses to Users of Pm-147 Wristwatches and Clocks(1).

Position or Location of Timepiece for Exposed Group	No. of Group Members	Population Dose (Person-rem/yr)	
		Pm-147	Pm-147+146
<u>Wristwatches</u>			
Outside of wrist (24 hr)	1.0E+6	5.0	7.8
Outside of wrist (16 hr)	1.0E+6	2.8	4.4
Inside of wrist (24 hr)	1.0E+6	210.0	210.0
Inside of wrist (16 hr)	1.0E+6	90.0	95.0
		310.0	320.0
<u>Clocks</u>			
In bedroom	21.6E+6	28.0	30.0
In home	5.8E+6	3.6	3.9
In office	1.4E+6	3.6	3.8
		35.0	38.0

(1) From McDowell-Boyer, 1978a

Table 2.18. Collective Organ Doses From Exposure to 3.7 Million Watches and 4.4 Million Clocks Containing Pm-147(1).

Organ	Dose (Organ-rem)[1]
Total Body	1.9E+2
Liver	1.6E+2
Bone	2.8E+2
Red Marrow	2.8E+2
Lungs	1.7E+2
G. I. Tract	1.5E+2
Gonads	1.8E+2

(1) Assumes 0 ppm Pm-146

If the precise amount of radium present in a timepiece is known, the radiation exposure to the user can be calculated with relative ease and accuracy. However, as Seelentag and Schmier (1963) have pointed out, a considerable range of activities in timepieces exists, especially in older timepieces. Klein, et al. (1970) have reported an average activity for military pocket watches of 1.0 uCi, whereas Moghissi and Carter (1975) estimated an average activity of 0.5 uCi. Our data are in close agreement with the latter estimate, indicating an average activity of about 0.4 uCi.

Seelentag and Schmier (1963) have summarized gonadal and forearm dose estimates of several different authors. These estimates are presented in Table 2.19. Additional estimates of the genetically significant dose (or GSD, which may be defined as the product of a collective gonadal dose and the total number of future children expected from the exposed persons) were made by McCarthy and Mejdahl (1963). Those authors evaluated data on age and sex distribution of radium watch users in Denmark, and concluded that the GSD to the Danish population from radium wrist watches was about 1.2 percent of the GSD from the natural background. The exposure value used by McCarthy and Mejdahl was 103 mR/yr per uCi of radium. This corresponds to a collective gonadal dose of 63 mrem/yr (24 mrem/yr for female wearers and 39 mrem/yr for males) per uCi of radium in use.

If it is conservatively assumed that no attenuation occurs within the watchcase, that the average watch-to-organ distance is 30 cm, and that the average usage time is 16 hours per day, the gonadal dose to the male population can be calculated as:

$$D_{tw} = \frac{(0.825\ \frac{\text{R-m}^2}{\text{hr-Ci}})(5840\ \text{hr/yr})(0.76\ \text{rem/R})(10^{-6}\ \text{Ci/uCi})}{(0.3\text{m})^2}$$

$$= 0.04\ \text{rem or } 40\ \text{mrem/uCi}$$

For females, the dose to the ovaries would be:

$$D_{ow} = \frac{(0.825\ \frac{\text{R-m}^2}{\text{hr-Ci}})(5840\ \text{hr/yr})(0.46\ \text{rem/R})(10^{-6}\ \text{Ci/uCi})}{(0.3\text{m})^2}$$

$$= 0.025\ \text{rem or } 25\ \text{mrem/uCi}$$

Table 2.19. Radiation Doses From Radium Wristwatches of Different Activities(1).

Author	Activity (ug Radium Equivalent)	Annual Gonadal Dose(2)		Annual Dose To Forearm(2)	
		mrems	Percent of Natural Radiation(3)	mrems	Percent of MPD(4)
Seelentag & Klotz, 1959	0.04	5.3	5.3	2,600	34
Seelentag & Schmier, 1961	0.014	1.8	1.8	850	11
(minimum value)	0.002	0.3	0.3	122	2
(maximum value)	0.885	115	115	54,000	720
Flatby & Koren (day & night use)	0.19	23	23	--	--
Flatby & Koren (daytime use only)	0.19	4.4	4.4	--	--
Joyet (men's watches)	0.14	9.8	9.8	--	--
Joyet (ladies' watches)	0.06	3.9	3.9	--	--
Joyet (maximum value)	4.5	310	310	270,000	3,600
Eikodd et al., 1961	1.0	108	108	--	--
Seelentag	1.0	130	130	60,000	800
Flatby, 24 hours	1.0	120	120	--	--
Flatby (days only: 16 hours)	1.0	23	23	--	--
Joyet (men's watches)	1.0	70	70	--	--
Joyet (ladies' watches)	1.0	65	65	--	--

(1) From Seelentag, 1963

(2) Doses of Seelentag and co-workers were calculated, and those of Flatby and Koren, as well as Joyet, were measured on wearers of watches which had up to 100 times the activity of normal illuminated dials.

(3) Background genetic radiation dose is taken as 100 millirems a year.

(4) Maximum permissible dose for the forearm (population = 7.5 rems = 7,500 mrems per year).

As can be seen, the results of the above calculations are in very close agreement with the gonadal doses estimated by McCarthy and Mejdahl. It should be remembered, however, that very few radium watches are currently being distributed in this country, and that most of the user exposure results from clocks. The assessment of the collective dose from exposure to radium clocks assumes the following:

(1) Ten million clocks (i.e., annual distribution = $1.25\text{x}10^6$; average life = 8 years) with an average activity of 0.5 uCi are located in homes.

(2) 75 percent of the clocks are located in bedrooms, exposing four persons for 2920 hours per year at an average distance.

(3) 20 percent of the clocks are located elsewhere in the home exposing four persons for 8760 hours per year at an average distance of two meters.

(4) Five percent of the clocks are located in offices exposing four persons for 2000 hours per year at an average distance of two meters.

(5) No credit is taken for attenuation by the clock glass, air, clothing, etc.

The collective exposure to this population resulting from radium clock usage would be:

$$(0.825\ \frac{\text{R-m}^2}{\text{hr-Ci}})(5\text{x}10^{-7}\ \text{Ci})\left[\frac{(7.5\text{x}10^6)(4)(2920\ \text{hr/yr}) + (2.0\text{x}10^6)(4)(8760\ \text{hr/yr}) + 4.0\text{x}10^5)(4)(2000\ \text{hr/yr})}{(2\text{m})^2}\right]$$

$$= 16{,}675\ \text{R/yr}$$

Specific organ doses to the population resulting from this scenario are given in Table 2.20.

Table 2.20. Collective Organ Doses from Exposure to 10^7 Radium Clocks.

Organ	Dose (Organ-rem)[1]
Total Body	9.5E+3
Liver	9.2E+3
Bone	9.2E+3
Red Marrow	8.7E+3
Lungs	9.2E+3
G. I. Tract	8.2E+3
Gonads	9.7E+3

(1) Dose conversion factors used are averages of values for males and females using an effective energy of 0.8 MeV for Ra-226 in equilibrium with its daughters (Kathren 76).

2.2.2.4 Product Disposal

Waste Collection

Some exposure of the municipal waste collection population to ionizing radiation will occur as a result of the collection of radioluminous timepieces in municipal refuse. The external doses resulting from waste collection are calculated according to the assumptions presented in Appendix D. The results of these calculations are presented in Table 2.21.

Although some internal exposure could result from the breakage of GTLS timepieces during waste collection, the difficulties involved in quantitatively estimating the resultant doses are such that no dose estimation is attempted. However, factors such as the relatively small number of these type of timepieces involved, and the rapid and extensive dilution of the leaked HT gas by the atmosphere, indicate that such exposures would be minimal.

Disposal In Landfills

The radiological impact resulting from the disposal of radioluminous timepieces in landfills is assessed according to the methodology and assumptions presented in Appendix D. It is assumed that ten percent of the timepieces disposed of in landfills have been previously incinerated and have released all of their radioactivity as a result. For GTLS timepieces, it is assumed that the integrity of one-tenth of the tritium tubes is breached during burial, and that all of the HT gas is immediately released. It will be conservatively assumed that 90 percent of the HT gas will escape by diffusion, and ten percent of the gas is assumed to be immediately oxidized and to behave the same as percolation. The remainder of the tritium tubes are assumed to leak at a rate of one percent per year (0.01 yr^{-1}), as are the tritium luminous compound, Pm-147, and radium timepieces.

The tritium gas which is released from the broken tubes and diffuses from the landfill will also result in exposures. Applying the results of McDowell-Boyer and O'Donnell (1978b) to this case, the population dose resulting from gaseous tritium diffusion would be about 90 person-rem, or about 1.5 percent of the dose resulting from the groundwater ingestion pathway (6240 person-rem).

Table 2.21. Total Body Doses to Waste Collection Population from Radioluminous Timepieces.

Timepiece	Total Body Dose (rem or person-rem)(1)	
	Average Individual	Population
H-3 Paint	NA(2)	NA
H-3 Gas	NA	NA
Pm-147(3)	3.5E-8	6.2E-3
Ra-226	1.4E-4	2.4E+1

(1) Assumes total body dose conversion factor of 0.56 rem/R for Pm-147 and 0.57 rem/R for Ra-266.

(2) NA = not assessed; external doses from tritium timepieces would be extremely low.

(3) Assumes 0 ppm Pm-146.

The results of the reference case calculations are presented in Table 2.22. For comparison, Table 2.23 shows how the results vary when the values for certain reference case parameters are changed.

Incineration

Since approximately ten percent of all municipal refuse is incinerated, it is assumed that a similar percentage of radioluminous timepieces are disposed of via the incineration route. It is also assumed that all of the timepiece radioactivity is released as a result of the incineration process. A listing of other assumptions, as well as a description of the dose assessment methodology, is presented in Appendix D. The results of this assessment are presented in Table 2.24.

2.2.2.5 Accidents

The evaluation of the radiological health impact of radioluminous timepieces would not be complete without an assessment of postulated accidental exposures. The two types of accidents considered are breakage of GTLS tubes and combustion of radioluminous timepieces in residential and warehouse fires. Since the probabilities of these events occurring as described in the scenarios are very low, accidental exposures should not be used as indices of risk without careful consideration of their likelihood. Rather, it is hoped that the results of these accident consequence calculations will illustrate the degree of hazard represented by the accident under consideration, and the relative risk posed by the different radionuclides.

GTLS Tube Breakage

It is highly desirable to have GTLS watches constructed in such a way that it is not easy to dismantle and gain access to the module, which consists of the liquid crystal display, the tritium tubes and their support, the printed circuit board, and the housing for these parts. Recent Anglo-Swiss standards recommend that access to the tritium tubes should only be possible by means of a special tool or procedure, or by severely damaging the module (Wilkins 79).

Table 2.22. Ingested Activity From Reference Case Landfill Disposal of Radioluminous Timepieces(1).

Timepiece	Amount Disposed of Annually (Ci)	Assumed Leach Rate (yr^{-1})	Migration Times (yr)			Total Activity Ingested (Ci)(2)
			Vertical	Horizontal	Total	
H-3 Paint	6.2E+3	1.0E-2	8.0E+0	1.1E+0	9.1E+0	1.2E+1
H-3 Gas	2.5E+5	(0.1) 1.0E+0 + (0.9) 1.0E-2	8.0E+0	1.1E+0	9.1E+0	7.8E+1
Pm-147	3.2E+1	1.0E-2	1.1E+4	1.6E+3	1.3E+4	0.0E+0
Ra-226	6.3E-1	1.0E-2	4.0E+3	5.6E+2	4.6E+3	1.9E-3

(1) See Appendix D for assumptions and methodology

(2) Activity ingested over all time

Table 2.23. Ingested Activity as a Function of Various Parameter Values for Landfill Disposal of Radioluminous Timepieces.

Timepiece	Parameter(1)	Parameter Value		Activity Ingested (Ci)(2)	
		This Case	Reference Case	This Case	Reference Case
H-3 Paint	R_t	1.0E-1 yr^{-1}	1.0E-2 yr^{-1}	5.4E+1	1.2E+1
	V_g	1.6E+3 m/yr	9.0E+2 m/yr	1.2E+1	
	d	1.0E+2 m	1.0E+3 m	1.3E+1	
	D	0.0E+0 m	1.5E+1 m	1.9E+1	
	all of the above(3)	as above		8.7E+1	
H-3 Gas	R_t	1.0E+0 yr^{-1}	(0.1) 1.0E+0 yr^{-1} + (0.9) 1.0E-2 yr^{-1}	3.2E+3	7.8E+1
	V_g	1.6E+3 m/yr	9.0E+2 m/yr	1.9E+3	
	d	1.0E+2 m	1.0E+3 m	2.0E+3	
	D	0.0E+0 m	1.5E+1 m	3.0E+3	
	all of the above(3)	as above		5.3E+3	
Pm-147	R_t	1.0E-1 yr^{-1}	1.0E-2 yr^{-1}	0.0E+0	0.0E+0
	K_i	1.4E+2	1.4E+3	0.0E+0	
	V_g	1.6E+3 m/yr	9.0E+2 m/yr	0.0E+0	
	d	1.0E+2 m	1.0E+3 m	0.0E+0	
	D	0.0E+0 m	1.5E+1 m	0.0E+0	
	all of the above(3)	as above		2.0E-2	
Ra-226	R_t	1.0E-1 yr^{-1}	1.0E-2 yr^{-1}	2.0E-3	1.9E-3
	K_i	5.0E+1	5.0E+2	1.1E-2	
	V_g	1.6E+3 m/yr	9.0E+2 m/yr	1.0E-2	
	d	1.0E+2 m	1.0E+3 m	2.3E-2	
	D	0.0E+0 m	1.5E+1 m	1.1E-2	
	all of the above(3)	as above		1.4E-2	

(1) See Appendix D for meaning of symbols

(2) Activity ingested over all time

(3) Represents "worst case" scenario

Table 2.24. Fify-year Inhalation Dose Commitments from Incineration of Radioluminous Timepieces(1).

Category	Activity Released (Ci/yr)	Activity Inhaled (uCi/yr)	Fifty-year Dose Commitment (Organ-rem)					
			Total Body	Liver	Bone	Lungs	G. I. Tract	Gonads
H-3 Paint								
Maximum individual	-	2.9E-2	2.3E-6	2.3E-6	2.3E-6	2.3E-6	2.3E-6	2.3E-6
Population around one incinerator	2.1E+0	1.0E+3	8.4E-2	8.4E-2	8.4E-2	8.4E-2	8.4E-2	8.4E-2
Total population	6.2E+2	3.1E+5	2.5E+1	2.5E+1	2.5E+1	2.5E+1	2.5E+1	2.5E+1
H-3 Gas								
Maximum individual	-	1.2E-1	9.3E-4	9.3E-4	9.3E-4	9.3E-4	9.3E-4	9.3E-4
Population around one incinerator	8.3E+1	4.2E+4	3.4E+0	3.4E+0	3.4E+0	3.4E+0	3.4E+0	3.4E+0
Total population	2.5E+4	1.3E+7	1.0E+3	1.0E+3	1.0E+3	1.0E+3	1.0E+3	1.0E+3
Pm-147								
Maximum individual	-	1.5E-5	4.8E-8	1.2E-7	1.3E-6	9.9E-7	8.2E-8	4.8E-8(2)
Population around one incinerator	1.1E-2	5.4E-1	1.8E-3	4.3E-3	4.6E-2	3.6E-2	3.0E-3	1.8E-3(2)
Total population	3.2E+0	1.6E+2	5.3E-1	1.3E+0	1.4E+1	1.1E+1	9.0E-1	5.3E-1(2)
Ra-226								
Maximum individual	-	2.9E-7	5.1E-5	1.4E-9	7.5E-5	7.1E-5	1.7E-7	5.1E-5(2)
Population around one incinerator	2.1E-5	1.1E-2	1.9E+0	5.3E-5	2.7E+0	2.6E+0	6.3E-3	1.9E+0(2)
Total population	6.3E-3	3.2E+0	5.9E+2	1.5E-2	7.9E+2	7.9E+2	1.9E+0	5.9E+2(2)

(1) See Appendix D for assumptions and methodology

(2) No inhalation dose conversion factor (DCF) data available; DCF for gonads assumed to be equal to that for total body

In addition, rigid integrity test specifications are currently applied by the British National Radiological Protection Board to GTLS timepieces. The program carried out by that organization consists of pressure, temperature, vibration, drop, puncture, and impact testing of complete GTLS watches. Results of the testing program indicate that, in general, GTLS watches perform creditably, although shortcomings in construction were relatively common during the initial phase of testing (Wilkins 78).

As in the case of chronic tritium leakage, the magnitude of the doses resulting from GTLS tube breakage are greatly influenced by factors such as the rate of oxidation of HT gas and the room ventilation rate. This assessment will assume that all of the released tritium gas is in the form of tritiated water vapor (HTO or T_2O), and that the residential ventilation rate is either 0.25 or 1.0 air changes per hour. It will also be assumed that the breathing rate of individuals present is 1.2 m^3/hr, and that an amount of tritium equal to that inhaled will be absorbed throught the skin.

The average concentration of HTO in a room (C in uCi/cm^3) following the breakage of a GTLS timepiece containing $2.0x10^5$ uCi of tritium would be:

$$C = A\,(1-e^{-kt})/Vkt$$

where

A = activity released at t = o (uCi)

k = ventilation rate (volume changes/hr)

t = time over which C is averaged (hr)

V = volume of air (cm^3)

Since breakage can occur in a wide variety of building environments, and since any number of persons may be present for any length of time, the results are presented in tabular form (Table 2.25.). This method facilitates the easy identification of doses associated with assumed accident conditions.

Table 2.25. Individual Doses (rem per person present) Resulting from Tritium Tube Breakage in Enclosed Space.(1)

Building Volume (cm^3)	Ventilation Rate (changes/hr)	Time (hr)(2)		
		1	8	24
4.0E+7(3)	0.25	8.6E+1	3.4E+0	3.8E+0
	1.00	6.1E-1	9.6E-1	9.5E-1
5.0E+8(4)	0.25	6.8E-2	2.6E-1	3.0E-1
	1.00	4.9E-2	7.7E-2	7.7E-2
3.0E+9(5)	0.25	1.1E-2	4.4E-2	5.2E-2
	1.00	8.1E-3	1.3E-2	1.3E-2

(1) See text for assumptions

(2) Time person remains in room after accident

(3) 4.0E+7 cm^3 is representative of a small workshop

(4) 5.0E+8 cm^3 is representative of a residence

(5) 3.0E+9 cm^3 is representative of a small warehouse

Residential Fires

This assessment deals with the internal exposures to residents or firefighters as a result of firefighting, salvage, and cleanup operations associated with residential fires involving radioluminous timepieces. Fifty-year dose commitments to persons involved are estimated on the basis of the following assumptions:

(1) All of the radioactivity in the timepiece is released during combustion.

(2) Pm-147 and Ra-226 are released as respirable particulates while tritium is released as HTO.

(3) The volume of house air is 5.0×10^{8} cm^{3}.

(4) The breathing rate of involved personnel is 1.2×10^{6} cm^{3}/hr.

(5) An amount of HTO equal to that inhaled is absorbed through the skin.

(6) Four residents are exposed to the released radioactivity for 0.25 hours before leaving the building, and four firefighters are exposed for eight hours.

Since it is unreasonable to assume that firefighters would enter a burning building without first ventilating it, a considerable number of air changes will occur during their exposure period, resulting in a significant reduction in the concentration of airborne radioactivity. It is also unlikely that firefighters would enter the involved building without respiratory protection or that the building would not be well-ventilated during salvage and cleanup operations. This assessment assumes that the net effect of the above and other factors is a hundred-fold reduction in HTO absorption and HTO, Pm-147 and Ra-226 inhalation. The results of this assessment are presented in Table 2.26. Dose commitments are in terms of person-rem or organ-rem per timepiece involved.

Warehouse Fires

The storage of large quantities of radioluminous timepieces in manufacturer or distributor warehouses represents a potential exposure hazard in case of fires. Since relatively large quantities of radioactive material could be released, and since it is possible that firefighters or other involved personnel may be unaware of the radiological hazard which exists, a conservative estimate of the consequences of such an accident is warranted.

Table 2.26. Fifty-year Dose Commitments Resulting from Combustion of Radioluminous Timepieces in Residential Fires(1).

Timepiece	Group Exposed	Amount Inhaled or Absorbed (uCi)	Dose Commitment (person-rem or organ-rem)(2)					
			Total Body	Liver	Bone	Lungs	G.I. Tract	Gonads
H-3 Paint	Residents	8.9E+0	7.1E-4	7.1E-4	7.1E-4	7.1E-4	7.1E-4	7.1E-4
	Firefighters	2.8E+0	2.2E-4	2.2E-4	2.2E-4	2.2E-4	2.2E-4	2.2E-4
H-3 Gas	Residents	9.6E+2	7.7E-2	7.7E-2	7.7E-2	7.7E-2	7.7E-2	7.7E-2
	Firefighters	3.1E+2	2.5E-2	2.5E-2	2.5E-2	2.5E-2	2.5E-2	2.5E-2
Pm-147	Residents	1.1E-1	3.5E-4	8.7E-4	9.2E-3	7.3E-3	6.1E-4	3.5E-4(3)
	Firefighters	3.5E-2	1.1E-4	2.8E-4	2.9E-3	2.3E-3	1.9E-4	1.1E-4(3)
Ra-226	Residents	1.2E-3	2.2E-3	5.8E-6	3.0E-1	2.9E-1	7.1E-4	2.2E-3(3)
	Firefighters	3.8E-4	6.9E-4	1.8E-6	9.6E-2	9.2E-2	2.3E-4	6.9E-4(3)

(1) See text for assumptions.

(2) Total dose to exposed group per timepiece involved.

(3) Assumes dose conversion factor for gonads same as that for total body.

As in the case of residential fires, the primary radiological hazard in warehouse fires involves the inhalation hazard (and, for tritium timepieces, absorption of HTO through the skin) represented by airborne radioactivity. This assessment is based on the following assumptions:

(1) The number of timepieces in storage at the time of the fire is 1,000.

(2) The warehouse volume is 3.0×10^9 cm^3.

(3) Firefighters are present in the building for a period of eight hours following the outbreak of the fire.

All other assumptions are the same as in the residential fire assessment. Organ doses resulting from this warehouse fire scenario are presented in Table 2.27. These doses are in terms of rem per person involved.

2.2.2.6 Storage in Homes

In addition to the exposure scenarios evaluated above, other situations may exist which result in personnel exposures. For example, some people may be reluctant to discard a watch even if it is broken or has reached the end of its useful life. In such cases, one or more timepieces may be stored in dressers or jewelry boxes for many years, continuing to cause a small amount of exposure without providing any commensurate measure of benefit. For Pm-147 timepieces, these exposures would be exceedingly small since there is little or no radionuclide leakage from these timepieces, and external radiation levels are low. Total body exposures from stored tritium timepieces would be similar to those incurred from use of the timepieces. Since total body exposures result from exposure to released HTO, the magnitude of these exposures is not highly dependent on proximity to the timepiece. In the case of radium, the total body dose received by a person spending 2920 hours per year two meters from a dresser (1 cm wood) containing two timepieces (0.5 uCi each) would be less than 1 mrem/yr.

While individual doses associated with timepiece storage in the home may be small, the large number of timepieces distributed in the past could result in a significant population dose from this practice. Also, as mentioned previously, the persons incurring these exposures are not receiving any commensurate benefit. These exposures should therefore be considered as undesirable and unwarranted.

Table 2.27. Fifty-year Dose Commitments Resulting from Combustion of Radioluminous Timepieces in Warehouse Fires.(1)

Timepieces	Amount Inhaled or Absorbed (uCi)	Dose Commitment (rem or organ-rem)(2)					
		Total Body	Liver	Bone	Lungs	G.I. Tract	Gonads
H-3 Paint	1.2E+2	9.6E-3	9.6E-3	9.6E-3	9.6E-3	9.6E-3	9.6E-3
H-3 Gas	1.3E+4	1.0E+0	1.0E+0	1.0E+0	1.0E+0	1.0E+0	1.0E+0
Pm-147	1.5E+0	4.6E-3	1.2E-2	1.2E-1	9.6E-2	7.9E-3	4.6E-3(3)
Ra-226	1.6E-2	2.9E-2	7.5E-5	4.0E+0	3.8E+0	9.6E-3	2.9E-2(3)

(1) See text for assumptions.

(2) Dose per person involved.

(3) Assumes dose conversion factor for gonads same as that for total body.

2.2.3 Non-Radiological Impact

There are no significant hazards of a non-radiological nature associated with the use of tritium, Pm-147, or radium in timepieces.

2.2.4 Summary of Impact

A summary of the total estimated health risk, in terms of possible cancer deaths or serious genetic defects, associated with quantities of radioactive material currently distributed in timepieces in the U. S. is contained in Table 2.28. Not surprisingly, the lifespan activity which represents the highest risk for most timepiece types is normal product usage. Gaseous tritium timepieces, however, are possibly an exception to this rule. The primary health impact from GTLS watches, if the conditions assumed in this assessment are reasonable, appears to be associated with landfill disposal of the timepieces. However, it must be remembered that it was assumed that ten percent of the tritium tubes were ruptured in the burial process. Considering the manner in which the tubes are contained within the timepiece module, this may substantially overestimate the real case. If breakage is neglected, the landfill disposal health risk would be reduced from 2.2 to 1.4 cases, a figure which is still higher than those associated with other lifespan activities.

Since wide variations exist with respect to numbers of the different types of timepieces in use, a meaningful comparative analysis must evaluate the total radiological health impact on a per curie or per timepiece basis. This comparison is presented in Table 2.29. Based on the scenarios assessed in this report, the timepiece which appears to possess the most desirable radiological health characteristics is the one utilizing Pm-147, followed in order by tritium luminous compounds, gaseous tritium light sources, and radium. On a per curie basis, GTLS timepieces score most favorably, followed in order by tritium luminous compounds, Pm-147, and radium.

The main conclusion which can be drawn from the analyses performed is that there is a small but positive health risk associated with the total number of radioluminous timepieces currently in distribution. Although the level of uncertainty is large, it appears as though an upper level estimate of the combined number of cancer deaths and serious genetic defects associated with the number of timepieces distributed annually (averaged over the last several years) is somewhere in the region of one to ten.

Table 2.28. Total Estimated Health Effects from Annual Manufacture, Use, and Disposal of Radioluminous Timepieces.

Activity	Cancer Deaths or Genetic Defect(1)			
	H-3 Paint	H-3 Gas	Pm-147	Ra-226
Manufacture(2)	3.7E-2	7.0E-3	NA(3)	5.6E-4
Distribution	1.9E-2	3.2E-3	2.9E-4	1.4E-4
Use	1.2E+0(4)	3.6E-2(5)	6.5E-2	3.3E+0
Waste Collection	NA	NA	2.1E-6	3.3E-5
Landfill Disposal	3.9E-1	2.2E+0	0.0E+0	2.3E-4
Incineration	1.0E-2	3.5E-1	8.1E-4	1.8E-1
Total	1.7E+0	2.6E+0	6.6E-2	3.5E+0

(1) See Appendix B for health effect risk factors

(2) Includes other aspects of timepiece processing such as storage, repair, and refinishing

(3) NA = not assessed

(4) 2.0E+0 if bystanders are included

(5) 1.3E-1 if bystanders are included

Table 2.29. Total Number of Estimated Health Effects from Radioluminous Timepieces as a Function of Activity and Timepiece Number.

Timepiece	Cancer Deaths or Genetic Defects(1)		
	Actual Estimated	Per Curie(2)	Per Timepiece(2)
H-3 Paint(3)	1.7E+0	1.1E-4	2.0E-7
H-3 Gas(3)	2.6E+0	6.5E-6	1.3E-6
Pm-147	6.6E-2	7.2E-4	3.3E-8
Ra-226	3.5E+0	5.6E+0	2.8E-6

(1) See Appendix B for health effect risk factors

(2) Per annual curie or timepiece distribution averaged over the last several years

(3) Does not include exposures from watch use to nonusers

2.3 ALTERNATIVES

2.3.1 Radiological Alternatives

The radionuclide which represents the main radiological alternative to tritium and Pm-147 is carbon-14 (C-14). This isotope has a number of characteristics which make it desirable for consumer product applications. Carbon-14 is a "soft" beta emitter (maximum energy = 0.155 MeV) with a very long physical half-life (5730 years), a short biological half-life (transportable C-14 is eliminated from bone with a half-life of 0.4 days, and from the total body with a half-life of 12.5 days), and a low radiological toxicity. According to Krejci and Zeller (1978), the high price of C-14 is probably the main factor in the lack of increase in demand for consumer product application of this isotope. However, reduction in the price of C-14 raw materials over the last three years has resulted in an increased use of C-14 luminous compounds in products such as rifle sights, instrument dials, and special timepieces. Thus, C-14 luminous compounds should be considered as major candidate materials for future applications in the commercial timepiece industry.

A possible alternative to tritium for use in liquid crystal display watches is krypton-85 (Kr-85). This isotope, which is already being used in other types of products such as electron tubes (see Section 9), is an inert gas with a half-life of 10.8 years and very low radiological toxicity. However, Kr-85 emits a 0.5 MeV gamma-ray in 0.4 percent of its decays, making it a potential external radiation hazard. It is not known at this time whether or not Kr-85 is being seriously considered as an alternative to tritium.

2.3.2 Non-Radiological Alternatives

For timepieces utilizing radioluminous compounds, a major alternative - one that is already in use to some degree - is the use of light-activated compounds. These materials exhibit the property of phosphorescence; that is, they are excited by incident energy (in this case, light), and slowly emit excess energy in the form of longer wavelength light. In contrast to fluorescence, the emission of light by phosphorescent material may continue for a considerable time after absorption of the incident energy.

The use of light-activated phosphors in timepieces offers several important advantages. The absence of radioactivity obviates the need for radiation protection measures during timepiece manufacture, and results in reduced material and administrative costs. Perhaps most importantly, there would be no exposure of timepiece users and bystanders to ionizing radiation. The application of these materials to widespread use is limited mainly by the inability of the material to phosphoresce for sufficiently long periods of time following exposure to light. Advances which increase the length of time during which the light yield remains above the minimum required level could eventually eliminate the need for radionuclides as activators.

The luminosity efficiency can be maximized by the proper choice of paint color. According to Krejci and Zeller (1978), the color of choice is natural green. The relative luminosity efficiency of several different colors follows (NEA 73):

Color	Relative Brightness
Natural Green	100
Yellow	95
White	75
Orange	55
Red	45
Blue	35

In recent years digital watches and clocks have become increasingly popular. The major non-radiological illumination alternative for digital timepieces is the use of light from electricity. This can be either through the use of batteries or, where electric clocks are concerned, from house current. Digital wristwatches usually employ either liquid crystals or light emitting diodes (LED) as a means of visual display. Timepieces utilizing LEDs do not require any additional light source for illumination, whereas LCD watches require "backlighting" if they are to be readable in the dark. The backlight can either be a GTLS or a small incandescent lamp which operates from a battery. Since the lamp would draw heavily from the battery if continuously operated, watches usually employ a spring return button to actuate the lamp so that light is produced only on demand. This method has the main disadvantage of requiring both hands in order to read the display in darkness.

2.4 COST AND RISK VERSUS BENEFIT

2.4.1 Cost

Little information exists with respect to cost differences between timepieces utilizing radionculide-activated luminous compounds and those that use either alternative means or no illumination. However, a reasonably accurate cost comparison can be made for LCD watches using tritium tubes versus those using incandescent lamps.

At the present time, there are three major types of batteries with widespread use in watches. These battery types are based on chemical systems using either silver oxide, mercury, or manganese dioxide. In addition, lithium-based batteries may become popular in the near future. The most commonly used battery type is the silver oxide battery, although recent dramatic increases in the price of silver are currently being reflected in commensurate battery price increases. Future price increases may significantly reduce the distribution of silver oxide batteries by making them less competitive on the open market. Watch batteries are currently priced from about $1 to $5, with the most popular types selling for $2 to $4.

The lifetime of an LCD battery depends on (1) the type and size of the battery (which in turn depends on the size of the module), (2) the average current draw of the chip, and (3) the current draw of the incandescent lamp. Battery capacities are usually rated at between 30 and 75 milliampere-hours (mA-hr), with the most popular types rated between about 35 to 40 mA-hr. If a tritium light tube or no illumination mechanism is used, a low drain battery is sufficient since the chip requires only a few microamperes (uA) of current. If an incandescent lamp is used, however, a high drain battery capable of providing 10 to 15 mA is necessary. The amount by which the battery lifetime is reduced by the use of lamps depends, of course, on how often the lamp is used. If it is assumed that the average battery has a capacity of 38 mA-hr and an average cost of $4, and that the average lamp draws 12 mA, the results depicted in Figure 2.1 are obtained. Thus, if the lamp is used for an average of ten seconds per day,

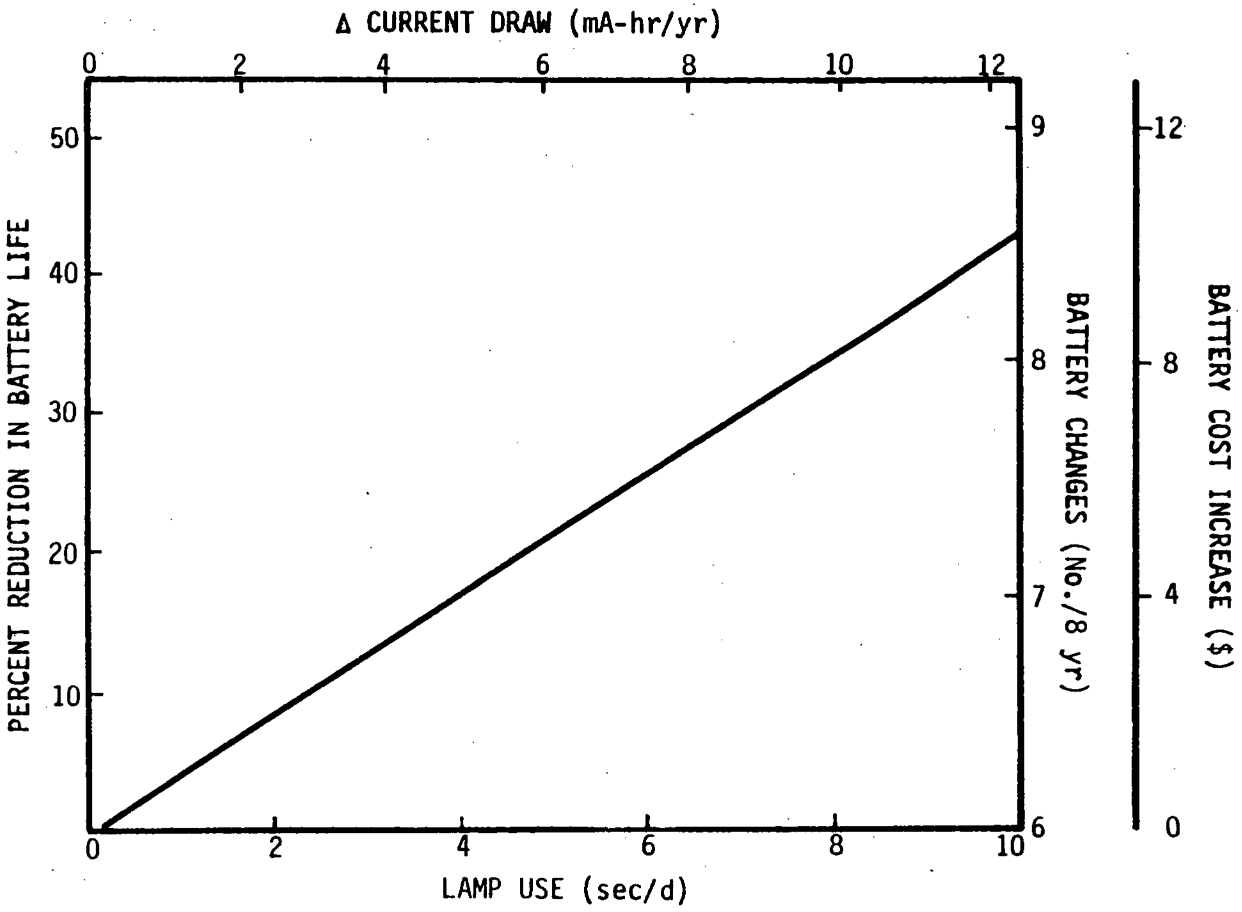

Figure 2.1. The Effect of Lamp Use on Watch Battery Life and Cost (see text for assumptions).

the additional cost to the consumer over the lifetime of the watch (assumed to be eight years) would be about $10.

It is difficult to accurately estimate the cost differences between incandescent and tritium lighted watches. Based on conversations with manufacturer representatives, it is estimated that manufacturing costs are on the order of $1.50 higher for tritium watches. This difference, however, does not include costs incindental to initiating and maintaining a radiation protection program. In this assessment, it will be assumed that the total manufacturing cost of a tritium tube LCD watch is $2 higher than an incandescent lamp watch, and that this increase is passed on to the consumer in the form of a $4 increase in the retail sales price. Under these conditions, if a person used the lamp about four seconds per day for the life of the watch, the cost increase arising from reduced battery lifetime would approximately equal the increase in cost of a tritium light tube watch. Although any estimate of the actual average lamp usage must be considered conjecture, a few seconds per day seems to be a reasonable estimate. Therefore, it can be concluded that tritium tube watches are cost-effective when compared to the principal alternative product, incandescent lamp watches.

The total dose resulting from the manufacture, distribution, use, and disposal of $2\text{x}10^6$ GTLS timepieces is estimated at about 925 person-rem, or about $4.6\text{x}10^{-4}$ rem per timepiece. The cost of reducing this dose is simply the cost difference between the GTLS and the alernative (incandescent lamp) timepiece. This can be represented in equation form as follows:

$$C_{DR} = \Delta C/\Delta D$$

where

C_{DR} = Cost of dose reduction ($/person-rem)

ΔC = Cost of alternative - cost of GTLS timepiece ($)

ΔD = Dose reduction resulting from use of alternative (person-rem)

As previously mentioned, the cost difference depends on how frequently and for how long the incandescent lamp is used. The cost of dose reduction, therefore, is also a function of lamp use. This is graphically depicted in Figure 2.2. As can be seen from these results, the cost of dose reduction rapidly reaches a signficant level when lamp usage exceeds four seconds per day.

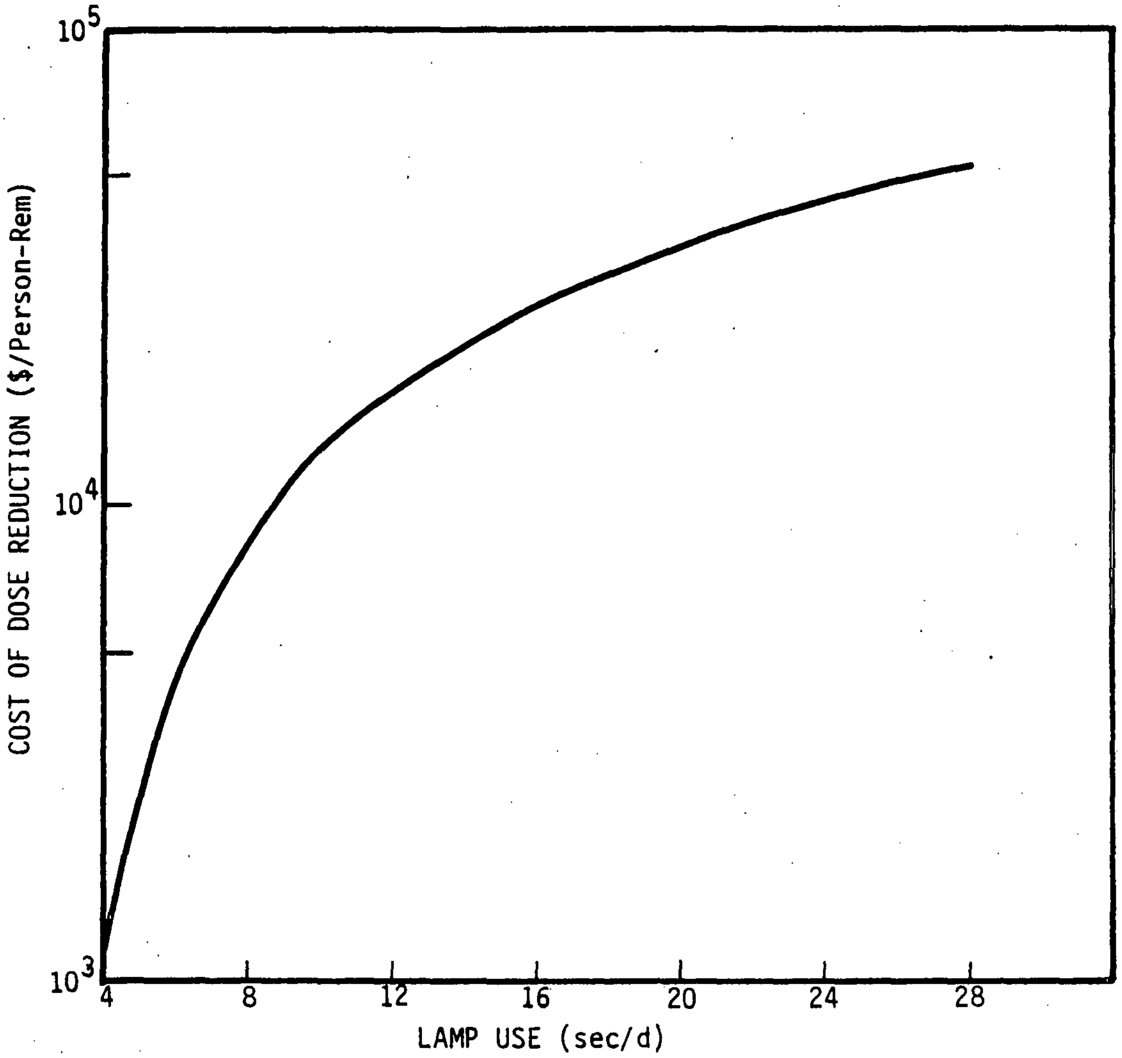

Figure 2.2. Cost of Dose Reduction for Use of Incandescent Lamps as an Alternative to Tritium Light Tubes in Watches.

When usage is less than that level, the cost of the alternative product is estimated as being less than the GTLS timepiece, and the cost of dose reduction is no longer a meaningful concept.

Estimates of the monetary value of avoiding the detriment possibly associated with a collective dose of one person-rem have been attempted in the past. Although such estimates are largely subjective in nature, values ranging from 10 to 1000 dollars per person-rem have been used. If 1000 dollars per person-rem is used as the point at which a dose reduction alternative becomes viable (10CFR50, Appendix I used this for total body person-rem accumulated within 50 miles of a nuclear power plant), the corresponding ΔC is found to be

$$\Delta C = (4.6\times10^{-4} \text{ rem/timepiece})(\$1000/\text{rem})$$

$$= \$0.46$$

From Figure 2.1, it can be found that a battery cost increase of $0.46 corresponds to a lamp usage rate of about 0.5 seconds per day. It was previously estimated that a lamp usage rate of four seconds per day is required to offset the initial cost difference between GTLS and incandescent lamp timepieces. Therefore, if the incandescent lamp timepiece is to meet the 1000 dollar per person-rem criterion for dose reduction, lamp use must be restricted to less than 4.5 seconds per day.

2.4.2 Product Analysis

The radiological health impact of the different timepiece types is summarized in Section 2.2.4. As previously mentioned, the most meaningful basis for comparison is the level of risk per timepiece. This can be evaluated in two ways:

(1) What is the level of risk associated with the amount of a particular radionuclide currently used in a timepiece? or

(2) What is the level of risk associated with the amount of a particular radionuclide required to produce the desired degree of illumination.

The latter question implies that the amount of activity used in timepieces is not always the minimum required for the desired level of illumination. Section 2.2.1

listed the amounts of tritium and Pm-147 required to produce the same level of brightness as 1 uCi of Ra-226. Using this light yield data as a basis for comparison, the relative level of risk associated with the same level of benefit can be calculated (see Table 2.30). This type of an evaluation appears to favor Pm-147, followed by tritium luminous compounds and radium. Gaseous tritium light source timepieces cannot be fairly compared to "hand and dial" type timepieces on this basis, since the tritium tubes are used exclusively in liquid crystal display timepieces, which have a larger area requiring illumination. There seems to be a tendency on the part of the GTLS industry to use the maximum amount of activity allowed by present day standards (i.e., 200 mCi). If this value is used as a basis for comparison, GTLS timepieces appear to have more risk associated with them than either the tritium luminous compound or Pm-147 timepieces, but not radium. Once again, however, one must bear in mind that the level of uncertainty associated with the estimated GTLS health impact is large. This follows from uncertainties in the land disposal model, which, for this particular timepiece, represents the major pathway of human exposure.

The radiological impact from radium timepieces would be significantly greater except for the fact that virtually all of the radium currently being distributed is used in clocks and not watches. Unlike tritium, radium is primarily an external hazard, and the level of exposure is highly dependent on proximity of the timepiece with respect to the user.

2.4.3 Alternatives Analysis

As previously mentioned, C-14 and Kr-85 are the major radiological alternatives to radionuclides currently being used for timepiece applications. C-14, like H-3 and Pm-147, is a pure beta emitter. The presence of external radiation, therefore, will be limited to that from bremmstrahlung production. The external doses to uses of C-14 timepieces would be lower than those from Pm-147 timepieces by a factor corresponding to the differences in the beta particle energies (0.155 MeV for C-14; 0.224 Mev for Pm-147) and the relative amounts of each nuclide used. Since C-14 has a much longer physical half-life than Pm-147, but only a slightly reduced beta energy, a lesser quantity is required for the production of a given amount of illumination over the useful life of the product.

Table 2.30. Relative Number of Health Effects from the Amount of Radionuclides Required to Produce the Same Level of Illumination.

Timepiece(1)	Activity Per Timepiece (uCi)	Relative No. of Health Effects
H-3 Paint	7,140(2)	4.6
H-3 Paint	3,570(3)	2.3
Pm-147	240	1.0
Ra-226	1	32.9

(1) GTLS timepieces are not compared since the area requiring illumination on these timepieces is much greater than on hand and dial type timepieces. If 2×10^5 uCi is used, the relative number of health effects is 7.6.

(2) Assumes light yield of 0.1 ucd/mCi, which is typical of paints used in recent years.

(3) Assumes light yield of 0.2 ucd/mCi, which has been reported as attainable in modern paints.

One of the main radiological health advantages of Pm-147 is the fact that promethium has a short half-life and a high affinity for soil particles. Thus, the uncontrolled burial of this nuclide does not constitute a radiological health hazard. The same cannot be said for C-14. It must be remembered, however, that C-14 is a naturally occurring radionuclide. It is being continuously produced through thermal neutron capture by nitrogen nuclei in the atmosphere. The current global inventory of C-14 from cosmogenic production and from the atmospheric testing of nuclear weapons is about $3x10^8$ Ci. C-14 is removed from the atmosphere by the exchange of radioactive CO_2 gas at the ocean's surface, making the ocean the dominant sink for C-14. As a result of this removal mechanism, only two or three percent of the C-14 currently resides in the atmosphere (NCRP 75).

The rate at which carbon migrates through a groundwater system is highly dependent on the chemical form of the molecule incorporating the C-14 atom. Soluble forms of C-14, however, are highly mobile and can be expected to move at a velocity close to that of the groundwater. If it is assumed that the C-14 migrates at the same rate as groundwater, an annual distribution of 92.5 Ci of C-14 (i.e., an amount equal to the Pm-147 distributed annually) would result in an ingestion of abount 2 Ci in the reference landfill disposal case. This compares unfavorably to Pm-147 (no activity ingested), but it is still below the level of activity which would be expected to result in one health effect. Thus, it is reasonable to conclude that there is certainly little difference in the radiological characteristics of timepieces employing either C-14 or Pm-147, and that C-14 is more desirable, from a radiological health standpoint, than H-3 or Ra-226.

Kr-85 has the main advantage of being an inert gas and, as such, is relatively non-toxic. Also, the leakage of Kr-85 from timepieces would be much less than for tritium. The higher beta energy (0.67 MeV vs 0.018 MeV for tritium) and 10.8 year half-life are also favorable characteristics for a candidate phosphor activator. The main disadvantage of Kr-85 is the presence of a 0.514 MeV gamma-ray in 0.4 percent of its decays.

The level of external radiation produced by Kr-85 decays is significant. For example, assuming that a user of a wristwatch containing 2 mCi of Kr-85 is exposed for 5840 hours per year at an average distance of 0.3 meters, the annual exposure to this person would be about 170 mR. This is certainly an unacceptably high level. It is not known whether or not the minimum required

Kr-85 activity could be made low enough to lower the external exposures to users to within acceptable levels.

Non-radiological alternatives such as LED's, incandescent lamps, or light-activated phospors do not have any significant health impact, radiological or otherwise, associated with their use. The main disadvantages of these alternatives are matters such as inconvenience, reduced efficiency or lifetime, and higher energy requirements and associated cost. The degree to which these alternatives will compete with radionuclide-activated phosphors for application in timepieces is not known at this time. A major advantage of such alternatives - one that significantly increases their attractiveness as timepiece illuminators - is the fact that a radiation protection program is not required for their manufacture or processing.

2.5 FUTURE USE

The future of the application of radioluminosity in timepieces is highly uncertain at this time. However, a few generalizations can be made:

(1) The increased popularity of digital (as opposed to hand and dial) timepieces will probably result in a reduction of the amount of radionuclide-activated phosphors used in timepieces in the future.

(2) A reluctance on the part of digital timepiece manufacturers or importers toward the use of GTLS devices will certainly have a significant impact on the future distribution of GTLS timepieces. Recent developments in the nuclear industry (most notably the Three Mile Island incident and the Arizona tritium contamination incident) have heightened the degree to which radiation is perceived by the public as an undesirable and risky entity. The perceived risk will undoubtedly influence the timepiece market, and the timepiece to be most affected will be the GTLS timepiece, partly because of the relatively large quantities of radioactive material used, and partly because these timepieces are labeled as containing radioactive material.

(3) The use of radium in timepieces has been on the decline for several years. This decline is expected to continue to the point where very little, if any, radium wil be used for timepiece application.

(4) Tritium and Pm-147 will continue to be the major radionuclides used in luminous compounds for the next few years. Reduced costs for C-14 compounds could make this radionuclide a viable alternative to tritium or Pm-147 in the coming years.

(5) It is not known to what extent non-radiological alternatives will compete with radionuclides as timepiece illuminators in the future. This competition will be largely influenced by future developments in light-activated phosphors where hand and dial timepieces are concerned, and in LED's and other illuminating mechanisms where digital timepieces are concerned.

2.6 SUMMARY

Radioactive materials are currently in widespread use for the purpose of timepiece illumination. Tritium and Pm-147 are the most commonly used radionuclides, although radium is still used to a limited extent. The use of radioactive materials in this application is expected to continue in the future, although the magnitude of this use is uncertain.

A detailed evaluation of the radiological health impact of radioluminous timepieces has been performed. The main conclusion which can be drawn from the analyses performed is that there exists a small but positive health risk associated with the total number of radioluminous timepieces currently in distribution. Although the level of uncertainty is large, it appears as though an upper level estimate of the combined number of cancer deaths and serious genetic defects associated with the number of timepieces distributed annually is somewhere in the region of one to ten. When it is considered that the current cancer mortality rate is about 400,000 deaths per year, and that the one to ten health effects represent an upper level estimate of the total health impact to be realized over all time, the radiological impact of radioluminous timepieces use appears to be very low, if not insignificant.

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3. STATIC ELIMINATORS

3.1 PRODUCT DESCRIPTION AND BACKGROUND

3.1.1 General Discussion

The buildup of static electric charges can be detrimental in various situations. Static electricity is simply defined as "electricity at rest" (Baumeister 78). Static electric charges may develop when different materials are rubbed together. Materials that pick up excess electrons become negatively charged, while materials that give up electrons become positively charged. If the materials are non-conducting, such as plastics, the charge will remain as a static charge. The presence of the static charge may lead to various problems. In industry, the risk of fire or explosion is increased when static electricity builds up and discharges near combustible materials such as plastics. In research, dust may collect on precision instruments, hindering their delicate operation. Other problems include the buildup of dust and foreign particles on film, lenses, and assorted photographic components. The buildup of static charges on high fidelity records can cause two problems: (1) dust and foreign particles are attracted to the record, resulting in possible record damage and sound reproduction distortion and (2) discharge of the static charges from the record to the phono cartridge may produce unpleasant noises, such as clicks, to be transmitted through the system to the listener.

Because of the need to eliminate static electricity, various devices and methods have been developed and used over the years. The use of alpha radiation to eliminate static electricity has been an established practice for many years in research and industry. The use by consumers has essentialy been limited to photographic and high fidelity needs, though other unknown uses may have been practiced. Alpha radiation is used to ionize air through which the static charges on nonconductive materials are dissipated.

At present, only one line of products containing radioactive material is manufactured and sold in the U. S. and used by consumers for static elimination. This line of products is advertised for use as static eliminators for photographic and high fidelity needs. The radioactive material used in these products is polonium-210 (Po-210), an alpha-emitting radionuclide. The following discussion deals with Po-210 static eliminators.

3.1.2 Product Description

The available static eliminators consist of a plastic handle with soft bristles and a removable cartridge located at the end of the handle. The removable cartridge contains the Po-210 source. Figure 3.1 illustrates the geometry and construction of a typical static eliminator.

The static eliminators are constructed of A.B.S. Cycolac plastic and 5052--aluminum. Two sizes are manufactured with the approximate dimensions of 5"x1-1/8"x5/8" and 5"x3"x3/4". The smaller device is said to nominally contain 200 uCi of Po-210, while the larger device nominally contains 500 uCi of Po-210. In the past, devices were manufactured that contained 50 and 125 uCi, but they were discontinued in 1970.

The Po-210 source is located about 6 mm behind an open aluminum grille in the cartridge. Since the mid-60's the source has consisted of ceramic microspheres which are resin-bonded to an aluminum backing. Prior to this, Po-210 foils were used (Webb 75). The Po-210 was held in place by a thin layer of gold or similar plating material. However, experience showed that loose Po-210 contamination often resulted when the foils were used and as a result their use was discontinued (NCRP 77).

The microspheres are ceramic bodies of spherical shape and controlled particle size. They have a bulk density of 2 g/cc and have a melting point in excess of 1500^{0}C. Sintering may occur above 1200^{0}C, but the sphere integrity is not lost. The spheres are physically insoluble in most organic and inorganic solvents (Lahr 63). An experimental study at Oak Ridge National Laboratory (ORNL) found the range of diameters of examined microspheres to be 20-60 um with an average of 38 um (Niemeyer 78). Figure 3.2 illustrates the frequency distribution found by Niemeyer, et al.

The ceramic microspheres are prepared by sorbing the Po-210 ions throughout a porous microspherical, inorganic ion exchanger consisting of titanium, zirconium, or hafnium phosphates. Following this process, the microspheres are heat treated to fix the Po-210. Following fixation, the microspheres are washed with strong mineral acids to remove any loosely bound

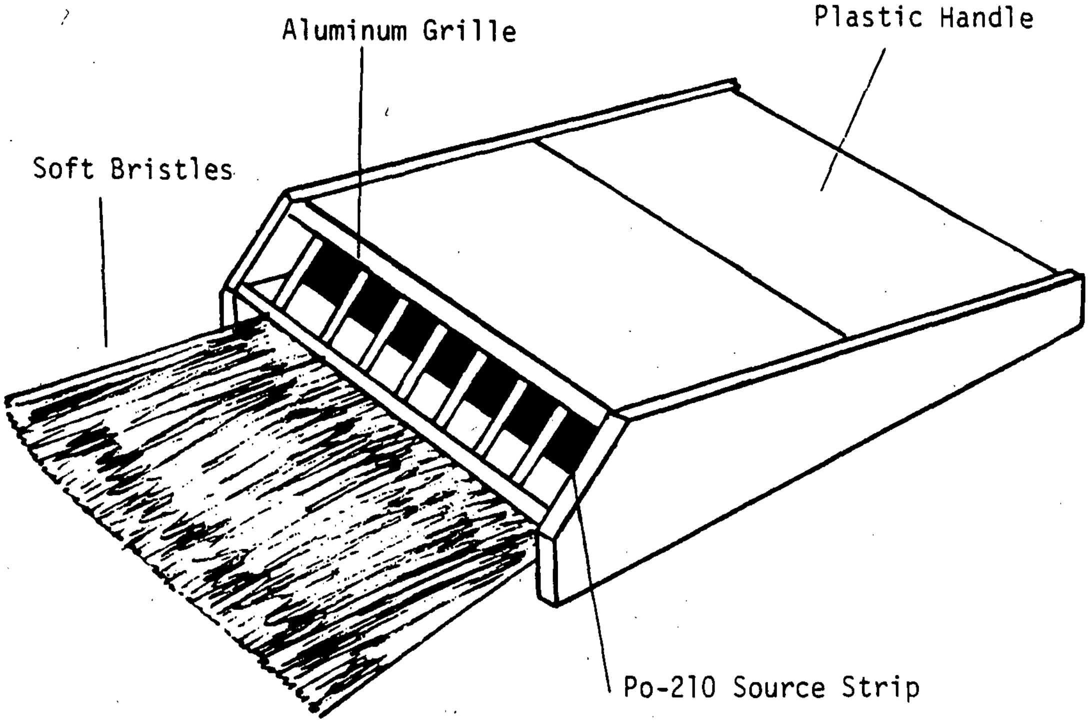

Figure 3.1. Geometry and Construction of Typical Po-210 Static Eliminator.

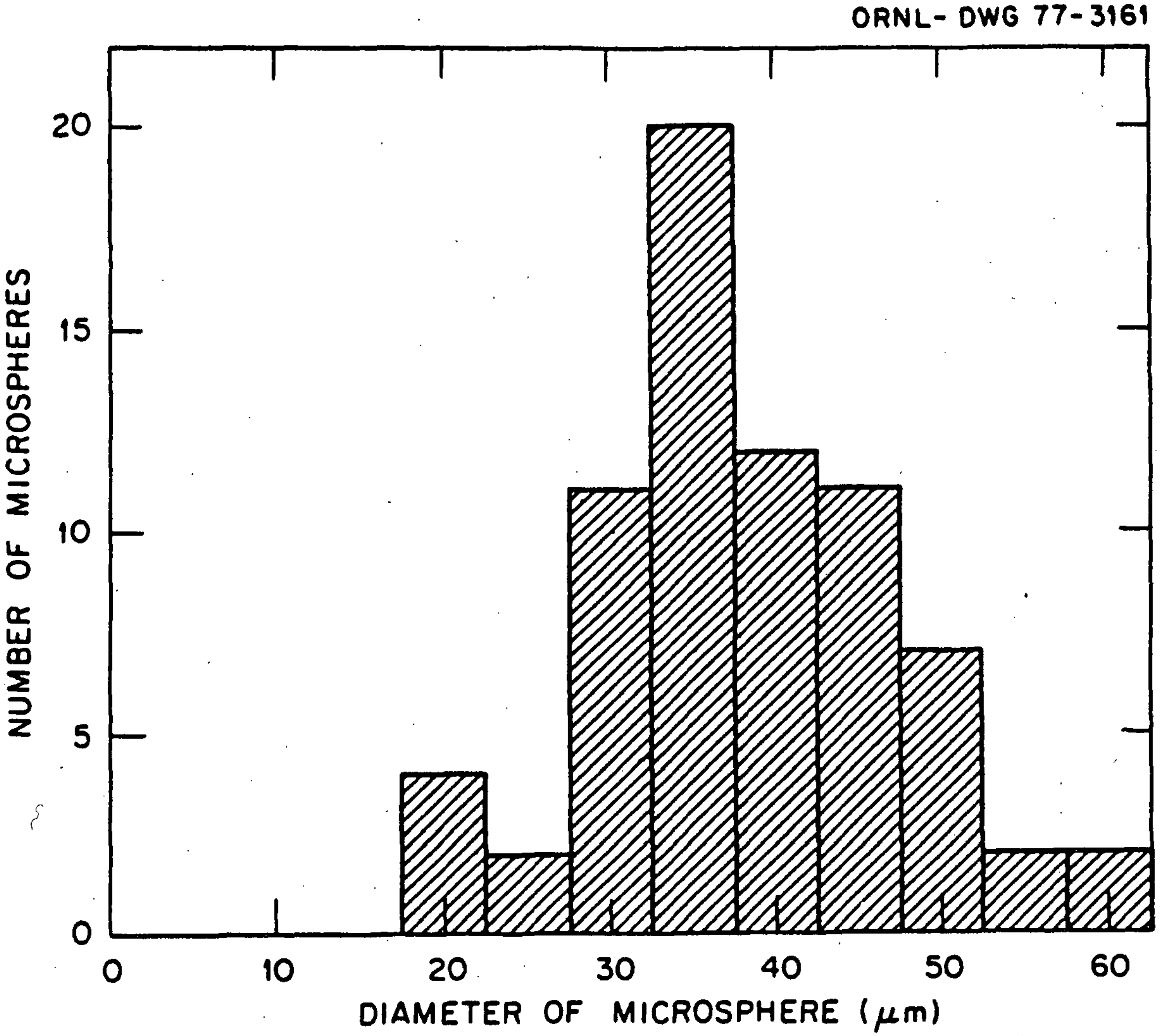

Figure 3.2. Size and Frequency Distribution of Po-210 Microspheres (Niemeyer 78).

activity (Lahr 63). The microspheres are then coated with 1-2 um of nickel and initially contain about 0.1 uCi of Po-210 each (Robertson 75).

All sources are assumed to remain effective for 18 months from the time of manufacture. Allowing six months for distribution and sale, the user is assured of one year of useful service (Webb 75). The expiration date, assumed to be 18 months after manufacture, is stamped on each source cartridge. The radioactive half-life for Po-210 is 138 days.

Tests conducted in England on the Po-210 content of nominal 200 uCi and 500 uCi static eliminators showed activities to be 280±30 uCi and 750±80 uCi, respectively. The expiration date on each static eliminator was 16 months after the date on which the measurements were made. Other measurements on similar units made 11-1/2 months before the expiration date indicated 180 uCi and 450 uCi, respectively (Webb 75).

The nominal activities of 200 and 500 uCi, therefore, apparently do not refer to the time of manufacture but to one year prior to the expiration date. Using this assumption, the amount of activity at the time of manufacture would be 1250 uCi and 500 uCi for the nominal 500 uCi and 200 uCi devices. At the end of 18 months, the values would be 80 uCi and 32 uCi, respectively. All radiological assessments of the devices should use the above numbers for consideration during the appropriate life span periods.

3.1.3 Distribution

Table 3.1 illustrates the distribution of the Po-210 static eliminators. The total activity is based upon the nominal activities given in the table. The maximum number of 500 uCi units distributed in one year was about 20,000. The maximum number of 200 uCi units was about 50,000, though not in the same year. The maximum nominal activity of Po-210 processed in one year was 18.8 Ci. As indicated earlier, only one company manufactures the units. Communication with the company indicated that until 1975 the consumer market had been basically the amateur photographer. At that time the 500 uCi units were introduced to the consumer through audio component stores. The consumers purchasing the device for this application are thought to be audiophiles who own expensive components.

Table 3.1. Distribution of Po-210 Static Eliminators.

Years	Number of Designated Units Distributed(1)				
	50 uCi(2)	125 uCi(2)	200 uCi(3)	500 uCi	Total Nominal Activity (Ci)
1956-60	61,800	11,300	0	6,600	7.8
1961-70	173,700	36,900	700	2,500	14.7
1971-79	0	0	365,400	106,100	126.1
Total	235,500	48,200	366,100	115,200	148.6

(1) Nominal Po-210 activity

(2) 50 uCi and 125 uCi units last manufactured in 1970

(3) 200 uCi units first manufactured in 1970

3.2 ENVIRONMENTAL IMPACT

3.2.1 Benefits

The benefit of Po-210 static eliminators has been discussed previously and is simply as the name implies, static elimination. For the consumer market no life-saving benefit or safety benefit is assumed. The device is used simply as a convenience. This is not the case for industrial uses, though, since static eliminators can be used and are used as safety devices, preventing fires and explosions. As a device for removing static electricity, the Po-210 units are found to perform quite well and in a manner favorable to the consumer. A discussion of their performance is included in Section 3.4.

3.2.2 Radiological Impact

Po-210 decays to stable lead-206 (Pb-206) via the emission of a 5.3 MeV alpha particle. The radiological data for Po-210 is given in Table 1.1. The 0.803 MeV gamma emission occurs only about 0.001 percent of the time according to Lederer (67). Appendix A contains the appropriate equations and conversion factors used to assess the radiological impact of Po-210.

This section assesses the total radiological health impact resulting from the manufacture, distribution, normal use, and disposal of Po-210 static eliminators. As in the assessment of other products, the assessment can only be considered to be a general estimate of the health risk associated with the devices.

3.2.2.1 Occupational Exposure

Communication with the only company manufacturing Po-210 static eliminators in the U. S. was made regarding the occupational exposure of workers in their plant. They stated that all their personnel submit urine specimens for analysis on a regular basis and that in over 25 years of checking, no Po-210 has ever been detected in the samples. Discussion of the potential hazards due to ingestion and inhalation of Po-210 is included in Section 3.2.2.3. Based upon the communication given above, no internal exposure of workers is assumed.

The external exposure due to the gamma radiation emitted during decay of Po-210 is quite small, as would be expected. From Appendix A, the total body dose rate, $D_{e,T}$, at one meter from one curie of Po-210 (no shielding) is found to be

$$D_{e,T} = \Gamma_i \, DCF'_{i,T}$$

where Γ_i = 4.9×10^{-6} R/hr at 1 m/Ci (see Table A.1)

$DCF'_{i,T}$ = 0.6 rem/R for 0.8 MeV gammas (see Figures A.1 and A.2)

Therefore, $D_{e,T}$ = (4.9×10^{-6} R/hr at 1 m/Ci)(0.6 rem/R)

= 2.9×10^{-6} rem/hr at 1 m/Ci.

For the sake of comparison, normal background radiation in most locations in the U. S. is about 100 mrem per year. Assuming 8760 hours in a year, the hourly dose rate is about 1.1×10^{-5} rem per hour from normal background radiation. Thus continuous exposure to one curie of Po-210 at one meter is only about one-quarter of normal background.

To assess external doses from Po-210 static eliminator manufacture, the peak year of product manufacture is considered. Based upon extrapolation from data supplied by the manufacture, the maximum year resulted in a nominal 18.8 curies of Po-210. Assuming two-and-one-half times the nominal activity at the time of manufacture (see Section 3.1.2), the resulting Po-210 throughput is 47 curies. Roughly, assuming 50 weeks of manufacture per year with 40 working hours per week, 2000 working hours per year are devoted to manufacture resulting in a Po-210 hourly throughput of

$$\frac{47 \text{ Ci}}{2000 \text{ hr}} = 2.4 \times 10^{-2} \frac{\text{Ci}}{\text{hr}} .$$

Therefore, a typical worker would experience an external dose rate far below natural background. For this reason, the external exposure is assumed to be essentially zero. No significant exposure of workers is assumed for this assessment.

3.2.2.2 Exposures During Distribution

Each static eliminator is packaged in a box before it is sent out in the distribution network. The devices are basically marketed in photographic stores and audio component stores. Since the external exposure from the Po-210 was determined to be insignificant in the preceding section and since no internal contamination is assumed to occur during normal distribution, the radiological impact during distribution is assumed to be essentially zero.

3.2.2.3 Exposures to Users

The use of Po-210 static eliminators by the public is essentially by photographers and audiophiles. The nominal 500 uCi units are basically used by audiophiles, while the nominal 200 uCi units are used by photographers. The user may be exposed either externally or internally. However, based upon the discussion in the Section 3.2.2.1, external exposure is assumed to be negligible. Therefore, the only source of exposure is assumed to be via ingestion, inhalation, or absorption into the skin. In order to have a source for internal exposure, the user must come into contact with available Po-210. Tests have been conducted in England and at ORNL to determine how much Po-210 is released due to various situations. Those pertinent to normal usage are discussed in the following text.

At ORNL, twenty-two 500 uCi static eliminators were tested (Niemeyer 78). The tests were designed and conducted with an effort to simulate an expected use environment. Wipe tests of the devices, the shipping containers, and source surfaces were made using cotton-tipped applicators. Wipe tests of the shipping containers and the devices, excluding the sources, resulted in less than 10 pCi of alpha activity, the minimum detectable activity by the instruments. The metal grids over the sources were found to have contamination on their inner faces. Rinsing with soap and water removed the contamination. Wipe tests of the exposed sources (Po-210 microspheres in epoxy resin on an aluminum strip) resulted in varying amounts of detectable alpha activity. Table 3.2 illustrates the results. It should be pointed out that Niemeyer, et al., estimated that the devices contained between 125 and 250 uCi of Po-210 at the time of the tests. Therefore, the maximum activity removed during the source wipe tests (3560 pCi) represented about 0.003 percent of the total activity in the source.

Table 3.2. Contamination Removed by Wipe Tests of Alpha Source Surfaces(1).

Source Number	Alpha Activity (pCi)
1	200
2	2,970
3	1,310
4	200
5	420
6	920
7	850
8	3,560
9	190
10	2,310
11	80
12	1,740
13	570
14	1,450
15	180
16	270

(1) From Niemeyer, 1978

Selected sources were also given ten consecutive wipe tests. The results are illustrated in Table 3.3. The maximum activity removed during any one wipe represented about 0.01 percent of the total activity in the source.

Five sources were also subjected to more vigorous wipe tests to determine the size of particles removed. These sources were pressed firmly against smear paper. Using a microscope, the particles were observed to be small with most between 1-5 um. About 40 percent were larger than 5 um. No intact microspheres were observed. Alpha activity ranged from 400 to 23,600 pCi. The high number represents about 0.02 percent of the total activity in the source.

An air sweep test of one static eliminator, in which air was blown across the device and through a filter, resulted in no detectable (<10 pCi) activity on the filter.

Vibration tests were conducted to simulate vibration stress that might be found in the home. A static eliminator was placed in a small plastic box and subjected to vibrations for 16 days. After the test, wipes of the box, the device, and the source found the removable activity to be 60, 60, and 726 pCi of alpha activity, respectively.

Leach tests of individual sources were made using tap water, soap and water, Kodak developer, Kodak fixer, acetic acid, and saliva. The amounts of alpha activity leached or removed ranged from 10 to 219 nCi, with the most being associated with tap water.

Similar wipe tests were conducted in England (Webb 75). Twenty-three 200 uCi devices were wiped with alcohol-moistened filter paper over the outside of the devices, excluding the source. The results ranged from no detectable alpha activity (<4 pCi on their instruments) to 48 pCi. The maximum represented about 0.000024 percent of the total activity in the source.

During normal usage, Po-210 contamination from the static eliminator may be picked up by the user and result in internal exposure via ingestion or inhalation. The tests by ORNL (Niemeyer 78) found no detectable alpha activity contamination on the device or shipping containers. The results of the English tests (Webb 75) found a maximum of 48 pCi of alpha activity. This corresponded to approximately 0.000024 percent of the total activity in the source. Thus, based upon these numbers, a crude estimate of the internal exposure due to normal usage of Po-210 static eliminators can be made.

Table 3.3. Contamination of Ten Consecutive Wipe Tests of Source Surfaces(1).

Smear No.	Alpha Activity (pCi)						
	Source Number						
	2	7	9	10	11	13	15
1	4,635	7,679	9,504	5,195	226	1,938	113
2	559	12,321	2,371	2,817	107	2,444	200
3	1,712	5,401	1,858	6,520	60	8,571	866
4	2,131	1,459	3,330	3,177	160	2,817	127
5	999	2,957	3,510	937	53	1,725	286
6	3,936	4,103	2,624	15,425	224	3,616	147
7	1,998	2,344	1,732	3,110	147	2,824	127
8	1,479	2,458	1,798	1,092	253	2,824	120
9	1,259	3,390	1,205	1,558	73	2,380	93
10	733	8,971	1,492	1,081	360	1,638	80

(1) From Niemeyer, 1978

For the individual user, the amount of activity that may become available is

$$(500\text{x}10^{-6}\ \text{Ci})(0.000024\%\ \text{activity removed})(\frac{1}{100\%})$$

$$= 1.2\text{x}10^{-10}\ \text{Ci}$$

for a nominal 500 uCi unit. If it is assumed ten percent may be ingested and 0.1 percent may be inhaled (Webb 75), then the following individual fifty-year dose commitments can be calculated. For ingestion, using Table A.3 of Appendix A, the fifty-year dose commitment to the kidney is

$$(1.2\text{x}10^{-10}\ \text{Ci})(0.1\ \text{fraction ingested})(5.0\ \text{rem}/10^{-6}\ \text{Ci ingested})$$

$$= 6.0\text{x}10^{-5}\ \text{rem}.$$

For inhalation, using Table A.2 of Appendix A, the fifty-year dose commitment to the lung is

$$(1.2\text{x}10^{-10}\ \text{Ci})(0.001\ \text{fraction inhaled})(63\ \text{rem}/10^{-6}\ \text{Ci inhaled})$$

$$= 7.6\text{x}10^{-6}\ \text{rem}.$$

For the population dose, consider the maximum amount of Po-210 processed in any one year, 18.8 curies. Using the same logic as in the individual case, the amount of activity that may become available is

$$(18.8\ \text{Ci})(0.000024\%\ \text{activity removed})(\frac{1}{100\%})$$

$$= 4.5\text{x}10^{-6}\ \text{Ci}.$$

The population ingestion dose to the kidney is

$$(4.5\text{x}10^{-6}\text{ Ci})(0.1\text{ fraction ingested})(5.0\text{ rem}/10^{-6}\text{ Ci ingested})$$

$$= 2.3\text{ organ-rem.}$$

The population inhalation dose to the lung is

$$(4.5\text{x}10^{-6}\text{ Ci})(0.001\text{ fraction inhaled})(63\text{ rem}/10^{-6}\text{ Ci inhaled})$$

$$= 2.8\text{x}10^{-1}\text{ organ-rem}$$

Table 3.4 illustrates the fifty-year dose commitment to various organs for both the individual and population dose. Note the results are the combination of ingestion and inhalation.

The absorption into the skin of Po-210 contamination was found to be insignificant (Webb 75). The calculations are not repeated here.

3.2.2.4 Product Disposal

As stated previously, each static eliminator is stamped with the date at which time the Po-210 source should be replaced. At this time the source should have about 80 uCi left for the nominal 500 uCi unit. The manufacturer states in the instruction booklet that accompanies the static eliminator that the consumer should send the source back to the manufacturer, if possible. As an alternative, it is suggested that the device be buried. Based upon a study of Am-241 ionization chamber smoke detectors (Belanger 79), it can be assumed that very few sources have been returned to the manufacturer. It is felt that normal garbage will be the disposal method. The following discussion deals with such disposal.

Waste Collection

No exposures of the municipal waste collection population is expected. The external dose as discussed previously is insignificant. It is possible some internal exposure could occur due to the Po-210 sources. However, it is felt that such internal exposures would also be insignificant.

Table 3.4. Internal Dose Commitments Due to Normal Usage of Po-210 Static Eliminators.

Dose	Organ				
	Total Body	Liver	Bone	Lung	G. I. Tract
Individual(1) (Rem)	2.0E-6	1.8E-5	8.6E-6	7.6E-6	1.4E-6
Population(2) (Organ-rem)	7.7E-2	6.8E-1	3.3E-1	2.8E-1	5.4E-2

(1) Based upon use of a nominal 500 uCi device for one year.

(2) Based upon distribution in one year of devices containing a nominal 18.8 curies.

Disposal in Landfills

Despite the short half-life of Po-210, an analysis of the consequences of disposal in landfills is analyzed in this section. The methodology and assumptions found in Appendix D are utilized for the analysis. The leach rate of Po-210 from the sources has been calculated using the work of Lahr et al., (1963). Soak tests on Po-210 microspheres indicated 0.005 percent was leached after one day, 0.01 after seven days, and 0.03 percent after 50 days. The extrapolated value for one year is 0.2 percent. Therefore, the value of R_t, the leach rate of activity from intact products, is 2×10^{-3} fraction per year. The leach rate of activity from previously incinerated products, R_c, is assumed to be a factor of 100 higher, or 2×10^{-1} fraction per year.

Using the reference landfill site data and the equations in Appendix D, the amount of activity reaching drinking water and being consumed is virtually zero. The reason is as follows. Assuming all the static eliminators produced during the peak year (nominal 18.8 curies of Po-210) were disposed of the following year, then the total activity disposed of would be 0.16 of 18.8 curies, or 3.0 curies (see Section 3.1.2).

For the reference case, t_m, the time required for activity to reach a ground water withdrawal point was determined using the reference site values and a retardation factor in reference soil for Po-210 of 710 (see Table D.5). The value for t_m was found to be 7300 years. Because of the 138 day half-life of Po-210 virtually no activity will be consumed with drinking water.

The analysis for the "worst case" as outlined in Appendix D results in some activity being consumed. For this case, the value for the retardation factor is 71, or one-tenth the reference value, and the depth of the landfill above the aquifer is assumed to be zero. The value for t_m is calculated to be 4.4 years assuming a groundwater velocity of 1600 m per year and a distance of 100 m to the nearest well. The resulting consumption of total activity over all time, A_{tot}, is

$$A_{tot} = 0.025 N_o A_o e^{-\lambda t_m} \left[\frac{0.9 R_t}{\lambda + R_t} + \frac{0.1 R_c}{\lambda + R_c} \right]$$

which reduces to

$$A_{tot} = 8.1\times10^{-4}\ e^{-1.83t_m}$$

for N_oA_o = 3.0 Ci of Po-210. Since t_m = 4.4 years, $A_{tot} = 2.6\times10^{-7}$ Ci.

The results of disposal are listed in Table 3.5.

Incineration

Since approximately ten percent of all municipal refuse is incinerated, ten percent of all disposed static eliminators are assumed to be incinerated. The methodology for calculating resultant doses from incineration is included in Appendix D.

Calculation of how much activity is released via incineration requires the determination of how much activity is released from the source during the burning process. Tests by Niemeyer et al. (1978) on a source with burning up to 925°C revealed 2.3 uCi were driven off and collected on a downstream filter. The amount remaining with the debris was 86.8 uCi. Therefore, only about 2.6 percent was released. However, Niemeyer had estimated the source originally contained between 125 to 250 uCi. Since the results are unclear, it is assumed here that about ten percent of the activity will be released via burning.

Using the methodology of Appendix D, a release fraction from the source of 0.1, and an activity of 0.3 Ci of Po-210 sent to incineration during the peak year (one-tenth of 3.0 Ci), the resulting consequences were calculated and the results are presented in Table 3.6.

3.2.2.5 Accidents

The evaluation of the radiological health impact of static eliminators requires an assessment of potential accidents or misuses. Three cases are analyzed here: (1) a residential fire, (2) warehouse fires, and (3) misuse by a small child.

Table 3.5. Ingested Activity and Resultant Population Dose Commitments for Reference Case and Worst Case Landfill Disposal of Po-210 Static Eliminators(1).

Case	Migration Times (yr)			Total Activity Ingested (uCi)
	Vertical	Horizontal	Total	
Reference	6.3E+3	1.0E+03	7.3E+03	0.0
Worst Case	0.0	4.4E+00	4.4E+00	2.6E-1

Case	Fifty-year Dose Commitment (Organ-rem)				
	Total Body	Liver	Bone	G. I. Tract	Kidney
Reference	0.0	0.0	0.0	0.0	0.0
Worst Case	4.4E-2	3.9E-1	1.9E-1	3.1E-2	1.3E+0

(1) Disposal of 3.0 curies of Po-210

Table 3.6. Fifty-year Inhalation Dose Commitments from Incineration of Po-210 Static Eliminators(1).

Category	Activity Released (Ci/yr)	Activity Inhaled (uCi/yr)	Fifty-year Dose Commitment (Organ-rem)				
			Total Body	Liver	Bone	Lungs	G. I. Tract
Maximum Individual	1.0E-5	1.4E-7	2.7E-8	2.7E-8	2.4E-7	1.1E-7	1.2E-8
Population Around One Incinerator	1.0E-5	5.1E-3	9.7E-4	9.7E-4	8.7E-3	4.0E-3	4.3E-4
Total Population	3.0E-3	1.5E+0	2.9E-1	2.6E+0	1.2E+0	9.5E+1	1.3E-1

(1) Incineration of 0.3 curies of Po-210

Residential Fire

Fires in which Po-210 static eliminators are involved have the potential for delivering a dose commitment to residents or firefighters because of airborne releases of activity. The following assumptions are used to assess residential fires:

(1) One-tenth of the radioactivity of one static eliminator is released during combustion.

(2) The Po-210 content is 500 uCi.

(3) All released activity is respirable.

(4) Volume of house air is 5.0×10^8 cm^3.

(5) The breathing rate of involved personnel is 1.2×10^6 cm^3/hr.

(6) Four residents are exposed to the released activity for 0.25 hours before leaving the building, and four firefighters are exposed for eight hours.

Since it is unreasonable to assume that firefighters would enter a burning building without first ventilating it, a considerable number of air changes will occur during their exposure period, resulting in a significant reduction in the concentration of airborne radioactivity. It is also unlikely that firefighters would enter the involved building without respiratory protection or that the building would not be well-ventilated during salvage and cleanup operations. This assessment assumes that the net effect of the above and other factors is a 100-fold reduction in Po-210 inhalation. Results of this assessment are presented in Table 3.7. Dose commitments are in terms of person-rem or organ-rem for one Po-210 static eliminator and are given for each group exposed in total.

Warehouse Fires

The storage of static eliminators in manufacturer or distributor warehouses represents a potential exposure hazard in case of fires. It is possible that firefighters or other involved personnel may be unaware of the radiological hazard that may exist. As in the case of a residential fire, the primary radiological hazard is from the inhalation of the airborne activity. The following assumptions are used to assess warehouse fires:

Table 3.7. Fifty-year Dose Commitments Resulting From a Residential Fire and a Warehouse Fire(1).

Type of Fire	Group Exposed	Amount Inhaled (uCi)	Dose Commitment (Person-rem or Organ-rem)				
			Total Body	Liver	Bone	Lungs	G. I. Tract
Residential	Residents	1.2E-1	2.3E-2	2.0E-1	9.5E-2	7.6E+0	1.0E-2
	Firefighters	3.8E-2	7.3E-3	6.3E-2	3.0E-2	2.4E+0	3.2E-3
Warehouse	Individual Firefighters	4.0E+0	7.6E-1	6.8E+0	3.2E+0	2.5E+2	3.4E-1

(1) See text for assumptions

(1) One-tenth of the radioactivity of 1000 static eliminators is released during combustion.

(2) The Po-210 content of each static eliminator is 1250 uCi (see Section 3.1.2).

(3) The warehouse volume is 3.0×10^{9} cm^{3}.

(4) Firefighters are present in the building for a period of eight hours following the outbreak of the fire.

(5) All other assumptions are the same as for residential fires.

Results of this assessment are presented in Table 3.7. Dose commitments are in terms of rem or organ-rem per individual firefighter.

Misuse By A Small Child

Because static eliminators may be left in a location where small children can find and misuse them, the situation is considered here. The following assumptions are made:

(1) The unit contains 500 uCi of Po-210.

(2) The child bangs the unit against an object.

(3) Based upon the vibration tests of Niemeyer, et al. (1978), 750 pCi are removed from the source.

(4) The child places the device in its mouth and ingests one-half of the contamination (375 pCi).

Appendix A includes only dose conversion factors for adults. Using the work of Hoenes and Soldat (1977) and correcting their reported dose conversion factors for a "child," ages one to eleven, by a factor of two to account for the suggested quality factor of alpha radiation (see Appendix A), the following ingestion dose commitment factors for a child are used in this assessment:

Total Body	7.3×10^{-1} rem/uCi
Liver	4.8×10^{0} rem/uCi
Bone	3.0×10^{0} rem/uCi
G. I. Tract	1.3×10^{-1} rem/uCi
Kidney	1.5×10^{1} rem/uCi

Assuming the kidney to be the critical organ, the intake of 375 pCi of activity only results in a fifty-year dose commitment of 5.6×10^{-3} rem. Therefore, the resultant dose is very small and does not present a significant health hazard.

Miscellaneous Misuses

As with all products, many postulated accidents or misuses can be envisioned to occur. The three cases above are believed to cover an adequate range of accidents. However, for historical purposes, the work of Robertson and Randle (1974) is reported here. They studied the hazards associated with industrial static eliminators. Such units contain about 2 mCi of Po-210 per inch with units up to 72 inches long. The contained activity in industrial devices is definitely much more than the activity contained in consumer devices. Part of their assessment dealt with the reported work of Caruthers and Maxwell (1971) on the contamination of a print shop at North Carolina State University from a 57 mCi Po-210 industrial device. Robertson and Randle felt the release of microspheres from the sources posed a "severe hazard" if they were crushed and then inhaled. Webb, et al. (1975) concluded, however, " ... if complete microspheres were released, it is unlikely that this would lead to significant doses to members of the general public ... ." The assessment performed for this report concurs and this area will not be discussed further.

3.2.3 Non-Radiological Impact

No significant non-radiological impacts were identified for the use of Po-210 static eliminators. It is possible that the plastic portions of the device could produce a variety of chemical compounds during pyrolytic decomposition or flaming combustion. The potential toxicity of the products is unknown and it is concluded here such a hazard is very small and not important to this assessment. The reader is referred to an earlier report by the authors of this report for a discussion of such hazards (Belanger 79).

3.2.4 Summary of Impact

The resultant estimated health risk from the manufacture, distribution, normal use, and disposal of existing consumer-oriented Po-210 static eliminators is found to be very small. This is attributable to the relatively short half-life of Po-210, the lack of a significant gamma exposure mechanism, and the fairly good integrity of the ceramic microsphere sources during normal use.

An analysis of a range of potential accidents involving Po-210 static eliminators indicates that inhalation of Po-210 activity is the main source of concern. For this reason, fires involving a number of the devices could be of potential concern. The results of a fire involving one unit indicates such an accident is not a serious problem with regard to potential dose commitments. However, the warehouse fire analysis indicates firefighters could receive a potential lung dose commitment of 250 rem. Such a dose commitment does indicate a potential health hazard.

3.3 ALTERNATIVES

3.3.1 Radiological Alternatives

Numerous radionuclides could be used in place of Po-210 for static elimination. In fact, for industrial use, other radionuclides have been used including Am-241, Ra-226, and H-3 (tritium). In terms of the attractive features of Po-210, it appears these radionuclides and others would not be as radiologically acceptable as Po-210. Such features include a relatively short half-life, insignificant gamma radiation, and a fairly immobile element. For these reasons, it is felt that Po-210, when compared to alternative radionuclides, is found to be quite acceptable as the source in static eliminators.

3.3.2 Non-Radiological Alternatives

Numerous non-radiological alternatives exist for static elimination. Such alternatives can be grouped into two categories: (1) those devices useful for only audio record static removal and (2) those devices useful for audio record static removal and for photographic static removal needs. The breakdown is made because numerous static elimination devices or systems are designed only for audio systems.

For the first category of devices, the following devices or systems are available: conductive mats, record-treatment solutions with antistatic properties (often with pads to clean the record), brushes with carbon-fiber bristles that bleed off static electricity, arm-mounted brushes with ground lines, and water systems that coat the record as it is played.

The conductive mat is placed on the record player beneath the record and acts to conduct the static charges away from the record. Tests on mats indicate the mats reduce static charge to some degree, but not totally (HF 80). Tone arm adjustments may also be required depending on the mat thickness in order to maintain correct vertical tracking when a mat is used.

Record-treatment solutions with antistatic properties are applied to the record surface and essentially act as a conductive medium. Certain of these solutions have proven to be quite effective (HF 80). One problem is the possible influence on record frequency response.

Brushes with carbon-fiber bristles are used to clean records and to bleed off static electricity. Their effectiveness is unknown to the authors. One problem is the apparent shedding of the units onto records (Berger 80). Arm-mounted brushes with ground lines are also available. Some apparently work quite well, but, again, shedding may be a problem.

Water systems put a film of water on the record as it is played, thereby conducting away static charges. The system may work but proves to be awkward to use and is not considered a viable alternative in this assessment.

The second category of devices includes devices which can remove static electricity from various objects. Included are the piezoelectric type and the battery-generated high voltage type. Diezoelectric types may contain small amounts of uranium, but for this particular assessment they are considered non-radioactive alternatives.

The piezoelectric static eliminator device uses a trigger to place stresses on a piezoelectric element. Squeezing the trigger develops a high voltage and positively ionizes the air around a pointed electrode at one end of the device. Releasing the trigger releases negative ions. Using this type of device on records "...takes luck to achieve a perfect balance of positive and negative ions..." (HF 80).

The high voltage type units using batteries generate a sufficiently high voltage to ionize air around an electrode. The units emit only positive or negative charges, but not both from the same unit. Those emitting positive charges appear to be better since records are normally negatively charged (HF 80).

3.4 COST AND RISK VERSUS BENEFIT

The use of static eliminators by consumers does not serve a safety function. The use is simply for convenience. As mentioned previously, the primary use is by audiophiles and photographers. In this section, Po-210 devices are compared to non-radioactive devices. Comparison to the use of other radionuclides versus Po-210 is not made for reasons indicated Section 3.3.1. Because of the use of Po-210 static eliminators in photography, only comparison of Po-210 devices to the piezoelectric devices and high voltage devices will be made.

3.4.1 Cost

The typical retail cost of a nominal 200 uCi Po-210 device is about \$10. The nominal 500 uCi device retails for about \$15. Replacement cartridges are about \$7 for the 200 uCi device and \$10 for the 500 uCi device. Expected replacement of the cartridges is every year, though based upon when the device reaches the consumer, the period of use may be up to 18 months. One year, however, is considered to be typical. The piezoelectric devices retail for about \$30 and the high voltage battery-operated devices cost between \$35 and \$40.

3.4.2 Product Analysis

Discussion of cost and risk versus benefit for Po-210 static eliminators will not be made because of the difficulty of relating a convenience benefit to cost or risk. The risks are discussed in Section 3.2. The risks associated with manufacture, distribution, normal use, and disposal were found to be minimal and of little significance. However, the potential exposure to firefighters from an accident in a warehouse containing 1000 devices was found to be high and could be considered to be a health hazard.

3.4.3 Alternatives Analysis

Since the non-radioactive static eliminators essentially pose little risk to the user, the comparison of these devices to the Po-210 devices must be made based upon cost and product effectiveness.

The useful lifetime of the piezoelectric devices and high voltage devices is not known. It is assumed to be in the range of five years. The piezoelectric devices are generally guaranteed for three years. The piezoelectric devices represent an initial cost only, while the high voltage devices require replacement batteries over an unknown interval. Because of the various uncertainties involved, a detailed cost evaluation cannot be made. However, it appears because of the relatively short replacement period (one year) and the relatively high cost of replacement (about 70 percent of original cost) that the nominal 500 uCi Po-210 device is not as cost-effective as the other devices for audiophile use. For photography, this may not be the case since the nominal 200 uCi device is normally used. Its initial cost is relatively low compared to the other devices and the cost of yearly replacement cartridges, though 70 percent of the initial cost, is not prohibitive.

The product effectiveness or ability to remove static electricity appears to be similar for all the devices. Tests by Webb, et al. (1975) comparing a piezoelectric device to Po-210 devices, concluded both methods "are effective in static elimination." They also concluded the activity levels in the Po-210 devices "are probably necessary."

3.5 FUTURE PRODUCTS

No new static elimination devices or methods are expected to be developed during the next decade. Instead, variations of present devices or methods will probably be introduced.

The projected distribution of Po-210 static eliminators from 1980 to 1986 is about 100,000 of the nominal 500 uCi devices and about 500,000 of the nominal 200 uCi devices.

3.6 SUMMARY

The judgement as to whether a consumer product containing radioactive materials should be manufactured and distributed is a complex problem. For the case of products, such as Po-210 static eliminators, where no life saving or safety benefit functions are involved, the problem becomes tougher. Judgement then must be made based upon cost effectiveness, product effectiveness, and an assessment of potential radiological health hazards (and comparison to any environmental impacts of alternatives).

The health hazards or risks from the manufacture, distribution, normal use, and disposal of Po-210 static eliminators were found to be low and insignificant. A potential health hazard was found to exist for firefighters involved in fires of warehouses where large quantities (around 1000 units) of Po-210 static eliminators are stored. However, other risks, even greater, may be taken by the firefighters in such situations simply because of their profession and its associated risks.

The cost effectiveness of the Po-210 device was found to be good when used for photography and poor when used for audiophile purposes. Its effectiveness as a static remover was found to be similar to non-radioactive devices.

3.7 REFERENCES FOR SECTION 3

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4. DENTAL PRODUCTS

4.1 PRODUCT DESCRIPTION AND BACKGROUND

4.1.1 General Discussion

When human teeth are missing or badly decayed, it becomes desirable and/or necessary to correct or replace them with artificial material. Should the surface of the tooth being repaired or replaced be visible when the patient speaks, it most frequently is desirable that the filling material give the appearance of natural teeth (as opposed to displaying gold or silver). Since the environment of the tooth is that of extremes of heat, cold, and abrasiveness, it is necessary that the natural-appearing material used in a replacement tooth be capable of withstanding these insults. Porcelain has long proven itself (over 30 years) to be a material that can meet these requirements.

One drawback to using porcelain as a tooth replacement material is that the material in itself does not give the natural fluorescent appearance of normal teeth. To impart this fluorescent quality to artificial porcelain teeth the porcelain material is doped with a small amount of uranium oxide.

4.1.2 Product Description

Not all human teeth have the same coloration or fluorescent quality. Because of this a variety of porcelain teeth and porcelain powders are available for the construction of dental prostheses. Analyses of porcelain teeth (Thompson 76, Weaver 75) have indicated that the porcelain used in artificial teeth in the U. S. averages 300 ppm uranium. This value was determined from samples of manufactured materials by means of gamma counting techniques and neutron activation analysis. A total of 39 teeth were surveyed to determine the uranium content and the range was found to be from 460 ppm to less than 100 ppm. Since the license exempt-level (see Appendix C) for a compounded material is 500 ppm, the 300 ppm uranium as found in artificial porcelain teeth is well within the legally-defined range for a manufactured product.

For porcelain powders used in the U. S., the report by Thompson gives some data points as to the concentration of uranium. Out of a sampling of 23 powders the uranium content, as determined by gamma counts, ranged from a high of 990 ppm to a low of less than background. Although a uranium content of 990 ppm far exceeds the 500 ppm license exempt-level, when this material is blended into a finished product the overall concentration is below the legal limit.

No study has been located that specifically deals with the uranium content of bridges and crowns constructed in dental labs in the U. S. In the report by Weaver (Weaver 75) uranium concentrations were calculated for three layers of three different crowns and an average of 260 ppm uranium was indicated. The range of uranium concentration in these teeth was from 453 ppm for core material of one tooth to a low of 150 ppm for opaque material of another tooth.

4.1.3 Distribution

Data on file with the Nuclear Regulatory Commission (NRC) indicate a consumption by two manufacturing members of the dental porcelain industry of over 100 pounds of depleted uranium in a single year. If the maximum reported number of manufactured porcelain teeth in a single year, 89 million (USBOC 80), is combined with an estimate of the amount of porcelain powder (one-third of tooth volume), there would be a rather large discrepancy between the uranium consumed compared to the uranium found in the manufactured product. This discrepancy would indicate that at least 80 pounds of uranium were missing somewhere in the manufacturing pipeline. A closer examination of the situation, which is not in government statistics, will bring the picture into focus.

When the dental porcelain manufacturer produces porcelain teeth and porcelain powder the processing results in a scrap or waste production of as much as 50 percent of the originial material. This scrap material is collected and disposed of by a licensed disposal company. The porcelain material that remains in the production pipeline now goes into three areas:

(1) Teeth - Manufactured in the U. S. - Sold for use in the U. S. and sold for export.

(2) Teeth - Manufactured outside the U. S. from powder manufactured in the U. S.

(3) Powders - Manufactured in the U. S. - Sold for use in the U. S. and sold for export.

Of these three categories only the first, teeth manufactured in the U. S. and sold for use in the U. S. or for export, would be the category that would show up in the census numbers. The uranium that is used in the manufacture of dental porcelain can now be seen to be distributed approximately as follows:

Proper Disposal	50 percent
Overseas Use	10-20 percent
Domestic Use	30-40 percent

Also, as explained by manufacturers' representatives, the amount of uranium purchased by one company, as indicated in NRC files, is far too high. These companies will, on occasion, purchase large lots of material, some of which will be intended to last more than a single year. Therefore, the amount purchased by a company in a specific year may not reflect the amount actually used that year by the company. Of the domestic use material, discussion with industry representatives indicates approximately 35 percent of domestic use of dental porcelain is in the form of dental powder. This powder is used in small dental laboratories to construct crowns and bridgework.

The distribution of porcelain teeth by U. S. manufacturers is a data point that is collected by the U. S. Bureau of Census under industry SIC 3843 - Dental Equipment and Supplies (Stat. Abs. 1979). Data for this census is developed at five-year intervals and as can be seen in Figure 4.1, the quantity shipped has steadily declined since 1958. In 1958, there were an estimated 64.9 million porcelain teeth shipped by U. S. manufacturers while in 1977 the quantity shipped had dropped to 38.7 million. Assuming 38.7 million teeth and an estimated 0.6 grams as the average porcelain content of a tooth, the annual amount of porcelain used in teeth would be 23,220 kilograms or 51,190 pounds. Using an average of 0.03 percent by weight for uranium content, the amount of uranium used in porcelain teeth would be 15.4 pounds per year.

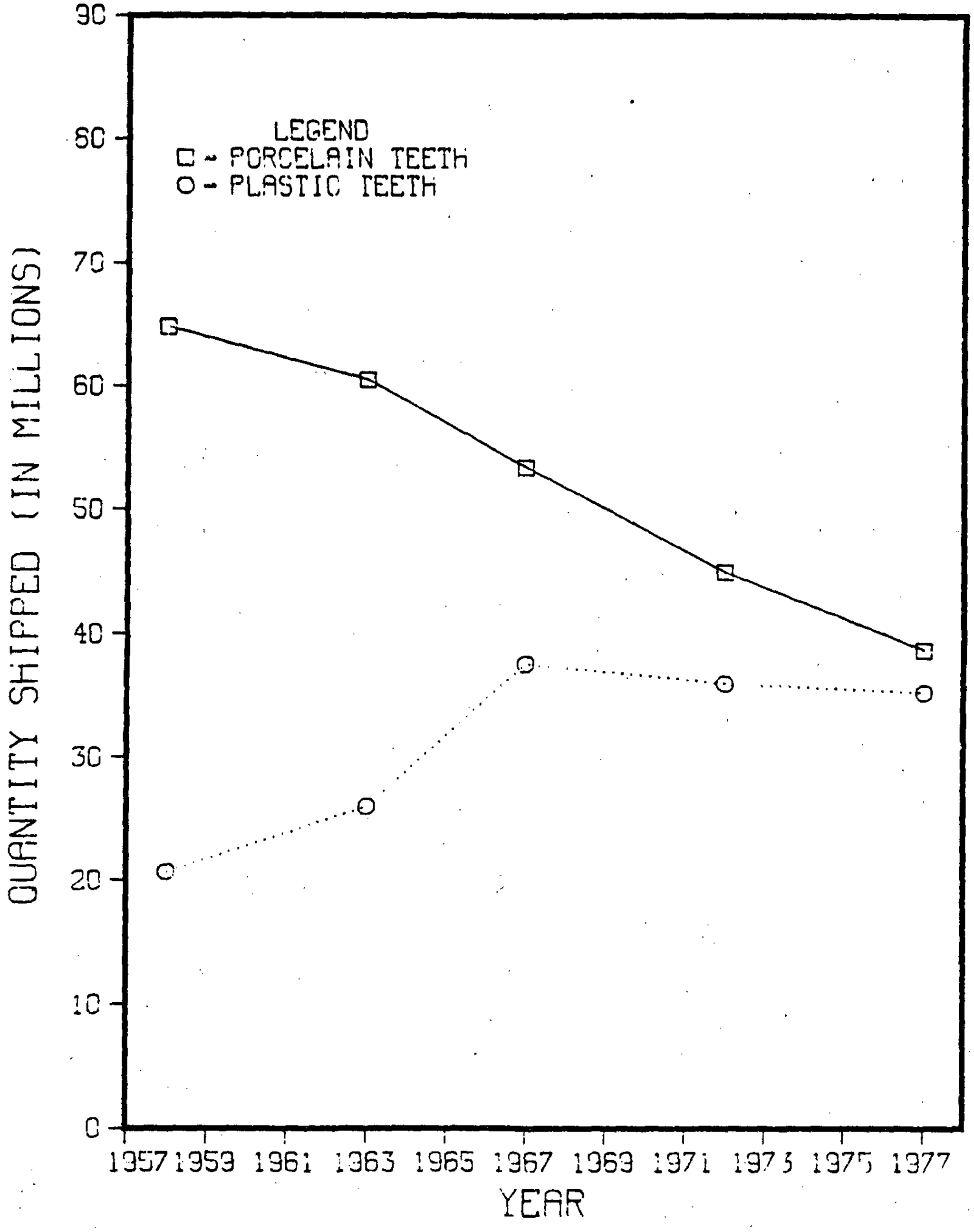

Figure 4.1. Porcelain Dental Teeth Shipped Per Year (USBOC 79).

The Census Bureau places a 90 percent accuracy on its porcelain teeth estimates. What the census data does not indicate is the import/export picture for this material. Separate data is collected on import/export amounts of artificial teeth but the values are only in estimated dollar amounts and are, therefore, of little use when estimating quantities. Conversations with dental trade associations and Census Bureau personnel indicate a reasonable balance of export versus import may occur in the U. S. and, therefore, the maximum number of porcelain teeth distributed in the U. S. will be considered to be the amount indicated in the Census Bureau data (38.7 million per year).

The decline in manufactured amounts of porcelain teeth can be attributed to at least two things. As can be seen in Figure 4.1, from 1958 to 1967 there was an increase in the amount of plastic teeth being distributed. From 1967 on there also appears to be a decrease in the manufactured amount of plastic teeth as well as porcelain teeth. This decline in tooth replacement has not necessarily resulted in as great a decrease in the use of dental porcelain as might be expected. This is explained by the fact that there has been a concomitant increase in the number of crowns and bridges. Much of this increase has resulted in the use of dental porcelain powder to construct the required prosthesis. It is, therefore, apparent that knowledge of the distribution of porcelain powder is also important in understanding the impact of dental porcelain material on the U. S. population.

Based on data of the National Center for Health Statistics (NCHS 79) 14.7 percent of the 1974 U. S. adult population aged 18-74 years (20 million persons) had lost all their permanent teeth. Another 9.2 percent (12.5 million persons) had lost all of their upper or lower teeth. Of this total population, 47.7 percent (15.5 million persons) were in need of construction of dentures. Figure 4.2 illustrates the distribution by percent of class of the numbers of persons in need of dentures or bridgework. For the person requiring bridgework this represents approximately 34 million individuals. The population numbers for this analysis were derived from Census Bureau data (USBOC 78). As these figures indicate, there is an extensive market within the U. S. population for artificial teeth. As to the number of crowns, the federal data base does not indicate an estimate for this category. Thompson (1976) indicates that the 1971 U. S. population included 60 million persons with dental crowns. Undoubtedly this number has increased with the increase in the U. S. population, an approximate additional 55 million persons by January 1, 1980.

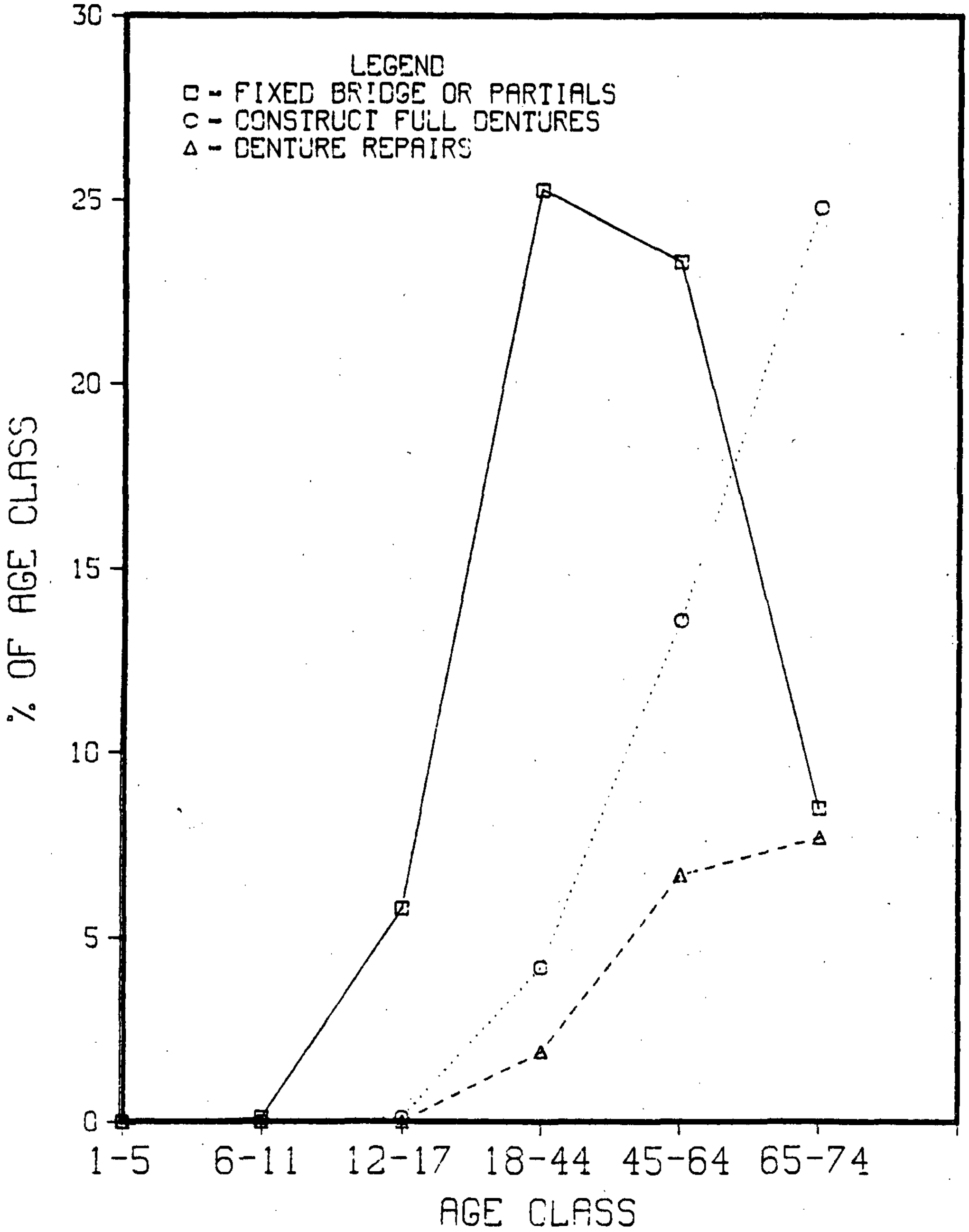

Figure 4.2. Percent of Persons Aged 1-74 Years with Specific Dental Treatment Needs (NHS 79).

Inquiries to the manufacturers of dental porcelain failed to develop information regarding the quantity of the material manufactured per year. It was the feeling of most manufacturers that this sort of information was proprietary in nature and, therefore, could not be released. According to the National Association of Dental Laboratories (NADL) there are approximately 8500 dental laboratories in the U. S. Of these labs about 20 percent primarily construct porcelain crowns while the rest are full-service labs. The average labs can produce approximately 55 crowns per week. Using this information the maximum number of porcelain crowns manufactured in the U. S. would be approximately 24 million per year. Although this amount is quite possibly high due to the construction of plastic crowns, we will use it to determine an upper bound. Based on an average tooth weight of 0.6 grams, approximately 32,700 pounds of porcelain powder are used annually in the U. S. At an average of 0.03 percent by weight for uranium content, the uranium utilized in this material would amount to 9.8 pounds per year.

Storage of porcelain teeth occurs in three basic locations (see Table 4.1): (1) the manufacturing facility, (2) the major distributors, and (3) the independent dental laboratories. Manufacturers have indicated that the maximum storage in a single location would be 15 million teeth. The 1977 Census of Manufacturers (COM 77) indicates a total of six companies in the U. S. with shipments of teeth worth $100,000 or more. Distributors keep on hand an average of 10,000 teeth and inquiries indicate there are approximately 60 of these locations across the U. S. The total teeth in these locations would be 600,000. The smallest storage depot for porcelain teeth are the independent dental labs. These locations, of which there are about 2550, keep on hand approximately 200 teeth for a total of 510,000. In summary, there are three possible locations for dental prosthesis with an estimated storage of 91.1 million porcelain teeth.

4.2 ENVIRONMENTAL IMPACTS

4.2.1 Benefits

The depleted uranium used in dental porcelain is of greater benefit today than it was back in the 1940's when it was first introduced. Two things must be considered to properly understand this statement. When uranium oxide was first introduced it succeeded in improving the fluorescent quality of artificial teeth but it did so only within certain limitations. Secondly, since the early

Table 4.1. Location and Distribution of Finished Porcelain Teeth.

Facility	Number of Locations	Number of Teeth per Location	Total Teeth Per Facility Class
Manufacturers	6	15,000,000	90,000,000
Distributors	50	10,000	600,000
Full Service Laboratories	2,550	200	510,000

1940's there has been a distinct change in life styles, both in jobs and in entertainment. These life-style changes have included a change in the type of artificial lighting that people are exposed to which in turn has, for psychological as well as appearance reasons, required an improvement in the fluorescent ability of porcelain teeth beyond what uranium oxide could do on its own.

An advance in the fluorescent quality of porcelain teeth was made in the 1950's when it was discovered that additions of cerium along with the uranium would produce a more satisfactory fluorescent color (O'Riordan 1974). A recent study was done by Wozniak and Moore on the luminescence emission spectra of dental material (Wozniak 78). In this study, the luminescence spectrum of powdered natural teeth was compared to artificial porcelain and vinyl teeth as well as porcelain powder. Powdered normal human teeth display an emission spectrum peak of approximately 415 nanometers (nm). Artificial porcelain teeth (including nine types of teeth from six manufacturers) displayed emission maximum of 380 nm to 530 nm with a mean emission maximum of 460 nm. Although the emission maxima of artificial teeth was not the same as that of natural teeth, it clearly is an improvement over using uranium oxide alone since this material has an emission maximum of around 550 nm.

As described above, the only benefit derived from the uranium in porcelain teeth is strictly of a cosmetic nature. But this cosmetic quality could be significant for persons employed in, or given to public exposure through the television or film industry. The lights used in these industries are such that they produce sufficient soft ultraviolet radiation to highlight visual defects due to poor tooth fluorescence. Such defects cannot be tolerated in these industries.

4.2.2 Radiation Impact

4.2.2.1 Occupational Exposure

Occupational exposure to porcelain teeth and porcelain powder is not a simple uniform exposure. Two means of exposure may possibly exist during manufacturer in the U. S. The most potentially serious exposure could occur in the mixing and developing processes that require the uranium oxide to be blended with porcelain material. Manufacturers contacted during the course of this study indicated that overall not more than 100 persons would be employed in this operation by all of the U. S. manufacturers. An estimate of approximately ten times this number is used to calculate the total number of persons employed in the manufacture of porcelain teeth that might be occupationally exposed to the finished product.

Exposures were calculated based on the methodologies outlined in Appendix A. For workers, the exposure period is estimated at 2000 hours per year. The determination of occupational doses is not easily calculated but can be shown to be very small using the following hypothetical assumptions. If each worker is assumed to be one meter away from all the stored teeth (an impossible situation), the total body dose is calculated to be 4.2×10^{-2} rem per year per worker in the manufacturing category. This is judged to be a very conservative estimate. The total worker dose is 50 person-rem per year as seen in Table 4.2. This dose is due essentially to the very large number of teeth (15 million) in storage. This large storage number was suggested by one manufacturer as being the highest that could ever be found at one time, but a highly unlikely situation.

4.2.2.2 Exposure During Distribution

For this category the distributors, dental laboratories, and dental offices are grouped together. Two specific situations arise here that must be noted. The laboratory workers that were contacted were totally unaware of any uranium content in the porcelain material. They worked with their hands directly in contact with the uraniated porcelain and with no inhalation protection. The second situation is the potential for inhalation or ingestion of the porcelain material by the dentist, dental assistant, or dental patient. Their exposures will be very low in any case but do present specific situations with physiological involvement.

The dose estimates are listed in Table 4.2. The highest individual exposure, based on storage, is 2.8×10^{-6} rem per year for the worker in the distributor category. Doses to dental office workers were found to be 0.13 person-rem per year. Doses to transportation workers were not assessed.

Table 4.2. Occupational External Doses Due to Uranium in Dental Porcelain.

Activity	Number of Locations	Workers Per Site	Number Teeth Stored Per Site	Dose		
				Rem/hr Per Worker	Rem/yr Per Worker	Total Person-rem Per Site
Manufacturing	6.0E+0	2.0E+2	1.5E+7	2.1E-5	4.2E-2	5.0E+1
Distributors	5.0E+1	5.0E+1	1.0E+4	1.4E-9	2.8E-6	1.4E-4
Laboratories	8.5E+3	4.7E+0	2.0E+2	2.8E-10	5.6E-7	7.1E-2
Dentists	1.2E+5	2.0E+1	2.0E+1	2.8E-11	5.6E-8	1.3E-1

4.2.2.3 Exposures to Users

From the Census Bureau data and information from Thompson (see Section 4.1.3), an estimated 65 million persons in the U. S. have at least one denture, including 20 million persons with full dentures of porcelain teeth. An average of four crowns per person was used to calculate gamma exposure due to the uranium. For the denture wearer, an average of 28 porcelain teeth per person was used for gamma exposure calculations. For both calculations the gamma dose rates were assumed to be $1.4\text{x}10^{-12}$ rem/hr/tooth. For the development of this value, see Section 4.2.2.1 and Appendix A.

Note, this value is for a one-meter distance with no shielding. For teeth, the source is in the mouth, obviously closer than one meter, but with body tissue as shielding material. It is felt that by assuming a one-meter unshielded distance, the resulting answer is conservative.

For the porcelain crown wearer the whole body gamma dose is calculated at $5.6\text{x}10^{-12}$ rem per hour or $4.9\text{x}10^{-8}$ rem per year (see Table 4.2). For the 65 million wearers of dental porcelain crowns, this dose is 3.1 person-rem per year.

For the full porcelain denture wearer, the dose commitment due to gamma-ray exposure is calculated at $3.4\text{x}10^{-7}$ rem per year and 6.8 person-rem for the estimated population of 20 million persons. Table 4.3 also lists organ dose commitments for both categories with the most significant dose being that to the bone marrow of the full denture wearers at $4.9\text{x}10^{-7}$ rem per individual and 9.9 person-rem for the population. It must be remembered, however, that the external dose conversion factors used here are for isotropic exposure, while in the case of a denture wearer, the exposure would be highly directional.

The question of radiation dose to tissue in the buccal cavity as a result of uranium in dental porcelain is complicated by several factors. For the incident radiation to be significant in terms of somatic damage it must reach germinal tissue. In regards to dental porcelain, the germinal tissues of concern are: gingivae (gums), buccal parietes (inner cheek and lips), and lingua (tongue). Each of these tissues is a system in which the germinal basal layers are immediately below the surface of the epithelium. The depth of the basal tissue is not well-defined. Estimates of the thickness of the epithelium have ranged from 20 microns (2 mg/cm^2) (Whitton 73) to 200 microns (20 mg/cm^2) (O'Riordan 74). To date all investigators concerned with potential dose from uranium in dental porcelain have chosen to consider the basal depth to be at least 30 microns. This is extremely significant in that it eliminates the

Table 4.3 Annual Gamma-ray Doses to Wearers of Dental Porcelain Products Containing Uranium

Exposure Group	Dose (Organ-rem or Person-rem)				
	Total Body	Liver	Bone	Lung	G. I. Tract
Full Denture					
Individual	3.4E-7	2.9E-7	4.9E-7	3.1E-7	2.7E-7
Population	6.8E+0	5.8E+0	9.9E-0	6.2E+0	5.4E+0
Crown					
Individual	4.9E-8	4.1E-8	7.1E-8	4.5E-8	3.9E-8
Population	3.1E+0	2.6E+0	4.6E+0	2.9E+0	2.5E+0

possibility of alpha particle dose to the basal layer, as 30 microns (3 mg/cm^2) is beyond the depth to which U-238 alpha particles can penetrate.

Alpha emissions due to uranium in dental porcelain are absorbed by the layers of plaque and saliva on the tooth surface, as well as a layer of calculus between the teeth and gingival tissue. Saliva also coats the tongue, cheeks, and lips to form another layer of absorbing medium. Added to these absorbers is the movement of the tongue, cheeks, and lips. All of these tend to greatly reduce the time of intimate contact between the teeth and soft tissue.

Measured alpha-particle emission from porcelain teeth range from 7.8 particles/hr/cm^2 (170 ppm) to 52.1 particles/hr/cm^2 (440 ppm)(Thompson 76). These values, based on Spiers' equation (Spiers 69), would be 100 rem/yr (170 ppm) to 259 rem/yr (440 ppm) for surface dose readings. Calculations based on the computational methods of Sairenji, et al. (Sairenji 80) would indicate a surface dose range of 136 to 352 rem/yr. In either case of computational methodology, the authors indicate that they would find the alpha dose to be essentially zero at a 30 micron basal tissue depth.

Beta particles are also emitted from porcelain teeth that contain uranium. The beta-particles are the result of the presence of uranium daughter nuclides and also potassium-40 (K-40) atoms. The K-40 contribution to the beta particle dose ranges from 0.14 rem/yr to 0.20 rem/yr with an arithmetic mean of 0.16 rem/yr (Thompson 76). While the uranium beta dose will vary according to uranium concentration in the porcelain, at 300 ppm it is calculated to be 0.81 rem/yr. Figure 4.3 gives the beta dose due to uranium daughters in dental porcelain. At 500 ppm, the maximum allowable concentration under 10CFR40, the tissue surface dose from the uranium would be approximately 1.35 rem/yr. If the apparent average concentration of uranium in dental porcelain is considered (i.e., 300 ppm), then the beta dose becomes 0.81 rem/yr uranium plus a K-40 contribution of 0.16 rem/yr for a total of 0.97 rem/yr.

For the U. S. population, the radiological impact to the buccal cavity of individuals wearing porcelain dental prosthesis containing uranium is approximately 1.0 rem/yr to the basal mucosa. This is considered a maximum dose estimate for a porcelain tooth containing 300 ppm uranium.

4.2.2.4 Product Disposal

The disposal of dental porcelain is a special situation in itself. Four distinct methods can be identified: licensed disposal sites, community refuse sites, cemeteries, and disposal via cremation.

When dental porcelain is manufactured into teeth there is a loss of approximately 50 percent of the original material (as per conversation with manufacturers). This material is collected by waste disposal companies that are licensed to handle radioactive materials. This material is then transported to low-level waste sites for controlled burial. Since this form of low-level waste control comprises a category onto itself, it will not be considered in terms of landfill or groundwater contamination.

Disposal of porcelain teeth in landfills will be less then the numbers of teeth disposed of via burial or cremation. Most porcelain teeth that are not kept by the original purchaser are returned to dental labs where they are reused after reconstruction. As a result, there are little, if any, porcelain teeth finding their way to landfills. This, therefore, does not constitute a situation which warrants a contamination analysis.

Because of the restricted use characteristics of burial sites and mausoleums, and because artificial teeth made of porcelain have extended ware characteristics similar to natural enamel, no transport calculations will be made for this type of disposal.

4.2.2.5 Accidents

Accidents in the form of a warehouse fire are not expected to be a radiological problem with respect to inhalation dose commitments resulting from the presence of large amounts of porcelain teeth. This is based on the assumption that most warehouse fires will not exceed 1300^{o}C and that the firing temperature of porcelain is around 1800^{o}C. The melting point of uranium is about 1132^{o}C and it is, therefore, conceivable that some uranium could be released from the porcelain matrix. Although uranium has a boiling point of 3818^{o}C, for this study it will be assumed that 20 percent of the uranium is released from the porcelain tooth matrix and that all this material becomes airborne. Using this assumption a fifty-year dose commitment to firefighters is assessed.

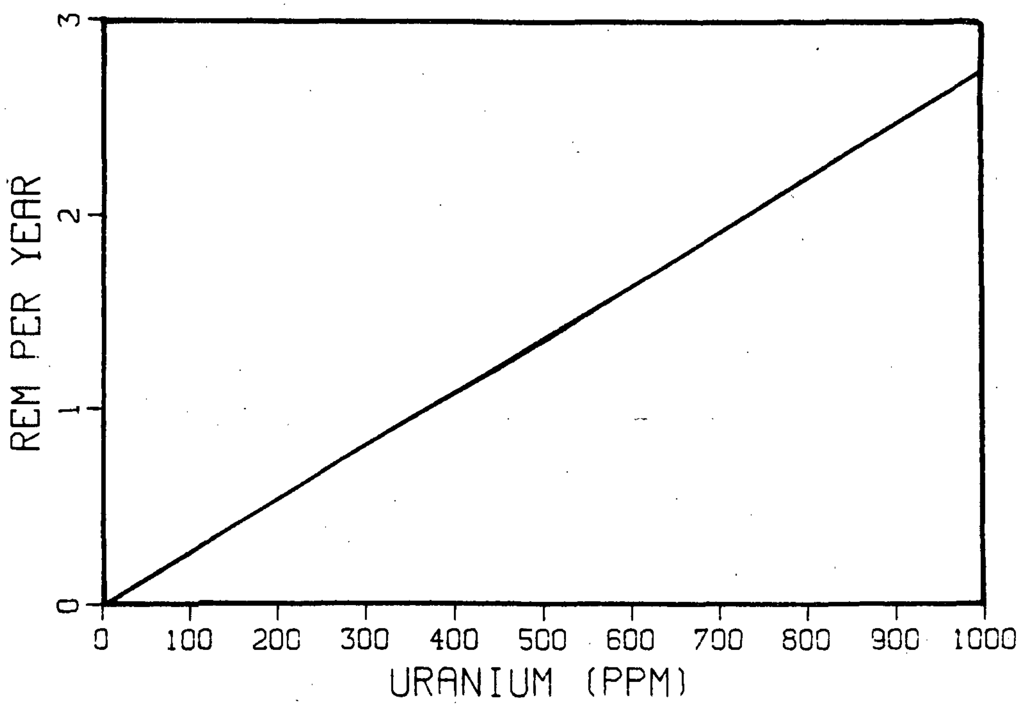

Figure 4.3. Beta Dose to Buccal Surface Mucosa from Uranium Daughters in Porcelain Teeth (Thompson 76).

Table 4.4 illustrates the fifty-year dose commitments to firefighters as calculated according to the assumptions and the methodology outlined in Section 2.2.2.4. The maximum fifty-year dose commitment to firefighters is 0.53 rem to the lungs. Fifteen million teeth are assumed to be in storage. This is a very conservative estimate, since it represents the highest possible amount for any expected storage situation related to porcelain teeth. It is probably a few orders of magnitude above what actually occurs.

As a comparison to the firefighters dose commitment, a fifty-year dose commitment was calculated for a dental lab technician (see Table 4.4). It was assumed that five percent of the powder lost during construction of a tooth was constantly in the workers room atmosphere (volume of 45 m^3) and that this amounted to five percent of the weight of an average tooth. Calculations based on the methodology used for the firefighter scenario (except the worker is exposed for 2000 hours) resulted in the dose commitments of Table 4.4. The highest fifty-year dose commitment to the technician from inhalation of the porcelain powder is 1.5 rem to the lungs.

4.2.3 Non-Radiological Impact

Non-radiological impact associated with porcelain teeth is not easily defined. No toxic consequences were found to be associated with the materials comprising the porcelain, especially when considering that the finished product is a very strong nonporous ceramic material.

4.2.4 Summary of Impact

The health risks associated with dental porcelain containing uranium are developed for consumers as well as workers in different sections of manufacturing and distribution. A discussion of potential health hazards in terms of cancer induction is included in Section 4.4.2.

One rem per year is potentially delivered to some areas of the buccal cavity of persons wearing a porcelain dental prosthesis, making this the single most important category. Individuals in the manufacturing industry could receive an estimated dose of 4.2×10^{-2} rem per year due to their proximity to the uranium content of 15 million teeth.

Table 4.4. Fifty-Year Dose Commitments from Inhalation of Dental Porcelain.

Organ	Dose Commitment (Organ-Rem)	
	Firefighter(1)	Dental Lab Worker(2)
Total Body	6.3E-3	1.8E-6
Liver	6.3E-3	1.8E-6
Bone	1.1E-1	3.0E-5
Lungs	5.3E-1	1.5E-4
GI Tract	3.9E-4	1.1E-7(3)
Gonads	6.3E-3(3)	1.8E-6(3)

(1) Firefighter exposed to fire involving 1.5E7 dental procelain teeth.

(2) Dental lab worker exposed to constant background of 3.0E-12 Ci in working atmosphere.

(3) Assumed to be same as total body.

For both categories the estimate of dose is very conservative. The problem of critical tissue depth and concentration of uranium per tooth greatly influences the assessment results for the consumer. For the worker in manufacturing, the dose is estimated at an average distance of one meter for the 15 million teeth. The assumptions regarding distance and number of teeth are both very conservative.

With respect to accidents, the resultant doses are so small as to be of little concern. The fifty-year dose commitment to firefighters is 0.5 rem to the lungs from an eight-hour exposure to a fire involving 15 million teeth. This again is a highly unexpected situation and, therefore, a very conservative estimate.

For product disposal little can be assessed. Due to the nature of the use of dental porcelain and its final disposition with the human body, it is not expected to present a landfill or water use problem. The fate of the material that is disposed of in regulated low-level waste sites is not considered in this assessment.

4.3 ALTERNATIVES

4.3.1 Radiological Alternatives

No documentation has been found that indicates whether or not radiological alternatives have been studied as replacements for uranium in dental porcelain. Manufacturers that were contacted could not recall that any radiological alternatives had been studied by their company or their competitors. Inquiries to the American Dental Association produced no known or suggested radiolgoical alternatives.

4.3.2 Non-Radiological Alternatives

Non-radiological alternatives for producing fluorescence in porcelain dental material are being actively investigated by various manufacturers. One critical operation that is required during the process of producing porcelain dental material is the application of very high temperatures. Because of this, most, if not all, of the fluorescing materials used in plastic teeth cannot be utilized in porcelain teeth. If materials such as zinc sulfide or various organic phosphors were incorporated into the porcelain, the 1800°C thermal requirements of the porcelain processing would cause these fluorescent materials to break down.

Many, if not all, of the "rare earth" elements can be used as an adjunct with uranium in dental porcelains. The addition of cerium reduces the uranium requirement by 50-80 percent. Europium is also being used in dental porcelain. Europium is potentially a material that could completely replace uranium. The situation apparently rests on the ability of uranium replacement materials to produce the correct coloration when fluorescense is a dominant characteristic.

Porcelain can be replaced by plastic and, in fact, is in many cases. Porcelain is superior to plastic in wear characteristics, but is brittle and can, therefore, be broken. Porcelain has the disadvantage of being so strong a material that, if improperly placed, it can cause an opposing tooth to wear down. A major advantage of porcelain over plastic is that the coloration of the porcelain is permanent while the coloration in plastic will change with time.

Based on the foregoing, it must be concluded that, of the available materials for dental prosthesis, porcelain presents many qualities that will assure its continued use. The amount of uranium in the porcelain is purely a function of cosmetic requirements. Uranium may eventually be replaced completely in dental porcelain by some rare earth materials. The U. S. dental industry has set this as a requirement upon itself and appears to be making progress in that direction.

4.4 COST AND RISK VS BENEFIT

4.4.1 Cost

At 300 ppm, the amount of uranium in a single porcelain tooth would amount to 0.18 milligrams. If the cost to the manufacturer is estimated at $22 per kilogram for depleted uranium, this would only be equivalent to less than one cent per tooth to the patient. At this level the cost to the patient for the uranium in the dental porcelain is completely absorbed within the finished product and is insignifiant. Table 4.5 is a cost comparison for porcelain versus

Table 4.5. Single Item Restoration Mean Cost for Service (1977 dollars)

	CROWN		BRIDGE	
Metal Backing	Porcelain	Plastic	Porcelain	Plastic
Non Precious	186	155	181	161
Semi Precious	193	163	188	164
Gold	201	172	198	174

(1) From American Dental Association, 1978

plastic dental prosthesis. Even for the least expensive porcelain tooth, the cost for the uranium in the tooth cannot be meaningfully computed.

Dental industry spokespersons contacted during this study did not feel that there would be any monetary impact to the dental patient as new elements were introduced to replace the uranium.

4.4.2 Product Analysis

The risks to the individual who elects to wear dental porcelain teeth or crowns that contain uranium are extremely difficult to determine. Part of this is related to the question of the depth of the germinal tissue layer. This is a critical issue with respect to the penetrating power of the alpha particles from the uranium and has yet to be settled. A second aspect is in relation to the actual concentration of uranium in the porcelain and its resultant surface alpha emissions.

The fact that most of the benefit the consumer receives from wearing dental porcelain is of a cosmetic nature does make any risk associated with this material questionable. However, some measure of psychological benefit is also achieved by the likeness which porcelain teeth bear to natural teeth. It should also be noted that during the course of this investigation, no consumer or dental lab technician, and few dentists were aware of the fact that uranium was a part of the porcelain composition. In light of today's media handling of information concerning radioactivity and the resultant general population confusion, it would be of interest to determine how many people would elect to wear dental porcelain if they were informed that the material contained uranium.

To try to relate the uranium in the dental porcelain to cancer induction is nearly an impossible task. Squamous cell carcinoma is extemely rare in the ginginal tissue. This tissue (gums) is in the most continuous contact with the porcelain material, but the likelihood of cancer induction based on known probabilities simply cannot be estimated with presently known data.

The estimated incidence of oral cancer for 1980 is presented in Table 4.6. The cause of oral cancer has not yet been identified, but, as with most forms of cancer, a host of factors may contribute to its development. The death rate from mouth cancer for smokers is four times higher than for nonsmokers (ACS 80). Poor oral hygiene and chronic irritation such as jagged teeth, projecting fillings, and ill-fitting dentures are considered potential

Table 4.6 Estimated Oral Cancer Incidence for 1980(1)

Cancer Type	New Cancer Cases			Cancer Deaths		
	Number	Percent	Per 10^6 Persons	Number	Percent	Per 10^6 Persons
Total Buccal Area	18,500	2.35	83	4,800	1.18	21.6
Lip	4,400	0.56	20	175	0.04	0.8
Tongue	4,800	0.61	22	2,000	0.49	9.0
Salivary Gland, Floor of Mouth and Others	9,300	1.18	42	2,625	0.65	11.8

(1) From Silverberg, 1980.

factors. It would appear that the probability of cancer induction from these factors is higher than that from the potential dose from uranium in dental porcelain.

4.5 FUTURE USE

The use of uranium in dental porcelain is expected to continue for some years. Although other elements such as cerium and europium have been incorporated into dental porcelain to reduce the required amount of uranium, there is currently still a small requirement for the uranium. As the American population continues to gain in affluence, the cost of improved appearance is within the reach of more people. A smile with a perfect set of teeth will continue to be a quality actively sought by more people.

4.6 SUMMARY

Uranium, in combination with other materials, is used in dental porcelain to impart a fluorescent quality similar to natural teeth. The sole function of the uranium, and thus the fluorescence, is cosmetic. The uranium is held to a concentration of 300 ppm or less. The radiation dose rate to the basal mucosa for 300 ppm uranium is approximately 1 rem/yr. This dose is due to alpha and beta emissions from the uranium, and its daughters.

For gamma-ray exposure, the full porcelain denture wearer would receive 3.4×10^{-7} rem per year. The highest occupational exposure is calculated to be to workers in the manufacturing category at 4.2×10^{-2} rem/yr. This is based on an exposure to 15 million teeth and is extremely conservative due to the geometries of the storage situation. Fifty-year dose commitment to a firefighter exposed to a fire involving storage of 15 million porcelain teeth would be 0.53 rem to the lungs. This, again, is a highly unlikely scenario and thus a very conservative estimation.

Alternatives to dental porcelain are metals such as gold or silver, and plastic. The plastic material is not as suitable as porcelain in that it is not as durable and colors with age.

Uranium is expected to continue to be employed in dental porcelain for some years.

4.6 REFERENCES FOR SECTION 4

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5. INCANDESCENT GAS MANTLES

5.1 PRODUCT DESCRIPTION AND BACKGROUND

5.1.1 General Discussion

Incandescent mantles containing natural thorium have been in use for many years. Incandescent lighting utilizing mantles was a primary source of home lighting, street lighting, and even automobile lighting in the past. For practically any use one might find for an electric bulb today, a gas mantle may have been used for the same purpose in the past. Today, incandescent lighting (mantles) is used primarily for outdoor lighting, particularly camping, and some home use. Mantles are also used sparingly for indoor incandescent lighting. Approximately 20.5 million mantles are used each year by campers and an additional 4.5 million by homeowners (O'Donnell 78).

The history of the use of thorium in gas mantles began in 1885 when the Austrian Baron Carl Auer von Welsbach began investigating the properties of rare earth minerals. He discovered the brilliant light-emitting qualities of their oxides and conceived the basic idea of the incandescent gas mantle (Santmyers 30). The first mantles were constructed of the oxides of lanthanum and zirconium, to which a little cerium oxide was added, but they did not prove to be altogether satisfactory. The amount of light emitted was small compared to the amount of gas they consumed. Further research revealed that a gas mantle impregnated with thoria, which is thorium oxide, and one to two percent of other oxides produced maximum luminosity from the gas burned.

Most incandescent gas mantles were cotton, ramie, or artificial silk fiber woven in hose or tube form. Today rayon fabric is used almost exclusively. In the original process the fabric was bleached, washed in distilled water, and dried in a current of hot air. The fabric was then treated with thorium and cerium oxides and small quantities of beryllium nitrate and magnesium nitrate were added to strengthen the ash solution. The mantle was then toughened with a thorium solution containing alumina and alkaline earths. Finally it was dried, shaped, and burned off to convert the nitrates to oxides.

The use of electric light drastically decreased the use of thorium for illumination. As late as the 1950's, gas mantles still lit the streets of some communities. During World War II mantles by the millions were used by the Armed Forces. In 1952, 65 percent of the thorium production in the U. S. went into gas mantle production (Cuthbert 58). As stated above, the gas mantles today are used primarily for outdoor lighting and, in particular, for portable lanterns.

5.1.2 Product Description

Incandescent mantles are used primarily in gas-fired lanterns and operate in a relatively simple manner. The vortex passes gas from a small supply, usually external to the lantern, to the mantle. The vortex can usually be controlled allowing the user to adjust the brightness or spectral irradiance. The mantle resembles a fine screen or wire mesh which surrounds the vortex. The gas flame produced from the vortex heats the mantle causing heat energy to transfer into light energy which is emitted from the mantle. The thorium in the mantle increases the emissivity of the mantle and the evenness of light produced.

The mantle is prepared by dipping rayon fabric into a solution of the nitrates of thorium, cerium, and beryllium. Magnesium, aluminum, and silicon are also present in small quantities. The cerium functions to improve the whiteness of the incandescence and the beryllium is added to harden the delicate ashlike structure of the mantle. The thorium is the material that incandesces. After drying, the mantle is coated with nitrocellulose in order to fix the salts of the fabric support material. The nitrocellulose also assists in the preburning of the mantle. The preburning is done using a flame in the absence of fuel to prepare the mantle for its initial use. This simple ignition is kindled by the combustible nitrocellulose coating and burns away the fabric support material, converting the thorium, cerium, and beryllium nitrates to their respective oxides.

Thorium becomes incandescent when heated to a temperature between 1375 and 1750^{0} C. The emissivity of thorium oxide with about one percent cerium oxide has a pronounced selectivity in the visible part of the spectrum. The actual location of the light emission is apparently at grain boundaries of crystallites (Ryshkewitch 60).

Gas mantles are fragile and break rather easily. Two types of mantles are available to the consumer. One is hard and very brittle. The other is relatively malleable and the type normally purchased. The difference in physical properties does not effect the exposures received by the consumer.

Each incandescent mantle used for camping and at home contains 250 mg to 400 mg of thorium as oxide (O'Donnell 78). A flattened bag may be 1-10 cm long x 1-7 cm wide. As a result, the user and the general population are exposed to radioactivity. The degree to which the general population and the user are exposed, and the potential pathways by which exposures occur, are discussed in a Section 5.2.

5.1.3 Distribution

Exact numbers of incandescent mantles containing thorium manufactured and distributed each year in the U. S. was not easily obtainable. Barker and Tse (1978) indicated that 20 million mantles were produced in 1974. For purposes of this study it is assumed 20.5 million mantles are used annually by campers and 4.5 million mantles are used annually by homeowners and others (O'Donnell 78). Communication with O'Donnell indicates the probable breakdown of percent usage is:

Portable lanterns	82 percent
Residential lights	
Outdoor	10 percent
Indoor	2 percent
Trailer lights	2 percent
Unspecified (includes commercial signs and lights)	4 percent

5.2 ENVIRONMENTAL IMPACT

5.2.1 Benefits

The benefits associated with incandescent lighting utilizing mantles containing natural thorium include portabilty, ease of use, and a greater amount of light produced per unit (mantle) relative to alternative means of lighting. In addition, the ethereal effect created for camping cannot be quantified but is probably a major reason for purchasing an incandescent light. Each benefit is discussed below.

Portability

The vast majority of incandescent mantles are used for camping today. One of the reasons for this is that this type of lighting is very easy to carry or move and, obviously, does not require an energy supply other than that which travels with the camper, i.e., propane and kerosene.

Ease of Use

Incandescent lights are easy to use and serve many purposes. For example, each light unit can be adjusted for many uses including light for reading, lighting of large ground areas, and use as dim lighting. Other lighting alternatives do not always provide this range of beneficial uses.

Amount of Light Produced per Unit

Incandescent lights can produce considerably more light per unit (mantle) than any other alternative. On a lumen basis, gas mantles can produce five to ten times more light than a battery-operated light of the same size or portability.

Ethereal Effects

Many people when camping will not consider battery operated lighting based merely on the feeling that "they don't want 'electricity' around"(1).

5.2.2 Radiological Impact

Thorium has thirteen known isotopes, of which Th-232 and Th-228 occur naturally. The half-life of Th-232 is 1.4×10^{10} years. The half-life of Th-228 is 1.9 years and it is included in the decay scheme of Th-232. Figure 5.1 illustrates the decay scheme of Th-232.

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(1) Based on a conversation with one of the largest retailers of camping equipment in the Northern Virginia area.

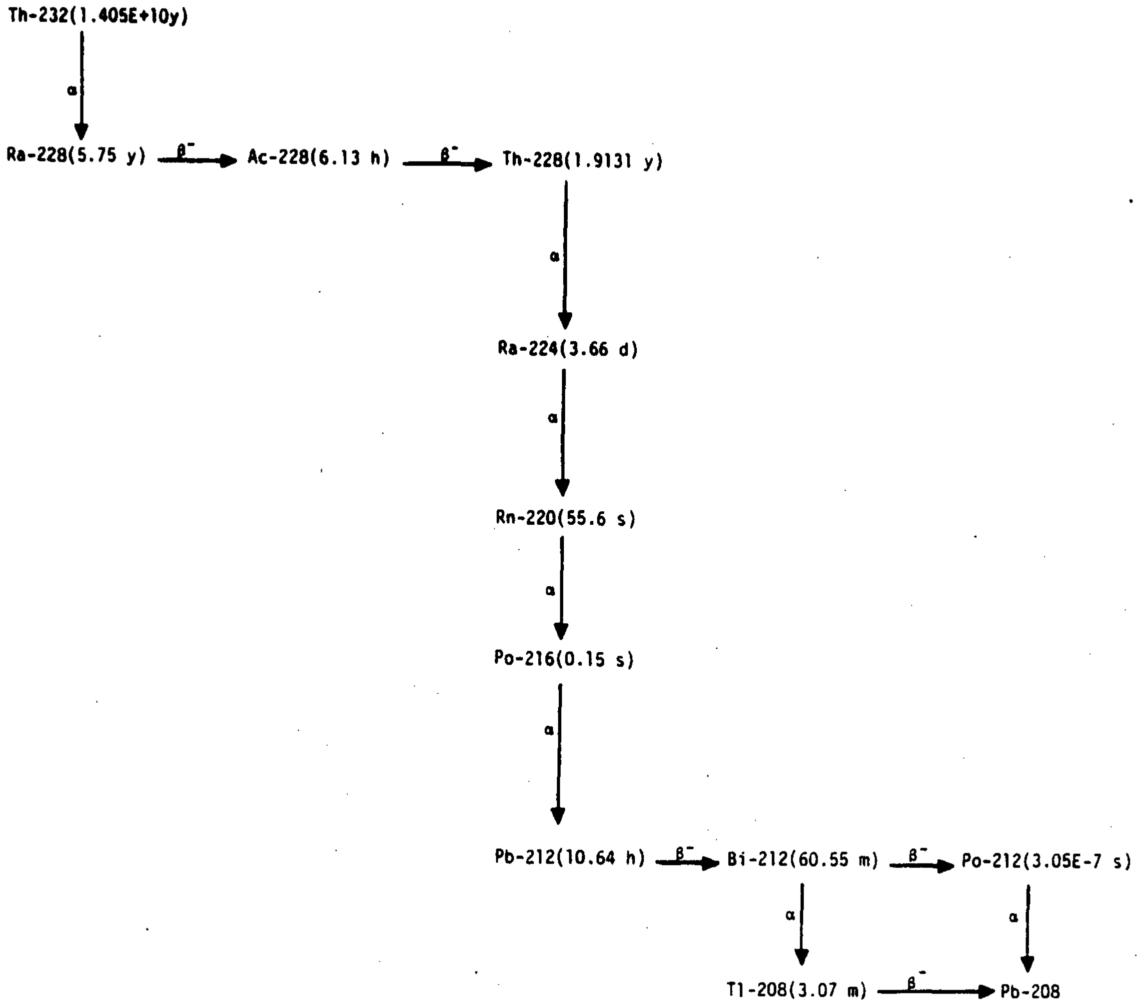

Figure 5.1. Decay Scheme of Th-232 Decay Chain.

The main source of thorium is from mineral monazite, which contains from three percent to nine percent thorium oxide along with mostly rare-earth minerals. When the thorium is commercially processed, the non-thorium elements are removed and only Th-232 and Th-228 remain. Following the processing, the other radionuclides begin to buildup again. Table 5.1 illustrates the buildup and decay of the Th-232 decay scheme assuming at the time of process the relative activities of Th-232 and Th-228 are both unity. After 20 years, the Th-232 daughters have reached about 91 percent of equilibrium, except for Th-228 and its daughters, which are about 87 percent of equilibrium. After 100 years, equilibrium has essentially been reached. It should be noted that thallium-208 (Tl-208) and polonium-212 (Po-212) have relative equilibrium activities of 0.36 and 0.64, respectively, because bismuth-212 (Bi-212) decays to Tl-208 and Po-212 according to those ratios.

This section assesses the total radiological health impact resulting from the manufacture, distribution, normal use, and disposal of thorium incandescent mantles. As in the assessment of other products, the assessment can only be considered to be a general estimate of the health risk associated with the mantles.

The life span of an incandescent mantle will introduce various segments of the population to radioactive material at various times and places. No all encompassing description of the mantle life span can be made in a manner which is easily used. For this reason, it is necessary to model the life span by grouping similar events. These events are manufacture, distribution, use, and disposal.

The distribution scheme envisioned in this study starts at the point of manufacture or entry into the U. S. and proceeds to the consumer. The gas mantles are manufactured domestically or arrive in the U. S. and are stored in warehouses until they are shipped to various distributors or wholesalers. At this point, the mantles are transported to manufacturers of camping equipment or gas home lighting. Here the mantles, as part of camping equipment or gas lights, may be stored in warehouses or transported directly to a retail outlet at which point the consumer buys the mantles. The distribution scheme is displayed in Figure 5.2. This scenario is undoubtedly simplified compared to the actual sequence which may occur, however, the events or types of locations or movements would be similar.

Table 5.1. Relative Activities of Th-232 Decay Chain Members Following Commercial Recovery of Thorium(1).

Time (Years)	Relative Activity(2)										
	Th-232	Ra-228	Ac-228	Th-228	Ra-224	Rn-220	Po-216	Pb-212	Bi-212	TL-208	Po-212
0	1.000	0.000	0.000	1.000	0.000	0.000	0.000	0.000	0.000	0.000	0.000
1	1.000	1.139E-01	1.138E-01	7.145E-01	7.177E-01	7.177E-01	7.177E-01	7.181E-01	7.181E-01	2.585E-01	4.596E-01
10	1.000	7.015E-01	7.014E-01	5.921E-01	5.915E-01	5.915E-01	5.915E-01	5.914E-01	5.914E-01	2.129E-01	3.785E-01
20	1.000	9.106E-01	9.106E-01	8.670E-01	8.668E-01	8.668E-01	8.668E-01	8.667E-01	8.667E-01	3.120E-01	5.547E-01
100	1.000	1.000E+00	1.000E+00	1.000E+00	1.000E+00	1.000E+00	1.000E+00	1.000E+00	1.000E+00	3.600E-01	6.400E-01

(1) Based on calculation at SAI using Basis Computer Code.

(2) At time of recovery (time = 0) relative activities of Th-232 and Th-228 are 1.

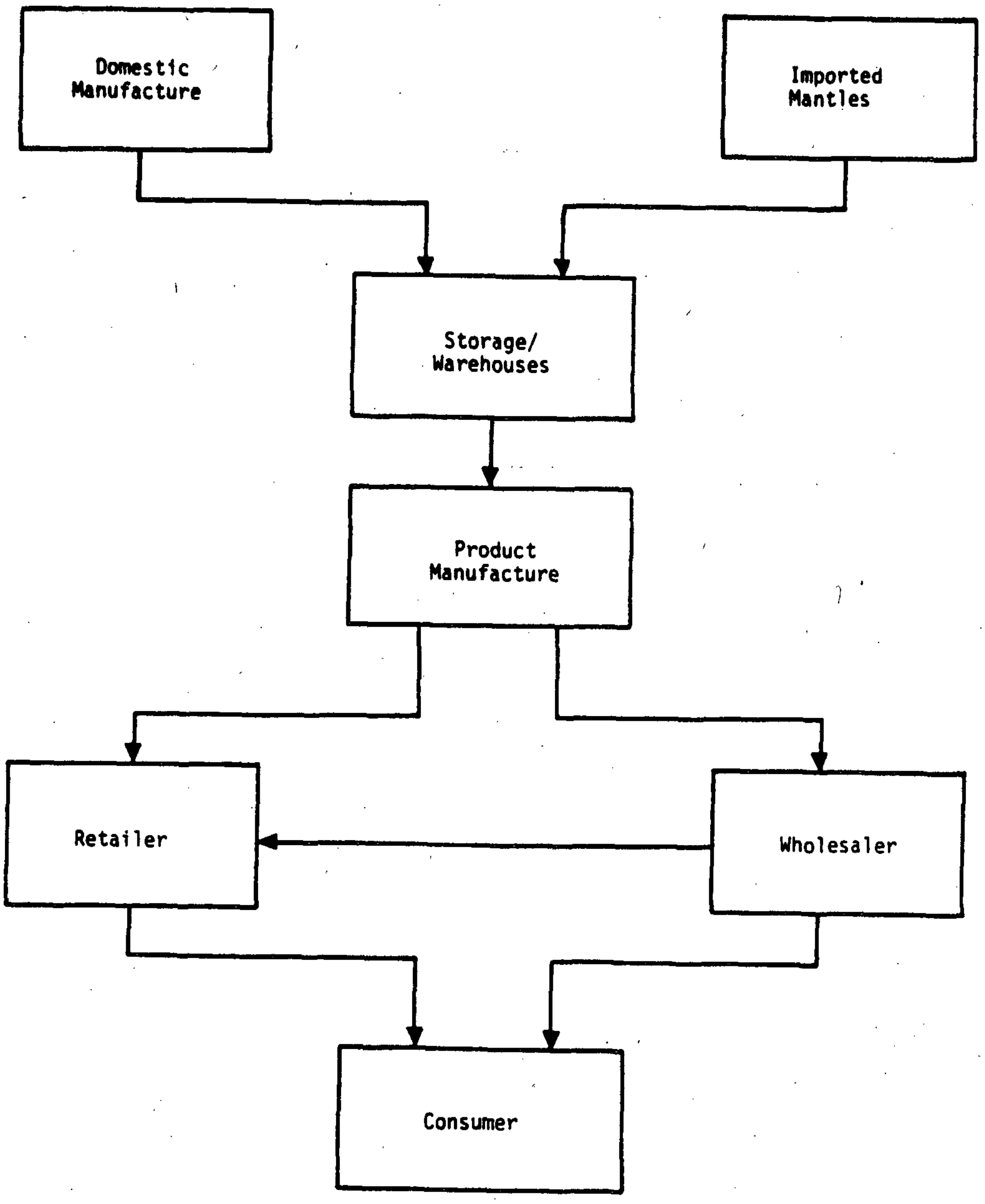

Figure 5.2. Simplified Incandescent Mantle Distribution.

Several assumptions must be made in order to assess the impacts of mantle distribution. These include, for example, the mode by which the mantles are transported, the number of people exposed during distribution, duration of exposures, distances, and source strengths. O'Donnell (1978) has estimated 1.71 million persons are involved in the distribution of mantles. Transportation workers constitute the majority of persons (1.6 million) while distribution workers (warehouse workers) comprise the rest, or 110 thousand persons.

Approximately 1.7 million campers are estimated to be exposed to radioactivity from mantles annually. In addition, 5.1 million persons comprise the campers' families as an additional exposed group. Users of indoor gas-fired lamps having thoriated mantles total about 400 thousand persons, while five million are estimated to be exposed to outdoor gas-fired lamps on an annual basis (O'Donnell 78).

Generally, a mantle is discarded when it has reached the end of its useful life, by breaking, in the same manner as any other article in the home. Gas mantles are fragile and break rather easily. This disposal method would probably lead to the mantle being incinerated or buried in a landfill.

5.2.2.1 Occupational Exposure

Data on occupational exposure proved hard to acquire. Since numerous mantles are made outside of the U. S. and because no satisfactory occupational exposure was found, no data are presented here.

5.2.2.2 Exposures During Distribution

Because of the complexity of the transportation matrix associated with incandescent lamps (gas mantle) distribution, both in terms of exposure times and geometries, it is not possible to model the exposures from this mode in an easy fashion. A distribution model has been developed by Oak Ridge National Laboratory (ORNL) but was not available to include in this study. However, preliminary results using the subject distribution model have been published (O'Donnell 78). The individual dose commitments for transportation workers involved in the distribution of 25 million incandescent mantles containing natural thorium ranged from 5×10^{-5} mrem to 1 mrem annually. The individual dose commitments to distribution workers such as warehouse workers ranged from 7×10^{-4} to 15 mrem annually. Table 5.2 illustrates the results of O'Donnell.

5.2.2.3 Exposures to Users

Table 5.2 presents the results of O'Donnell (1978) for all groups considered in his study. The users include campers, campers' families, indoor lamp users, and outdoor lamp users. O'Donnell states, "very conservative exposure conditions were assumed; more realistic conditions should reduce the predicted doses by a factor of two."

The base assumptions concerning the state and behavior of thorium in the mantles used by O'Donnell to determine doses were: (1) each mantle contains 350 mg of insoluble thorium, (2) the thorium has aged 20 years since chemical purification, and (3) Rn-220 does not emanate from the mantles. O'Donnell conservatively estimated average dose commitments to the total body by assuming the average dose to the total body from inhaled and ingested thorium to be equal to the sum of the average doses to the lung from inhalation and to the gastrointestinal (GI) tract from ingestion.

The model of O'Donnell incorporates assumptions regarding time duration of exposure, distances campers and their families would be from incandescent lamps, and the uses of indoor and outdoor incandescent lights for the home.

Preliminary results which were reported by O'Donnell estimated that campers would receive individual dose commitments ranging from 0.07 to 0.4 mrem annually as a result of the annual use of 20.5 million mantles. Campers' families by the same token might receive 0.01 to 0.08 mrem annually. Users of indoor and outdoor gas lamps could potentially receive individual doses on the order of 0.02 to 0.01 mrem (indoor) and 0 to 0.001 mrem (outdoor).

Outdoor lamp installers, however, received comparatively larger doses ranging from 0.003 to 3 mrem annually. Campers and outdoor lamp installers could ingest or inhale radionuclides as a result of mantle replacement. For example, a mantle changer could get thorium on the hands and fingers while removing a broken mantle and subsequently ingest thorium particulate. In addition, thorium could potentially become airborne and be inhaled while replacing a mantle. These pathways of exposure account for approximately 80 percent of the predicted annual dose commitments to campers and outdoor lamp installers (O'Donnell 78).

Table 5.2. Summary of Potential Annual Dose Commitments From 25 Million Incandescent Mantles Containing Natural Thorium (Base Case)(1)

Exposed Group	Number of Persons In Group	Range of Individual Dose Commitments (mrem)	Population Dose Commitments (person-rem)
Transportation workers	1.6×10^6	5×10^{-5} to 1	5
Distribution workers	1.1×10^5	7×10^{-4} to 15	90
Campers	1.7×10^6	7×10^{-2} to 4×10^{-1}	700
Campers' Families	5.1×10^6	1×10^{-2} to 8×10^{-2}	400
Indoor lamps - users	4.0×10^5	2×10^{-2} to 1×10^{-1}	20
Outdoor lamps - users	5.0×10^6	0 to 1×10^{-3}	14
- installers	1.8×10^3	3×10^{-3} to 3	50
General Public	1.0×10^8	0 to 2×10^{-4}	3
All		0 to 15	1282

(1) From O'Donnell, 1978

The use data of O'Donnell was determined to be adequate for this study. The exposures of campers (1.7 million people) and their families (5.1 million people) were analyzed using generated assumptions. The results follow.

From Table 1.1, the specific activity of Th-232 is found to be 1.1×10^{-7} Ci/g or 1.1×10^{-10} Ci/mg. Since the range of thorium in individual mantles was found to range from 250 to 400 mg, the range of Th-232 activity is 2.8×10^{-8} Ci to 4.4×10^{-8} Ci. If 325 mg is taken to be the average thorium content in a mantle, the corresponding average activity is 3.6×10^{-8} Ci of Th-232. As can be seen in Table 5.1, the activities of the various daughter products vary depending upon time since commercial recovery. For the bulk of his work, O'Donnell used 20 years for the age of the thorium. This number appears to be quite reasonable. If 100 years is used, the members of the Th-232 decay chain will have essentially reached equilibrium.

The exposure rate at one meter from a one Ci source of 20-year old thorium is found to be 1.3 R/hr using Table A.1 in Appendix A. For one Ci of thorium in equilibrium, the exposure rate at one meter is 1.5 R/hr. Daughter products are assumed to contribute to the exposure rate.

The external dose rates to various organs from a typical mantle can be calculated assuming 3.6×10^{-8} Ci of Th-232 content. The results are presented in Table 5.3.

One member of the Th-232 decay chain is a gas, Rn-220. Since this gas may emanate, the emanation rate from a single mantle was calculated and found to be 3.9×10^{-10} Ci/sec. This number will be useful in some of the analyses to follow.

A camper is assumed to receive potential doses or dose commitments from three possible pathways: (1) direct exposure, (2) contamination during mantle replacement, and (3) exposure to Rn-220 and its daughters due to emanation of Rn-220 from the mantles in confinement areas such as a tent or trailer.

To assess the dose commitments to a typical camper, the following assumptions are made:

(1) Each camper has one lamp with one mantle.

(2) Camping is for two weeks per year with ten nights of overnight camping.

(3) The camper spends two hours/day one meter from the lamp.

(4) The camper spends eight hours/day sleeping two meters from the lamp.

Table 5.3. External Dose Rates to Various Organs from Typical Gas Mantle(1)

Dose Rate(2) (rem/hr)

Organ	Total Body	Liver	Bone	Lungs	G.I. Tract
Dose Rate	3.1E-8	2.9E-8	3.0E-8	2.9E-8	2.8E-8

(1) Typical mantle contains 3.6E-8 Ci of Th-232 (20 years since recovery).

(2) Dose rate at one meter.

(5) The camper changes the mantle two times per year.

(6) During the eight hours of sleep, the camper is exposed to Rn-220 emanation into the tent or camper.

(7) The tent/camper volume is 14 m^3 and the air change is two volumes/hour during the night.

(8) When replacing mantles, 0.001 fraction of the activity becomes airborne, of which 0.1 is respirable. Contamination of the hands contains 0.001 fraction of the activity. The fraction ingested is 0.01 and the fraction of respirable airborne activity inhaled in 0.1.

(9) The camper's breathing rate is 20 m^3/day.

The results of calculating the dose commitments via the three pathways are given in Table 5.4. The calculation of dose commitments from inhalation of Rn-220 and its daughters was calculated assuming a near equilibrium in the tent/camper volume. The calculated equilibrium air concentrations in the tent/camper for various volume changes is illustrated in Table 5.5. For this inhalation pathway, the important contributors were found to be Pb-212 and Bi-212. The resulting dose commitments due to contamination resulting from mantle replacement were found to be contributed to most from the ingested Th-232, Ra-228, and Th-228.

Resulting population dose commitments for campers (1.7 million) and their families (5.1 million) are given in Table 5.6. The campers' families are exposed to the same sources as the camper, except for mantle replacement.

Dose commitments to other users as given in Table 5.2 are not calculated here since campers and their families dominate the results. The maximally exposed camper is assumed to camp out 200 days per year and change ten mantles annually. All other assumptions are the same as for the typical camper. Calculations for the maximally exposed individual, assumed to be the maximally exposed camper, were performed and are given in Table 5.7.

The results given in the preceding analysis are dependent upon the assumptions made by the authors. The actual numbers for Rn-220 emanation and mantle replacement contamination are not known, and since the results are dominated by these mechanisms, the results should be looked upon in this context.

Table 5.4. Annual Dose Commitments to a Typical Camper From Thorium Mantles(1)

Dose Commitment (rem)

Pathway	Total Body	Liver	Bone	Lungs	G.I. Tract
Direct Exposure	1.2E-6	1.2E-6	1.2E-6	1.2E-6	1.1E-6
Mantle Replacement	2.5E-4	1.6E-4	4.0E-3	2.2E-3	1.3E-6
Rn-220 Emanation	5.2E-5	1.7E-4	6.3E-4	1.5E-4	4.0E-5
Total	3.0E-4	3.3E-4	4.6E-3	2.4E-3	4.2E-5

(1) See text for assumptions.

Table 5.5. Equilibrium Air Concentrations in a Typical Tent or Camper for Rn-220 Emanation from a Typical Mantle (1)

	Volume Changes (vol/hr)				
Radionuclide	0	0.5	1	2	5
Rn-220	3.1E-8	3.1E-8	3.0E-8	3.0E-8	2.8E-8
Po-216	3.1E-8	3.1E-8	3.0E-8	3.0E-8	2.8E-8
Pb-212	3.1E-8	3.5E-9	1.9E-9	9.4E-10	3.6E-10
Bi-212	3.1E-8	2.1E-9	7.6E-10	2.4E-10	4.3E-11
Tl-208	1.1E-8	7.1E-10	2.5E-10	7.5E-11	1.1E-11
Po-212	2.0E-8	1.3E-9	4.8E-10	1.5E-10	2.8E-11

(1) Assumes one mantle with Rn-220 emanation rate of 3.9E-10 Ci/sec.

Table 5.6. Annual Population Dose Commitments to Campers and Their Families From Thorium Mantles

Dose Commitment (Organ-rem)

Group	Number in Group	Total Body	Liver	Bone	Lungs	G.I. Tract
Campers	1.7×10^6	5.1E+2	5.6E+2	7.8E+3	4.1E+3	7.1E+1
Campers' Families	5.1×10^6	2.7E+2	8.7E+2	3.2E+3	7.7E+2	2.1E+2
Total	6.8×10^6	7.8E+2	1.4E+3	1.1E+4	4.9E+3	2.8E+2

Table 5.7. Annual Dose Commitments to Maximally Exposed Camper From Thorium Mantles(1)

Dose Commitment (rem)

Pathway	Total Body	Liver	Bone	Lungs	G.I. Tract
Direct Exposure	2.4E-5	2.4E-5	2.4E-5	2.4E-5	2.2E-5
Mantle Replacement	1.3E-3	8.0E-4	2.0E-2	1.1E-2	6.5E-6
Rn-220 Emanation	1.0E-3	3.4E-3	1.3E-2	3.0E-3	8.0E-4
Total	2.3E-3	4.2E-3	3.3E-2	1.4E-2	8.3E-4

(1) See text for assumptions.

5.2.2.4 Product Disposal

The thorium mantles are assumed to be disposed of in the same manner as ordinary trash. As in the assessment of all consumer products, the disposal will be assumed to be ten percent by incineration and 90 percent by landfills.

Waste Disposal

The exposures of the waste collection population is expected to be very low because of the low external dose rates from mantles and because of the rather large shielding factors involved. For these reasons, dose to the waste collection population is not performed in this study.

Disposal in Landfills

The radiological impact resulting from the disposal of thorium mantles in landfills is assessed according to the methodology and assumptions described in Appendix D. The assessment assumes the disposal of 25 million mantles per year. Each mantle is assumed to contain thorium processed 20 years prior to disposal with a Th-232 content of $3.6\text{x}10^{-8}$ Ci. Thorium is an element which occurs naturally in the earth's crust, and landfill disposal may be viewed by some as simply returning the material to the place where it came. According to Appendix D, thorium does not move or migrate very fast, but instead actually migrates extremely slowly. Calculations using the Appendix D methodology and assumptions result in a migration time of $4.8\text{x}10^{5}$ years for thorium to reach the groundwater withdrawal point for the reference site. The authors of this report feel that because of the great time lapse from initial disposal to arrival at the point of water withdrawal, any attempt to estimate future doses or health effects is meaningless. For this reason, only a "worst case" assessment is performed here.

The worst case assessment described in Appendix D results in a 310 year migration time for thorium to reach the point of water withdrawal. A look at the Th-232 decay scheme in Figure 5.1 shows that only three radionuclides have half-lives greater than a few days, Th-232, Ra-228, and Th-228. As shown in the paragraph above, the thorium radionuclides migrate very slowly. Radium migrates faster and using the assumptions in Appendix D the migration time for radium to reach the point of water withdrawal for the "worst case" site is 3.1 years. For

Table 5.8. Fifty-year Organ Dose Commitments from Worst Case Scenario of Landfill Disposal of 25 Million Thorium Mantles.

Organ	Collective Fifty-year Dose Commitment (Organ-rem)	
	Th-232(1)	Ra-228(2)
Total Body	2.8E+6	2.8E+4
Liver	5.1E+3	4.6E+0
Bone	2.8E+6	2.6E+4
Lungs	2.8E+6(3)	2.8E+4(3)
G. I. Tract	2.9E+3	2.8E+2
Gonads	2.8E+6(3)	2.8E+4(3)

(1) Dose commitments from ingestion of 2.3E+4 uCi of Th-232 in equilibrium with its daughters

(2) Dose commitments from ingestion of 2.3E+2 uCi of Ra-228 in equilibrium with its daughters

(3) Dose conversion factor (DCF) data unavailable; DCF assumed to be same as for total body

the reference site, the migration time is 560 years or nearly 100 half-lives for Ra-228, so essentially no initial Ra-228 will reach the water source for the reference site.

The worst case assessment assumes that the thorium isotopes and Ra-228 will be leached from the mantles at the rate of one percent per year from previously incinerated mantles and 0.1 percent per year from intact mantles. The results for the worst case landfill disposal of 25 million thorium mantles is presented in Table 5.8.

Radon emanation is not considered in the landfill assessment because Rn-220 has a short half-life of 55.6 seconds and because the mantles are buried under soil.

Incineration

Ten percent of all disposed thorium mantles are assumed to be incinerated. Typical incinerator temperatures are controlled between 1000 to 1300°C. The work of Griggs (1973) indicated that the thorium in mantles did not appear to volatilize and become airborne during use. For incineration, it will be assumed that one-tenth of Th-232 and its daughters, except for Rn-220, will be released from the mantles. All Rn-220 in the mantles at time of incineration will be assumed to be released.

Calculations were performed to estimate the dose commitments received by the populations surrounding incineration sites. Ten percent of all Th-232 and daughters (100 percent Rn-220) were assumed released from the mantles. Ten percent of this (100 percent Rn-220) was released to the atmosphere and dispersed. Buildup and decay of all progeny and parent radionuclides were considered. The results are given in Table 5.9. The maximum individual dose commitments are estimated to be 2.2×10^{-4} rem to the bone and 9.4×10^{-6} rem to the total body. The total population dose commitment from the release is estimated to be 2400 rem to the bone and 110 rem to the total body. The Rn-220, or thoron, releases and resultant dose commitments are found to be insignificant when compared to the release of the other thorium chain radionuclides.

Table 5.9. Fifty-year Dose Commitments Resulting from the Incineration of 25 Million Used Thorium Mantles.

	Dose Commitment (Person-rem or Organ-rem)											
	Total Body		Liver		Bone		Lungs		G. I. Tract		Gonads(1)	
	I(2)	II(3)	I	II	I	II	I	II	I	II	I	II
Maximally Exposed Individual(4)	9.4E-6	5.0E-11	9.1E-6	1.6E-10	2.2E-4	6.8E-10	1.2E-4	1.5E-9	2.9E-8	3.3E-12	9.4E-6	5.0E-11
Average Individual(4)	4.8E-6	3.0E-11	4.6E-6	9.8E-11	1.1E-4	4.1E-10	6.0E-5	9.0E-10	1.4E-8	2.0E-11	4.8E-6	3.0E-11
Population - One Incinerator	3.6E-1	2.1E-6	3.4E-1	7.2E-7	8.2E+0	3.0E-5	4.3E+0	6.5E-5	1.0E-3	1.5E-6	3.6E-1	2.1E-6
Total Population	1.1E+2	6.5E-5	1.0E+2	2.1E-4	2.4E+3	8.9E-3	1.3E+3	1.9E-2	3.1E-1	4.5E-4	1.1E+2	6.5E-5

(1) Dose conversion factor (DCF) data unavailable; DCF assumed to be same as for total body

(2) Column I dose commitments are from Th-232 and daughters

(3) Column II dose commitments are from Rn-220 and daughters

(4) For the Rn-220 assessment, the maximally exposed individual is assumed to reside at a distance from the stack of 300 meters; the average individual is assumed to reside at a distance of 1,000 meters

5.2.2.5 Accidents

Accidents involving thorium mantles are essentially the same accidents involving other consumer products, namely, fires and misuse. For this assessment a warehouse fire and misuse by a small child will be considered.

Warehouse Fire

As has been stated in the previous section on the incineration of used thorium mantles, the work of Griggs (1973) indicates that thorium in mantles does not appear to volatilize and become airborne during use. In order to at least bound the upper dose commitments due to release of Th-232 and daughters from fires, ten percent of the activity is assumed released in a fire. The following assumptions are used to assess a warehouse fire:

(1) One-tenth of the radioactivity of 10,000 thorium mantles is released during combustion.

(2) Each mantle contains 3.6×10^{-8} Ci of Th-232 plus daughters due to 20 years since recovery.

(3) The warehouse volume is 3.0×10^{9} cm^{3}.

(4) The breathing rate of firefighters is 1.2×10^{6} cm^{3}/hr.

(5) Firefighters are exposed for eight hours during the fire and during cleanup.

(6) A 100-fold reduction in activity intake is assumed due to air changes and respiratory protection.

Results of the assessment are presented in Table 5.10. Using the same assumptions and assuming 400 ug of beryllium is released per mantle, the average beryllium air concentration is found to be about 13 ug per cubic meter, which is below the recommended 25 ug per cubic meter. (See Section 5.2.3 for information.)

Misuse by a Small Child

After a thorium mantle has been used, a small child could then possibly play with the mantle and consume it. For this assessment a small child is assumed to eat or swallow the mantle. The resulting fifty-year dose commitments are given in Table 5.11. The resultant dose commitments are not excessively high

Table 5.10. Fifty-year Dose Commitments to Firefighters from a Warehouse Fire Involving Ten Thousand Thorium Mantles.

Organ	Dose Commitment(1) (Organ-rem)
Total Body	2.9E-1
Liver	2.8E-1
Bone	6.7E+0
Lungs	3.7E+0
G. I. Tract	8.6E-4
Gonads	2.9E-1(2)

(1) Dose commitments per firefighter involved from inhalation of ten percent of Th-232 and daughters (20 years after thorium recovery)

(2) Assumed to be same as for total body

Table 5.11. Fifty-year Dose Commitments to a Small Child from Intake of One Used Thorium Mantle.

Organ	Dose Commitment(1) (Organ-rem)
Total Body	1.6E+1
Liver	3.2E-2
Bone	1.4E+1
Lungs	1.6E+1(2)
G. I. Tract	1.6E-1
Gonads	1.6E+1(2)

(1) Dose commitments to a small child are conservatively assumed to be four times equivalent adult dose commitments

(2) Assumed to be same as for total body

and since it is the belief of the authors that the amount of ingested mantle would be much less, such an accident or misuse is not judged to be critical.

5.2.3 Non-Radiological Impact

The emission of toxic metal fumes from mantles during burning presents a non-radiological health hazard. The presence of beryllium poses the greatest threat. Griggs (1973) conducted tests on eight new unused mantles. Analysis showed that the beryllium content ranged from 550 to 700 ug with the average being about 650 ug. After one hour of use in a lantern, the beryllium content of the mantle residues was found to be 112 to 288 ug with the average being about 200 ug. Most of the missing 400 ug of beryllium had volatilized and became airborne during the first 15 minutes of mantle use.

The inhalation of beryllium may produce two forms of beryllium disease. The chronic form can develop from exposure to low concentrations of beryllium over long periods of time. To protect against this hazard, industrial hygienists recommend that exposures not exceed 2 ug per cubic meter of air averaged over an eight-hour period. Non-occupational limits are 0.01 ug of beryllium per cubic meter of air averaged over a month (HGS 64).

The acute form of beryllium disease could result from a single exposure of beryllium over a short time. To prevent this form, it is recommended that concentrations in excess of 25 ug of beryllium per cubic meter of air never be exceeded (HGS 64).

For the sake of comparison, Griggs found a dual mantle lantern gave off about 260 ug of beryllium during initial burning. He estimated a large camper to have a volume of 14 cubic meters, which, considering a uniform concentration, results in a camper air concentration of 18 ug per cubic meter. He also felt the process of lighting a new mantle was "particularly hazardous" since there is a greater potential for inhalation during this time (because the operator's face is in close vicinity to the mantle).

Other non-radiological impacts essentially are not related to the use of thorium in the lanterns. The biggest impact is due to the risk of fires because of the potential to spill fluid when a lantern is used.

5.2.4 Summary of Impact

A summary of the collective dose commitments and resultant health effects due to the use of 25 million thorium mantles and their disposal is given in Table 5.12. The largest contribution to the total impact is from use of thorium mantles by campers and their families. The mechanisms most reponsible for exposure were mantle replacement by campers and Rn-220 emanation from lanterns used by campers and their families. It is believed that the reported health effects represent an upper bound of the actual impact. It is not easy to assess the level of uncertainty associated with assumptions and estimates used. Accidents were found to be potential health hazards, but were not found to be excessive.

5.3. ALTERNATIVES

5.3.1 Radiological Alternatives

The function of thorium in mantles is not related to its radioactive properties. For this reason, a discussion of radiological alternatives is not warranted.

5.3.2 Non-Radiological Alternatives

Alternatives to incandescent lights using thorium mantles are any sources that produce light of a nature sufficient to be used in place of the incandescent lights. Such alternatives include candles, oil-burning lamps, and electric devices such as flashlights. Because of ease of convenience desired and amount of light required, only electric devices prove to be good alternatives to the thorium mantle devices. A discussion of the use of electric devices versus incandescent lights using thorium mantles follows.

5.4 COST AND RISK VERSUS BENEFIT

The use of thorium mantles does not serve a safety function in most applications. However, in certain emergencies requiring a portable light source,

Table 5.12. Total Number of Estimated Health Effects from Use and Disposal of 25 Million Thorium Mantles. (1)

Organ	Collective Dose Commitment (Organ-Rem)	Cancer Deaths or Genetic Defects
Total Body	8.9E+2	6.2E-2
Liver	8.7E+2	1.3E-2
Bone(2)	5.6E+3	1.4E-1
Lungs	6.2E+3	1.6E-1
GI Tract	2.8E+2	4.2E-3
Gonads	8.9E+2	1.8E-1
Total		5.5E-1

(1) Includes landfill disposal for reference case.

(2) Includes bone tumors and leukemia.

a safety function is served. In this section, the use of thorium mantles in lamps is compared to other non-radioactive alternatives.

5.4.1 Cost

The only alternative to be discussed is an electric device such as a flashlight. The retail cost of an incandescent light (one mantle) utilizing propane gas as fuel is about $30. Two and three mantle lights are available for prices up to $45. Individual replacement mantles cost $1-2 apiece. The retail cost of a battery-operated neon or fluorescent light is typically about $15. Such units use eight D-size batteries or the equivalent.

5.4.2 Product Analysis

Discussion of cost and risk versus benefit for thorium mantles in incandescent lights will not be made because of the difficulty of relating a convenience benefit to cost or risk. The risks associated with the use of thorium mantles are discussed in Section 5.2.

5.4.3 Alternatives Analysis

A comparison of incandescent lights with thorium mantles versus battery-operated lights can be made in terms of benefits and cost-effectiveness. In terms of benefits, the use of incandescent lights with thorium mantles has a number of significant advantages. The three most important advantages are:

(1) Incandescent lights produce considerably more light per unit than any alternative. On a lumen basis, gas mantles can produce five to ten times more light than a battery-operated light of the same size or portability.

(2) Incandescent lights are easy to adjust for various lighting situations. They can be used for flood lighting as well as for dim lighting. Such a range of lighting is not always possible with alternatives.

(3) Incandescent lights operate for longer periods of time (estimated to be 20 to 30 percent longer) for a typical supply, i.e., a propane cartridge, than for a typical battery source such as eight D-size batteries.

Disadvantages of using incandescent lights with thorium mantles include:

(1) Potential for fires because of the fuel supply required.

(2) Exposure to Th-232 and its progeny.

Because of the inability to easily compare intangible benefits such as those listed above, it is very hard to perform a cost-effectiveness comparison. The cost of using incandescent lights with thorium mantles can not be justified or dismissed on this bases.

5.5 FUTURE PRODUCTS

The use of decorative lighting with thorium mantles appears to be on the decline, owing mostly to the growing emphasis on energy conservation. The use of thorium mantles for camping or outdoor occasions should remain constant or increase slightly. For these reasons, it is estimated that thorium mantle usage should remain at about 25 million units per year up until 1986.

5.6 SUMMARY

The health hazards or risks due to thorium mantles were found to be of two types. The first is due to the radioactive nature of Th-232 and its progeny. The second is due to the use of beryllium in mantles and the potential for the development of beryllium disease. The radiological health hazards or risks are reported in Table 5.12. As can be observed, the upper estimate indicates about one fatal cancer per year of use of 25 million mantles may result.

The use of thorium mantles results in some benefits which are not available with the major alternative, battery-powered lights. The main benefits are light output, variability of light output, and length of typical operation.

5.7 REFERENCES FOR SECTION 5

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6. THORIATED TUNGSTEN WELDING RODS

6.1 PRODUCT DESCRIPTION AND BACKGROUND

6.1.1 General Discussion

Several different welding processes are currently used by industry. These processes include brazing, flow welding, resistance welding, induction welding, thermit welding, gas welding, forge welding, and arc welding. The process considered in this assessment is arc welding. Arc welding produces coalescence of two metal objects by heating the work metal with an electric arc or arcs (depending on the number of "passes" required) with or without applying pressure and with or without using filler material (Phillips 58). Arc welding equipment uses one of two kinds of electrodes: a carbon electrode or a metal electrode, which in turn, may be shielded or unshielded. The electrode of concern here is a nonconsumable metal electrode, and is shielded with an inert gas. One type of metal electrode is the tungsten electrode. Tungsten electrodes consist of pure tungsten or are alloyed with ThO_2 (historical name "thoria") or ZrO_2. Alloyed tungsten electrodes, particularly the thoriated type, have advantages relative to pure tungsten. Reported advantages include easier starting, greater arc-stability, and less weld metal contamination (Breslin 52, Gibson 52, Winsor 57). Approximately 85-88 percent of the tungsten electrodes manufactured today are thoriated (Bergen 80).

There are three classes of thoriated-tungsten electrodes. The three classes are distinguishable by the weight percent of thoria in each electrode. These classes are as follows:

0.35 - 0.55 percent ThO_2
0.8 - 1.2 percent ThO_2
1.7 - 2.2 percent ThO_2

The first class (0.35 - 0.55 percent ThO_2) is seldom used in the Tungsten Inert

Gas (TIG) arc-welding process. Almost all of the electrodes produced for and used in the TIG process contain one to two percent thoria. The most predominant electrode used by far contains approximately two percent thoria.

Almost all of the electrodes produced in the U. S. are sold to manufacturers of welding equipment who are primarily producers of industrial gases. There are three major producers of industrial gases in the U. S. In addition, some electrodes are sold directly to "wholesale distributors." This, however, is reportedly seldom done.

The TIG process is used primarily in the "premium industries" due to its relatively high cost. These industries include nuclear power plant construction, aircraft industry (welding titanium), petrochemical industry (welding stainless steel), and the food processing equipment industry. Gas and oil piping and shipyards use the TIG process to a lesser extent.

6.1.2 Product Description

An electrode is merely a metal wire which, in this case, serves to produce an electric arc. Tungsten is considered the best material for this use because it has an extremely high melting point and is not consumable. Moreover, thoriated-tungsten electrodes have a higher electron emissivity, better current carrying capacity, longer life, and greater resistance to contamination compared to pure tungsten.

Thoriated-tungsten electrodes are produced in nine standard diameters ranging from 0.254 mm to 6.35 mm, and six standard lengths ranging from 7.62 cm and 60.96 cm (Phillips 68). The electrode diameter is generally chosen to allow the electrode to operate at near-maximum current carrying capacity (ASM 71). The most common thoriated-tungsten electrode diameter is 3/32 inches (Bergen 80), and this is a direct result of the current carrying capacity. Generally, a welder will repoint an electrode (break a piece off and resharpen the tip) four to five times. The less experienced welder may contaminate the electrode or otherwise compromise the integrity of the equipment causing the electrode to be discarded before the end of its normal lifetime.

Radioactive thorium is used in the manufacturing of electrodes used in the TIG process. These radionuclides include Th-232 and Th-228. External exposure to radioactive thorium and daughters occurs as a result of exposure to direct radiations emitted during decay of isotopes present in the thorium decay chain, and to bremmstrahlung produced within the tungsten rod. Internal exposure due to inhalation of radioactive thorium and daughters occurs as a result of gaseous Rn-220 (historical name "thoron") released from the electrodes, and from radioactive materials volatilized during welding and incineration of discarded rods. McDowell-Boyer (1979) assumed for the purposes of calculating doses (internal and external) that the radioactive thorium (Th-232 and Th-228) were separated from daughter products 20 years prior to the marketing date of the electrode. Thus, the radioactive daughters of Th-232 and Th-228 would have achieved 91 percent and 87 percent equilibrium with the two thorium isotopes, respectively.

6.1.3 Distribution

Information is not available from published references regarding the number of thoriated tungsten electrodes produced and used annually in the U. S. However, based upon conversations with the largest domestic producers of thoriated-tungsten electrodes, it is estimated that approximately 3000-3300 pounds (1364-1500 kg) of thoria are used each year in the U. S. for the production of these electrodes. A reasonable assumption is that production meets demand. In addition, electrodes made in the U. S. are generally not exported. Foreign producers can make welding electrodes more cheaply than U. S. manufacturers as a result of the higher standards placed on domestic welding rods. Thus, electrodes manufactured in the U. S. are not competitive on the foreign market. Similarly, electrodes manufactured overseas are generally not imported because they do not meet U. S. standards.

Of all the electrodes produced annually, the average weight is 0.0144 kg (0.0317 pounds). Assuming that the predominant weight percent thoria is two, the number of electrodes produced and used in the U. S. annually is approximately $5.2x10^6$.

6.2 ENVIRONMENTAL IMPACT

6.2.1 Benefits

Alloyed tungsten welding rods, particularly the thoriated type, have important advantages over pure tungsten electrodes. The reported advantages include easier starting, higher electron emissivity, better current carrying capacity, greater arc-stability, greater resistance to weld contamination, and longer life. In addition, thorium metal added to tungsten arc welds improves the quality of the joins (Cuthbert 58). It would be very difficult to quantify the incremental increase in benefit obtained by adding thorium to welding rods. The fact that thoriated rods have important advantages over pure tungsten rods (or rods using other materials alloyed with tungsten) makes this type of material highly desirable for specific applications, many of which can be considered safety-related (e.g., nuclear power plant or aircraft welding). High standards are set for such applications, and incentives exist for manufacturers to use materials such as thorium if it will result in a superior product. It will be assumed for the sake of this analysis that the added benefit gained by incorporating thorium into tungsten welding rods is substantial, although not quantifiable.

6.2.2 Radiological Impacts

As is the case with other products, the radiological impacts of thoriated-tungsten welding rods must be summed over their total life span. For normal usage, the radiological impacts will be from external doses to exposed individuals, and from inhalation of radioactive material where possible. In order to evaluate the radiological impacts, the dose to the maximally exposed individual and the collective dose to the exposed population are reported for each specific life span activity. The maximally exposed individual is defined as the individual who receives the largest dose or dose commitment of any person in the exposed population associated with the specific life span activity. Dose commitments resulting from inhalation or ingestion of radionuclides utilize fifty-year dose commitment factors, which relate the dose an individual or organ will receive as a result of retention of radioactive material in the body during the 50 years following the intake (see Appendix A).

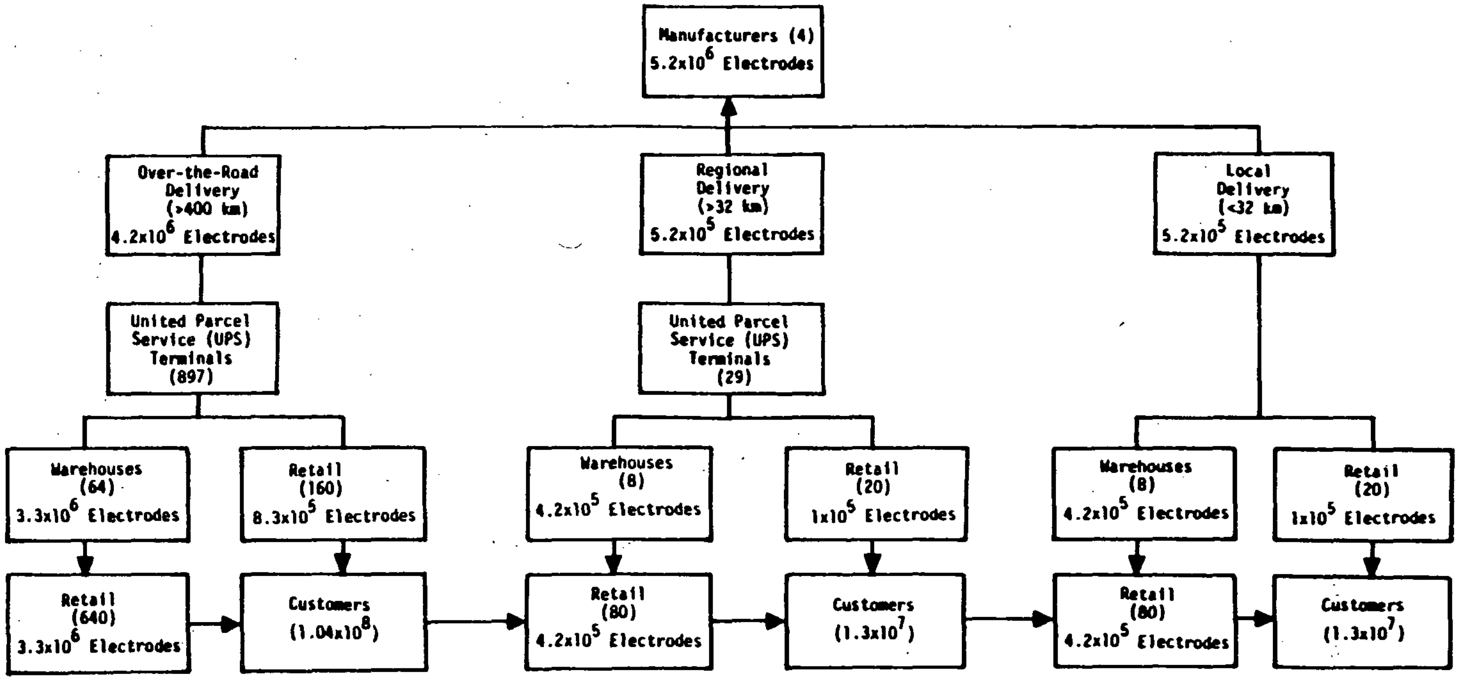
The following sections serve to summarize results from studies previously performed, and to present additional estimates of the doses resulting from the life span activities associated with thoriated-tungsten welding rods. Results are presented in terms of current annual production and usage. It is evident after reviewing previous work that specific information is lacking concerning many important factors. Assumed values have been necessarily employed in a number of calculations. However, based upon best estimates and conversations with other people knowledgeable in the welding industry, assumptions which have been made would lead to conservative results.

McDowell-Boyer (1979) has estimated doses received by distribution personnel, welders and nearby persons, and the general population from life span activities associated with thoriated welding rods. Many of the doses reported in this section are attributable to the McDowell-Boyer report. Since those results were based on an annual usage of 1.0×10^6 electrodes, the calculated doses have been scaled upward by a factor of 5.2 to coincide with this study's estimation of the number of electrodes produced and used annually.

Because of the complexity of the transportation matrix and the number of different points associated with the distribution of welding rods, both in terms of exposure times and geometries, it is not possible to model the exposures from this mode in one easy fashion. Study has led to the conclusion that the methodology used by McDowell-Boyer is as acceptable as can be found.

Distribution, use, and disposal of thoriated-tungsten electrodes will expose various segments of the population to radiation at various times and places. No all-encompassing description of the electrode's life span can be made in a manner which is easily used. In general, the electrode will pass through several tiers of distribution before it ever reaches the welder. For this reason, it is necessary to model the life span by grouping similar events or occurrences. The life span (Figure 6.1) can be divided into the following basic phases:

(1) Delivery
(2) Storage in warehouses (wholesale)
(3) Retail distribution
(4) Consumer use (welder, occasional welder, nonwelder)
(5) Disposal (general population)



(1) From McDowell-Boyer, 1979

(2) Numbers in parentheses reflect the number of distribution units of given type assumed to be involved at each step.

Figure 6.1. Flow Diagram for Distribution of 5.2×10^6 Thoriated-Tungsten Welding Electrodes$^{(1)(2)}$.

Consequently, doses can be calculated for the following population groups for Phases 1-3 inclusive:

(1) Truck drivers
(2) United Parcel Service (UPS) terminal employees
(3) Retail employees
(4) Warehouse employees
(5) Customers

Doses can be calculated for the heavy welder, occasional welder, and nonwelder (i.e., a person receiving a dose due to his or her close proximity to TIG processes) for Phase 4. Also, a potential dose can be calculated for the general population as a result of the disposal of thoriated-tungsten welding electrodes. Doses received under abnormal circumstances have also been considered.

6.2.2.1 Occupational Exposure

Data concerning exposures to workers involved in the manufacture of thoriated welding rods are not available. External exposures are probably very low since less than 1 Ci is used annually, and since the external radiation levels from thorium are relatively low. The level of internal exposure incurred by workers in the thoriated electrode manufacturing industry is highly dependent upon such matters as the nature and degree of automated processing, applied industrial hygiene practices, etc.

6.2.2.2 Exposures During Distribution

The exposures incurred by individuals and population groups during the transportation, storage, and retail sales phases of welding rod distribution are listed in Table 6.1. As can be seen from this table, the maximally exposed individual receives less than 1 mrem while the collective total body dose to the distribution population is only 3.8 person-rem. Since these doses are so far below those which would be required to produce one health effect, the calculation of doses to specific organs and resultant health effects from this mode of exposure is not warranted.

Table 6.1. Individual and Collective Total Body Doses Received During Distribution of 5.2×10^{6} Thoriated-Tungsten Welding Electrodes.

Population Group	Maximum Individual Dose (mrem)	Typical Individual Dose (mrem)(2)	Collective Dose (person-rem)
Truck Drivers	2.0×10^{-1}	1.5×10^{-3}	2.7×10^{-2}
UPS Terminal Employee	1.5×10^{-1}	1.9×10^{-4}	5.1×10^{-2}
Retail Employee	6.0×10^{-2}	6.0×10^{-2}	1.6
Warehouse Employee	1.5×10^{-1}	1.5×10^{-1}	3.2×10^{-1}
Customers	4.5×10^{-4}	1.4×10^{-5}	1.8
Total			3.8

(1) Modified from McDowell-Boyer, 1979

(2) Typical values represent average dose received by all individuals in each group

6.2.2.3 Exposures to Users

Exposure to thorium and its daughters may occur during use and preparation for use of thoriated welding electrodes through various modes. External exposure may occur during handling and storage of thoriated-tungsten electrodes in tool boxes. Persons exposed to external radiation include both welders and nonwelders in the vicinity of welding. Internal exposure may occur through inhalation of fumes containing thorium and daughters, and of thoriated particulates that become airborne during grinding of the electrodes to form a tip. Persons internally exposed may also include welders and nonwelders.

Volatilized thorium and its radioactive daughters may be contained in welding fumes due to the high temperatures achieved at the tip of the electrode during TIG arc welding. Thorium loss rates during welding have been estimated to range from 0.8 mg/hr to 6.0 mg/hr (Breslin 52, Gibson 52, Winsor 57). Dose ranges are presented to account for this variance.

Airborne concentrations of thorium and its daughters resulting from volatilization during welding depend on the fraction of the material lost that may condense on metallic surfaces present or be included in the molten metal weld. The doses calculated by McDowell-Boyer reflect a range in assumed volatilization efficiencies ranging from 20 to 100 percent due to a lack of consistent results from data surveyed. Both ranges for volatilization efficiencies and thorium loss rates were incorporated into the dose calculations.

Inhalation of thoriated particulates which may become airborne during tapering or tip grinding of electrodes is an additional mode by which welders may become exposed to radionuclides. The magnitude of the exposure depends on the amount of material lost by grinding, and the portion of that material which becomes airborne. It was assumed by McDowell-Boyer that ten percent of the material lost during grinding becomes airborne.

Doses for individuals and exposed populations as a result of the actual use of thoriated-tungsten electrodes have been estimated. Table 6.2 presents individual and collective total body doses for three population groups. Typical dose commitments represent average individual doses estimated for each group.

Table 6.3 presents the maximum fifty-year dose commitments to the entire population of users and bystanders under the most conservative assumptions. These doses, which are derived on the basis of the total body and bone doses reported by McDowell-Boyer, are probably substantial overestimates of

Table 6.2. Individual and Collective Total Body Doses and Dose Ranges Received During Use of 5.2x10^{6} Thoriated-Tungsten Welding Electrodes.

Population Group	Maximum Individual Dose (mrem)		Typical Individual Dose (mrem)(1)		Collective Dose (person-rem)	
	Internal(2)	External	Internal(2)	External	Internal(2)	External
Welders-heavy users(3)	6.8E-1 - 2.5E+1	1.5E-1	3.7E-1 - 1.4E+1	8.4E-2	2.5E+1 - 9.4E+2	5.7E+0
- with home welding(4)	2.4E+0 - 8.8E+1	1.7E-1				
Welders-occasional users(5)	1.3E-1 - 5.0E+0	8.4E-3	7.2E-2 - 2.7E+0	4.6E-3	4.7E+0 - 1.8E+2	3.0E-1
Non-welders(6)	1.3E+0 - 1.0E+1	2.4E-3	5.6E-1 - 4.2E+0	8.0E-4	2.2E+2 - 1.7E+3	3.2E-1
Total					2.5E+2 - 2.8E+3	6.3E+0

(1) Typical values represent average dose received by all individuals in each group

(2) Value represents a 50-year dose commitment

(3) Welding five days per week, 50 weeks per year

(4) A few heavy users (260) who also did welding in their home

(5) Welders working one day per week

(6) Three individuals in addition to the welders were exposed to fumes and external radiation

Table 6.3. Maximum Collective Fifty-year Organ Dose Commitments (Organ-rem) Received from Use of 5.2×10^6 Thoriated-Tungsten Welding Electrodes.

Population Group	Total Body		Liver		Bone		Lungs		G.I. Tract		Gonads	
	Internal	External	Internal	External	Internal	External	Internal	External	Internal	External	Internal	External
Welders (heavy users)(1)	9.4E+2	5.7E+0	9.0E+2	5.3E+0	2.2E+4	5.5E+0	1.3E+4	5.4E+0	3.2E+0	5.0E+0	9.4E+2	6.0E+0
Welder (occasional users)(2)	1.8E+2	3.0E-1	1.7E+2	2.8E-1	4.1E+3	2.9E-1	2.6E+3	2.9E-1	6.1E-1	2.6E-1	1.8E+2	3.2E-1
Non-Welders(3)	1.7E+3	3.2E-1	1.6E+3	3.0E-1	3.9E+4	3.1E-1	2.4E+4	3.1E-1	5.8E+0	2.7E-1	1.7E+3	3.4E-1
Total	2.8E+3	6.3E+0	2.7E+3	5.9E+0	6.5E+4	6.1E+0	4.0E+4	6.0E+0	9.6E+0	5.5E+0	2.8E+3	6.7E+0

(1) Welding five days per week, 50 weeks per year

(2) Welders working one day per week

(3) Three individuals in addition to the welders were exposed to fumes and external radiation

the actual collective doses. They are of value, however, in delineating the upper bound of radiological impact from this life span activity.

The group doses and the total collective doses given in Tables 6.2 and 6.3 were calculated by summing individual doses estimated in each group. The maximum individual fifty-year dose commitment to bone for welders was estimated at between 55 and 2,000 mrem for a one-year exposure. Welders not engaged in welding at home and occasional welders were estimated to receive a bone dose commitment of 16 mrem to 575 mrem and 1.3 mrem to 115 mrem, respectively. A maximum individual bone dose commitment range between 30 and 230 mrem was estimated for nonwelders. External doses for all group members were estimated to be less than 1 mrem.

A maximum potential collective dose commitment is estimated at ranging from $5.6x10^3$ to $6.5x10^4$ rem to the bone per year of exposure to $5.2x10^6$ electrodes. Both internal and external bone doses are included in this range. Collective total body dose commitments were calculated to range between $2.5x10^2$ to $2.8x10^3$ person-rem. These ranges represent the collective fifty-year dose commitment received by an assumed $5.2x10^5$ individuals involved directly or indirectly in the use of $5.2x10^6$ thoriated welding electrodes annually.

6.2.2.4 Product Disposal

Waste Collection

Although a large fraction of the thoriated electrodes disposed of annually are collected and processed as industrial waste, this assessment assumes disposal of the electrodes as municipal refuse. Exposures incurred as a result of waste collection are very low, so a refinement of this assessment is deemed unwarranted.

If it is assumed that the residual thorium content of used electrodes is one-half the initial content, the total external radiation to which a waste collection worker would be exposed according to the scenario described in Appendix D is less than $1x10^{-4}$ mrem. The external dose to the entire waste collection population would be less than 10 mrem. Dose commitments from suspended thorium particulates or radon gas are not calculated, but are expected to be very low.

Disposal in Landfills

The radiological impact resulting from the disposal of used electrodes in landfills is assessed according to the methodology and assumptions described in Appendix D. Thorium is an element which occurs naturally in the earth's crust, and landfill disposal may be viewed by some as simply returning the material to the place whence it came. Furthermore, thorium clings tenaciously to soil particles resulting in a very slow migration rate. In the reference case, it would take about 4.8×10^5 years for the thorium to reach the groundwater withdrawal point. Thus, while it is probably true that a significant fraction of the thorium in used electrodes will eventually reach and be carried by the groundwater, the time which will have elapsed by the time any of that material is withdrawn in drinking water is so great, that any attempt at estimating future doses or health effects is not meaningful. Therefore, only a "worst case" assessment is performed here.

In addition to Th-232, a "worst case" assessment of Ra-228 is also warranted, since that radionuclide has a fairly long half-life (5.8 years) and is also present as a decay product of Th-232. It will be assumed here that the thorium and Ra-228 will be leached from the electrodes at the rate of one percent per year from previously incinerated sources, and 0.1 percent per year from intact sources. The total activity which would be ingested in the "worst case" scenario (described in Appendix D) is 1.8×10^{-3} Ci of Th-232 and its daughters, and 1.8×10^{-5} Ci of Ra-228 and its daughter products. The collective fifty-year organ dose commitments associated with these intakes are given in Table 6.4.

McDowell-Boyer evaluated the collective dose resulting from thoron (Rn-220) releases from used electrodes which have been disposed of in open dumps. These dose estimates, which are based on very conservative assumptions (e.g., all thoron produced in the rods for one year is released to the atmosphere), are for both the inhalation and the contaminated food crop ingestion pathways. The collective dose from inhalation of thoron and its daughters was estimated to be 13.5 person-rem to the lung and 0.27 rem to the whole body. The estimated doses from ingestion of contaminated food crops were 9.9×10^{-4} rem to the bone and 7.3×10^{-5} person-rem whole body.

Table 6.4. Fifty-year Organ Dose Commitments from Worst Case Scenario of Thoriated-Tungsten Welding Rod Landfill Disposal.

Organ	Collective Fifty-year Dose Commitment (Organ-rem)	
	Th-232(1)	Ra-228(2)
Total Body	2.2E+5	2.2E+3
Liver	4.0E+2	3.6E-1
Bone	2.0E+5	2.0E+3
Lungs	2.2E+5(3)	2.2E+3(3)
G. I. Tract	2.2E+3	2.0E+1
Gonads	2.2E+5(3)	2.2E+3(3)

(1) Dose commitments from ingestion of 1.8E+3 uCi of Th-232 in equilibrium with its daughters

(2) Dose commitments from ingestion of 1.8E+1 uCi of Ra-228 in equilibrium with its daughters

(3) Dose conversion factor (DCF) data unavailable; DCF assumed to be same as for total body

Incineration

For the assessment of the doses resulting from incineration of used electrodes, it is assumed that one-half of the initial thorium decay series activity is present in the used rods, and that one-half of this activity is released by the incineration process. The latter assumption may seem very conservative in light of the fact that the melting point of tungsten is 3410°C, while typical incinerator temperatures are controlled to between 1000-1300°C. Information cited in the McDowell-Boyer report, however, indicates that the tungsten may become oxidized and subsequently evaporated at the temperatures achieved during incineration (Perkins 61). It will be further assumed that all of the thoron present will be released from the rods, and that this thoron gas will not be affected by installed air pollution control equipment.

The results of the incineration dose assessment are presented in Table 6.5. The maximum individual dose commitments are estimated at 9.3×10^{-5} rem to the bone, and 3.9×10^{-6} rem total body from released thorium series radionuclides. The total population dose commitment from these particulate emissions is estimated at 1,000 rem to the bone, and 44 person-rem total body. With respect to thoron emissions, the resultant dose commitments are found to be very low: 1.4×10^{-10} rem to the lungs of the maximum individual and 1.8×10^{-3} rem to the lungs of the entire exposed population.

6.2.2.4 Accidents

Two cases of accidental situations or misusage involving thoriated-tungsten welding rods are evaluated here: (1) a fire in a warehouse containing 50,000 welding rods, and (2) the case of a worker carrying these products on his or her person. The warehouse fire conditions which are assumed are the same as those described in Section 2.2.2.4. The dose commitments resulting from this accident scenario are presented in Table 6.6. As can be seen from the results of this assessment, internal doses to firefighters can be high. However, it must be remembered that the assumed conditions are very conservative, and doses actually received in a warehouse fire would probably be much less.

McDowell-Boyer has assessed the case of a worker who has a habit of carrying extra electrodes in the pocket of a workshirt. Such a condition is unlikely to exist to a significant degree. Nevertheless, the dose to the skin of

Table 6.5. Fifty-year Dose Commitments Resulting from the Incineration of 5.2×10^5 Used Thoriated-Tungsten Welding Electrodes.

Exposure Category	Dose Commitment (Person-rem or Organ-rem)											
	Total Body		Liver		Bone		Lungs		G. I. Tract		Gonads(1)	
	I(2)	II(3)	I	II	I	II	I	II	I	II	I	II
Maximally Exposed Individual(4)	3.9E-6	4.7E-12	3.8E-6	1.5E-11	9.3E-5	6.4E-11	5.0E-5	1.4E-10	1.2E-8	3.1E-12	3.9E-6	4.7E-12
Average Individual(4)	2.0E-6	2.8E-12	1.9E-6	9.2E-12	4.7E-5	3.8E-11	2.5E-5	8.4E-11	5.8E-9	1.9E-12	2.0E-6	2.8E-12
Population - One Incinerator	1.5E-1	2.0E-7	1.4E-1	6.7E-7	3.4E+0	2.8E-6	1.8E+0	6.1E-6	4.2E-4	1.4E-7	1.5E-1	2.0E-7
Total Population	4.4E+1	6.1E-5	4.2E+1	2.0E-4	1.0E+3	8.3E-4	5.5E+2	1.8E-3	1.3E-1	4.2E-5	4.4E+1	6.1E-5

(1) Dose conversion factor (DCF) data unavailable; DCF assumed to be same as for total body

(2) Column I dose commitments are from Th-232 and daughters

(3) Column II dose commitments are from Rn-220 and daughters

(4) For the Rn-220 assessment, the maximum individual is assumed to reside at a distance from the stack of 300 meters; the average individual is assumed to reside at a distance of 1,000 meters

Table 6.6. Fifty-year Dose Commitments to Firefighters From a Warehouse Fire Involving 5.0×10^4 Thoriated-Tungsten Electrodes.

Organ	Dose Commitment(1) (Organ-rem)
Total Body	5.4E+0
Liver	5.2E+0
Bone	1.3E+2
Lungs	7.0E+1
G. I. Tract	1.6E+1
Gonads	5.4E+0(2)

(1) Dose commitment per firefighter involved from inhalation of thorium series radionuclides; external doses not calculated but would be very much lower than internal dose commitments

(2) Assumed to be same as for total body

a person carrying three electrodes in a chest pocket for eight hours per day, 250 days per year, is estimated to be about 3.5 mrem.

6.2.3 Non-Radiological Impact

The practice of welding can release toxic, noxious, or nuisance fumes or particulates, the composition and quantity of which depend on the alloy being welded and the process and type of electrodes which are used. In the process in which thoriated-tungsten electrodes are used, reactive metals and alloys ,such as aluminum and titanium,are arc-welded in a protective, inert atmosphere such as argon. These types of welding processes create relatively little fume, but the formation of ozone can result from the intense arc produced (ACGIH 79).

There are at least three occupational exposure standards relevant to tungsten arc-welding operations. These are the threshold limit values (TLV's) for airborne concentrations of ozone, tungsten, and welding fumes not otherwise classified (NOC). These values are listed in Table 6.7. According to the American Conference of Governmental Industrial Hygienists (ACGIH), most types of welding, even with little ventilation, do not result in exposures inside the welding helmet above 5 mg/cm^3. Those which do require design or operational controls to limit personnel exposures to acceptable levels (ACGIH 79).

There are, of course, other types of hazards associated with arc-welding operations. Electrical and thermal hazards, for example, must also be taken into account. These types of hazards, however, are not unique to thoriated-tungsten welding operations and are not evaluated here.

6.2.4 Summary of Impact

An upper level estimation of the total health risk, in terms of cancer deaths or serious genetic defects, associated with the use of thorium in tungsten welding electrodes is summarized in Table 6.8. Virtually all of the health risk associated with these products is from the inhalation of thorium released during the normal use phase of the products' life span. The conservative nature of the conditions assumed for the normal use dose assessment ensures that this health impact estimate truly represents an upper bound of the actual impact. It is not easy to assess the level of uncertainty associated with these estimates.

Table 6.7. Threshold Limit Values for Chemical Contaminants Relevant to Tungsten Arc-Welding Operations(1).

	Adopted Values TWA(3)		Tentative Values (2) STEL(4)	
Substance	ppm	mg/m^3	ppm	mg/m^3
Ozone	0.1	0.2	0.3	0.6
Tungsten and Tungsten Compounds(5)				
Soluble	-	1.0	-	3.0
Insoluble	-	5.0	-	10.0
Welding Fumes (NOC)(6)	-	5.0	-	5.0

(1) From ACGIH, 1979

(2) Values intended to be adopted in 1979

(3) TWA = time weighted average concentration for a normal 8-hour workday or 40-hour workweek

(4) STEL = short term exposure limit to which workers may be exposed for a period not to exceed 15 minutes

(5) Measured as tungsten

(6) Total particulates not otherwise classified

Table 6.8. Total Number of Estimated Health Effects from Distribution, Use, and Disposal of 5.2x10^6 Thoriated-Tungsten Electrodes(1).

Organ	Collective Dose Commitment (Organ-rem)	Cancer Deaths or Genetic Defects
Total Body	2.9E+3	2.0E-1
Liver	2.7E+3	4.1E-2
Bone(2)	6.6E+4	1.7E+0
Lungs	4.1E+4	1.0E+0
G. I. Tract	1.5E+1	2.3E-4
Gonads	2.9E+3	5.8E-1
Total		3.5E+0

(1) Does not include health effect estimates from accident or "worst case" landfill disposal scenario.

(2) Includes bone tumors and leukemia

However, the average thorium release rate used in the normal use dose assessments may be high by an order of magnitude.

Doses resulting from the disposal of thoriated welding rods are fairly low for the reference case scenarios. Doses from landfill disposal are deemed negligible for the reference case, although significant doses and associated health effects would result if all of the products were disposed of according to the "worst case" scenario postulated in Appendix D. Collective inhalation bone dose commitments from the incineration of welding electrodes can be as high as 1,000 rem distributed over the entire population with the average individual bone dose being about 0.1 mrem.

With respect to accidents, considerable doses could be received by firefighters combating a warehouse fire involving 50,000 welding rods. However, due to precautions normally practiced in firefighting operations (e.g., extensive ventilation of the building, the use of respiratory protection, the avoidance of smoke or fumes, etc.), the probability of these exposures occuring is low.

In summary, an upper level estimate of the total number of radiogenic health effects which would be expected to result from the population dose determined in this assessment is about three.

6.3 ALTERNATIVES

The principal alternatives to the use of thorium as an alloy in tungsten welding rods are the use of other materials or alloys, or the use of pure tungsten electrodes. Thoriated-tungsten alloys have been shown to offer important advantages over pure tungsten electrodes, as well as the currently used alloys of other types, for certain types of applications. It is not known whether or not other types of alloys which are competitive with thoriated-tungsten will become available in the near future.

Another type of alternative which should be considered is the use of smaller quantities of thorium in welding electrodes. Currently, the thorium content in these products varies from 0.35 to 2.2 percent ThO_2, depending on the type of application. This thorium content, however, can be expected to be governed by the performance characteristics of the product. Thus, unless some incentive exists for a reduction in the percent ThO_2 on a performance basis, one cannot realistically expect such a reduction to occur.

6.4 COST AND RISK VERUS BENEFIT

6.4.1 Cost Versus Benefit

The cost of thoriated-tungsten welding rods will vary considerably as a function of the size and composition of the product. There are two aspects related to the cost of incorporating thorium into welding electrodes which should be kept in mind. First, thorium is a relatively abundant element which occurs in many minerals. The thorium content of the earth's crust is estimated at between 0.001 and 0.002 percent (Cuthbert 58). This abundance, coupled with the fact that thorium extraction and production technology is fairly advanced, has kept the costs of industrial grade thorium at moderate levels. Second, the fact that thoriated-tungsten electrodes have proved to be valuable in "premium" industry applications demanding a high degree of excellence reduces the importance of cost as a factor.

The benefits received from the use of this product have been previously mentioned, and its demand is testimony to its effectiveness. It can be concluded, therefore, that thoriated-tungsten electrodes are cost-effective in the sense that the benefits they provide are important, and the realization of these benefits is not achieved at a large increase in cost.

6.4.2 Risk Versus Benefit

The radiological health risk associated with life span activities of thoriated-tungsten welding rods is low. When it is considered that the benefits gained are often in safety-related applications, it would seem that this low level of risk is certainly well within acceptable levels. At least some of the benefit obtained by using thorium pertains to the quality of the weld. Since the failure of a weld on a commercial aircraft or a nuclear power plant could have health consequences which would be far greater than those resulting from thorium use, the benefits in the case of these products are at least commensurate with the risk, and may be far in excess of that risk.

6.5 FUTURE USE

The application of thorium in arc welding processes has become a firmly established practice over the past 30 years. The important advantages which are unique to thoriated-tungsten electrodes will ensure that these products remain in demand unless a cost-effective alternative becomes available. Since no such alternative is likely to capture a large share of the tungsten electrode market at the expense of the thoriated products, these products can be expected to remain in widespread use in the near future.

6.6 SUMMARY

The use of thorium as an alloy material in tungsten welding electrodes is widespread, and is expected to continue to be so in the near future. The radiological health implications of this practice have been evaluated and, in light of the important benefits achieved through the use of this product, found to be low and within acceptable levels.

6.7 REFERENCES FOR SECTION 6

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7. OPHTHALMIC LENSES

7.1 PRODUCT DESCRIPTION AND BACKGROUND

7.1.1 General Discussion

For many eye glass requirements, a single lens in front of each eye is sufficient to accomplish the corrective needs of the wearer. As the human eye ages its shape can undergo specific changes that result in the need of bifocals (or trifocals) in order that the wearer can clearly view things at a distance as well as closeup. This requirement of the eyes is satisfied by constructing glass lenses of two or more segments. In this case the segment that is fused to the ordinary lens is made of a glass of different density. The density of the glass is precisely controlled so as to give the fused segment a refracting index that is different from that of the ordinary lens.

7.1.2 Product Description

The materials employed to adjust the density and refractive index of ophthalmic lenses, as well as the tint of these lenses, are numerous. Barium oxides and zinc oxides are frequently employed in ophthalmic lenses as well as a host of 15 to 20 other oxides of various sorts. Addition of the oxides to the lens material overcomes clarity problems, increases chemical durability and combats reception diversion (a rainbow effect that can occur in some lens arrangements).

The base material of ophthalmic glass is silica. Silica can come in an infinite variety of grades, some of which have much higher contaminant factors than others. Silica also comes in various grain sizes. The grain size requirement for the basic silica is related to rates of melt needed in certain lens development.

Thorium and uranium are natural contaminants of silica and zinc. Because of these natural contaminants there is a small quantity of naturally occurring radioactivity associated with glass items. There does not exist any single source of data that clearly indicates which if any of the currently available ophthalmic glass lenses contain radioactivity above what might be considered a background amount. Manufacturers have indicated that no thorium, uranium, or any other radioactive material is intentionally added to ophthalmic glass. In 1974, a series of reports were developed for the U. S. Atomic Energy Commission (USAEC) on the subject of amounts of radioactive materials in ophthalmic glass (USAEC 74, Tobias 74, Casarett 74). The manufacturers of ophthalmic glass were made aware of the intended study and they supplied the agency study group with representative samples of their products. During the course of the current study, it was disclosed by one manufacturer that the glass lens sample supplied to the agency group was chosen by the manufacturers so as to represent the samples most likely to contain any excess radioactivity. These studies constitute the single most definitive informational source on radioactivity in ophthalmic glass, and represent a major source for the current study.

The USAEC study came about following the publicity of a few papers on the subject of potential radioactive contaminants in optical and ophthalmic glass (see especially: Notea 72, McMillan 73, Pecora 74). As a result of the USAEC study, the Optical Manufacturers Association published in November 1975, an Ophthalmic Glass Radiological Standard, the purpose of which is to establish a uniform maximum level for radiological emissions from ophthalmic glass. The limits set by this standard are as follows:

Alpha particles - less than 0.45 disintegrations/cm^2/min

For specified isotopes:

Actinium-228	30 dpm/g
Lead-212	30 dpm/g
Lead-214	30 dpm/g

By measuring actinium-228 or lead-212 activity, radionuclides within the thorium chain could be accounted for. By measuring lead-214, the radionuclides within the uranium chain could be accounted for. When the following statement accompanies ophthalmic glass, it attests to the fact that the glass meets the above established standards:

> "This ophthalmic glass conforms to all requirements of the Optical Manufactures Association Ophthalmic Glass Radiological Standard."

7.1.3 Distribution

The distribution of ophthalmic glass lenses is based upon information derived from the U. S. Census Bureau, spokespersons for various sectors of the ophthalmic lens industry, and spokespersons from various professional organizations centered around the ophthalmic lens industry. This study, as will be explained, is primarily concerned with glass lenses as opposed to plastic. The influence of the plastic lens in the market place is clearly represented in Table 7.1.

Although the glass lens market has increased annually by one to five percent, the plastic lens market has been increasing at a faster rate. At present, the ophthalmic lens market is split in the U. S. at 55 percent glass and 45 percent plastic (excluding an approximate ten percent of the total market occupied by contact lenses). It currently appears that the plastic lens will occupy approximately 50 percent of the ophthalmic lens market within the next two years. Whether this number will be reached or surpassed depends on a number of factors. Plastic lenses received a big boost in sales when larger lens sizes became fashionable. The plastic lens became desirable in this instance due to its lighter weight. This fashion trend is apparently on the wane at this time. A photochromic plastic lens has been developed, but is not as yet marketed in the U. S., and its photochromic quality has not proven as durable as that of glass lenses. Government regulations could give the plastic lens a boost if more stringent shatter resistance is required. The plastic lens does not break as easily as glass, but it does have much less scratch resistance. The scratch resistance problem with plastic has been remedied to some degree by coating the outer surface of the lens with a thin layer of quartz glass or by developing harder organic materials that can be added to the basic plastic material. Most factors designed to improve the quality of plastic lenses are frequently negative

Table 7.1. Annual Distribution of Ophthalmic Lenses (Millions).

Lens Material	1967	1972	1977	1979
Glass	26.1	27.4	29.9	30.3
Plastic	10.6	12.3	18.7	24.7
Total	36.7	39.7	48.6	55.0

influences in the market place as they tend to increase the cost of the plastic, which is already sold to the consumer at a higher price than equivalent glass. It is, thus, not possible at this time to discern the future market place for ophthalmic glass lenses beyond a possible 50 percent of the industry.

The significance of foreign imports is quite varied depending on which segment of the ophthalmic lens market is considered. Manufacturers have indicated that for prescription ophthalmic lenses, the domestically manufactured products occupy greater than 95 percent of the market. For non-prescription type sunglasses, the inexpensive drug counter variety, it is doubtful that more than 30 percent of these items are manufactured in the U. S. Of the remaining sunglasses market, 45 percent are manufactured in the Orient and 25 percent in Europe. The non-prescription sunglasses market is of considerable size in the U. S., representing approximately 90 million pairs of glasses per year.

7.2 ENVIRONMENTAL IMPACT

7.2.1 Benefits

The contaminants of opthalmic glass that have been considered are natural materials that are present in the glass and in some of the chemicals used to control quality of lenses being manufactured. Therefore, the benefits gained are not due to the purposeful addition of radioactive materials to ophthalmic glass. The benefit of ophthalmic glass is self-evident. To purify this lens material to such a state that no impure (radioactive) contaminants are present would increase the cost of eyeglasses to a level beyond the reach of most consumers. Beyond these considerations, no further benefits can be considered regarding the radioactive contaminants of ophthalmic glass.

7.2.2 Radiological Impact

The radiological impact asociated with ophthalmic glass is related primarily to the thorium and uranium contaminants of the basic glass materials. Tables 7.3 and 7.4 of Section 7.2.2.3 are abbreviated tables of constants for the daughter nuclides associated with thorium and uranium.

For the wearer of opthalmic glass lenses, the primary concern is the alpha-rays associated with the thorium and uranium daughter nuclides. As will be explained in Section 7.2.2.3, the primary concern for the consumer is the alpha-rays with energy levels above 6.5 MeV.

For the general population associated with the ophthalmic glass industry, the primary concern is the level of gamma radiation present in areas where large numbers of ophthalmic lenses might be stored.

For the disposal of opthalmic glass lenses in landfills, the primary concern is groundwater transport and any associated human impact. Since the material being discussed is glass and its radiological contaminant, its disassociation in landfills presents some specific considerations which will be addressed. It must be remembered, however, that we are dealing with naturally-occurring radioactive materials, and landfill disposal represents a case in which these materials are returned to the environment from which they were obtained.

7.2.2.1 Occupational Exposure

The ophthalmic glass industry is categorized into four segments: glass blank manufacture, optical lens manufacturer, wholesale optical labs, and optical glass dispensers. The U. S. Bureau of Census (USBOC 77) carries data on the ophthalmic glass industry as Industrial Code 285. Under this listing, for 1977, the number of production workers is indicated to be 21,700 (USBOC 80). These production workers represent those employed in the glass blank industry and the optical lens manufacturing industry, but do not include those employed in wholesale optical labs (finished prescription products) or dispensers. Additional estimates as to the number of workers in various segments of the ophthalmic glass industry (see Table 7.2) were developed from information supplied by representative companies and industrial trade organizations.

For the entire ophthalmic glass industry the total number of workers that handle, or are near ophthalmic lenses, is estimated to be 114,000. There is a wide range of numbers of employees in some categories. This is especially true for wholesale optical labs which can range from five or so employees to greater than 1000. Coinciding with the wide range of employees in specific categories is a wide range of the number of lenses stored at any one time. For the wholesale optical lab category the number of lenses stored can range from an estimated 300

Table 7.2. Distributions Within the Ophthalmic Glass Lens Industry

	Glass Blank Manufacturers	Optical Lens Manufacturers	Wholesale Optical Labs	Dispensers(1)	Customers
Number of Sites	3.0E+0	3.0E+1	1.6E+3	2.0E+4	3.0E+7
Total Number of Production Workers	2.0E+3	2.0E+4	3.2E+4	6.0E+4	—
Number of Lens Stored any One Time(2)	1.0E+6	1.5E+5	3.0E+3	2.0E+2	4.0E+0

(1) Dispensers do not include those members of the profession who prescribe but do not dispense

(2) Based on storage of three weeks supply

pair to greater than 50,000 pair. These wide swings in numbers make any estimations based on average numbers questionable when associated with the extremes.

Occupational exposure to ophthalmic glass is probably highest, in terms of individual exposure, for members of the glass blank manufacturing category. Individuals in the optical lens manufacturing category probably receive lower individual doses than glass blank workers, but the collective dose is roughly equal. Workers in wholesale optical laboratories and optical glass dispensors would receive individual and collective doses which are substantially lower than the former categories. Due to the complexities of the exposure scenarios (e.g., wide variations in shielding, geometries, exposure times, etc.), it is not possible to quantitatively determine the magnitude of these exposures with an acceptable degree of accuracy.

7.2.2.2 Exposure During Distribution

Individuals in this group include workers in the wholesale optical laboratories and dispensers. Of these two groups, the largest individual dose is probably to the worker in the wholesale optical laboratories group. Doses to members of the transportation population (e.g., parcel service employees, terminal workers, etc.) are not assessed, but they are clearly low.

7.2.2.3 Exposure to User

Two exposure evaluations are required for consumers use of ophthalmic glass. The first consideration is to the tissue of the eyes as they are exposed to alpha- and beta-rays from the thorium or uranium daughters. The second consideration is that of the whole body or organ doses that results from the gamma-rays as a result of the thorium or uranium series radionuclides in the ophthalmic glass.

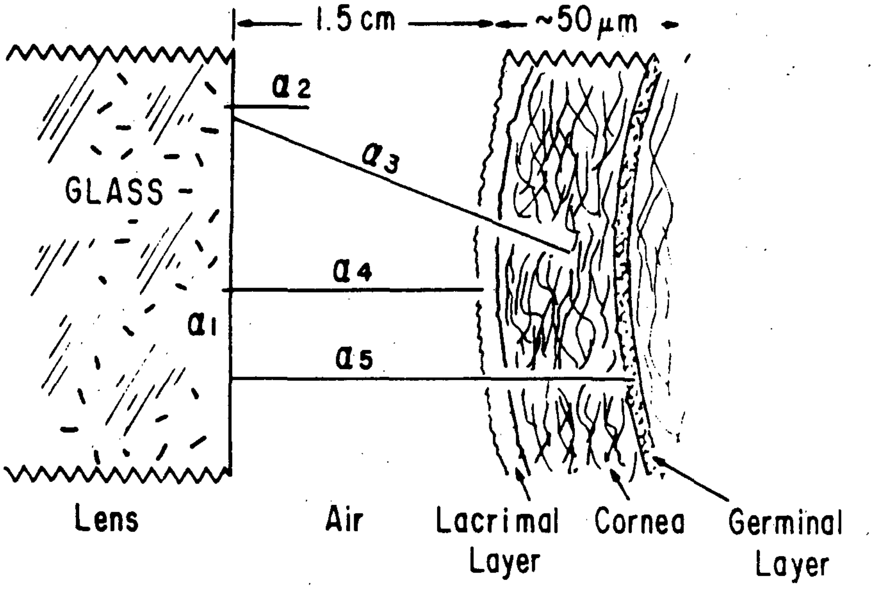
For consideration of the alpha- and beta-ray exposures that might be associated with ophthalmic glass lenses, the data from the 1974 USAEC studies are the best available sources. A summary report of these studies, by Goldman and Yaniv (Goldman 78), will be used as the source for the following analysis.

Based on the model of Goldman and Yaniv, the corneal germinal cell layer resides some 50 um beneath the lacrimal layer coating the outer surface of the eye (see Figure 7.1). The alpha particles that emanate from the back surface of glass ophthalmic lenses must transverse an air gap of 1.5 cm prior to reaching the lacrimal layer. The 1.5 cm air gap between the ophthalmic lens and lacrimal layer, along with the lacrimal layer and the 50 um depth of tissue before reaching the corneal germinal layer, require that alpha particles must have a minimum of 6.5 MeV to reach the critical germinal tissue. Tables 7.3 and 7.4 indicate the decay series for thorium and uranium and their associated emission energies. For the thorium series both Po-212 (8.78 MeV) and Po-216 (6.91 MeV) have sufficient energies to reach the critical tissue. For the uranium series, only Po-214 (7.84 MeV) has sufficient energy to reach this critical tissue layer.

Since the legal limit of radioactive material in a consumer product is 0.05 percent, this will be the expected "worst case" exposure considered. The fact that lens manufacturers are striving to meet the standards of the Optical Lens Manufacturers Association recommendations by limiting the sum of thorium and uranium activity to not more than 30 dpm/g also indicates that 0.05 percent is an upper limit.

For a consumer wearing ophthalmic glass lenses for 16 hours per day, the cumulative annual use would approximate 6000 hours per year. Based on this exposure time and a maximum thorium content of 0.05 percent, the exposure dose to the corneal germinal layer, as determined by the USAEC study group, would be 0.2 rads per year. Assuming a quality factor of 20 for alpha-rays, the dose to the corneal germinal layer, at 50 um tissue depth, would be four rem per year.

For whole body exposure to the consumer as a result of wearing ophthalmic glasses containing thorium, the gamma-ray exposure will be calculated based on the model of Appendix A. The glass lens for these calculations is estimated to weigh 30 grams and contains 0.05 percent by weight thorium. This weight of thorium (0.015 g) is equivalent to 1.7×10^{-9} Ci/lens. See Table 1.1. For an annual exposure of 6000 hours the whole body dose, based on an average unshielded exposure distance of one meter, is 1.8×10^{-5} rem per year. Two lenses are assumed per pair of glasses. The 1977 population of persons wearing eyeglasses (USBOC 79)was 96.3×10^{6} persons. Using this number and estimating the number of glass lenses based on 55 percent (5.3×10^{7}) the population dose to consumers is 950 person-rem per year.



Corneal Exposure Model Indicates that Alpha Particles Originating in Glass (α_1, α_2, α_4) or from Its Surface (α_3, α_5) Must Traverse 1.5 cm of Air and 50 um of Tissue Before Reaching the Critical Corneal Germinal Cell Layer. Alpha Energies Above 6.5 MeV are Required (e.g., α_5). (Goldman and Yaniv 78).

Figure 7.1. Corneal Exposure Model.

Table 7.3. Thorium Series - Abbreviated Table of Constants(1).

	Radiation	Half-life	Decay Energy(2) (MeV)	Beta Energy (MeV)	Gamma Energies(3) (MeV)
Th-232	α γ	1.41 x 10^{10} y	4.08	--	--
Ra-228	β γ	5.77 y	0.055	0.024	0.01
				0.048	0.026
Ac-228	β γ	6.13 h	2.14	0.45	0.057
				2.10	0.338
Th-228	α γ	1.913 y	5.52	--	0.084
					0.234
Ra-224	α γ	3.64 d	5.79	--	0.240
					0.650
Rn-220	α γ	55 s	6.41	--	0.542
Po-216	α	0.16 s	6.91	--	--
Pb-212	β γ	10.6 h	0.58	0.17	0.115
					0.415
Bi-212	α β γ	60.6 m	β 2.25	0.08	0.039
			α 6.21	2.27	1.809
Po-212	α	3.04 x 10^{-7} s	8.78	--	--
Ti-208	β γ	3.1 m	4.99	2.38	--
Pb-208	stable	stable	--	--	--

(1) From Lederer, et al. (1967), CRC (1973), Goldman and Yaniv (1978).

(2) Alpha-particle energies are somewhat below the decay energy.

(3) Among several energy levels, the lowest and highest are shown.

Table 7.4. Uranium Series - Abbreviated Table of Constants(1).

	Radiation	Half-Life	Alpha Decay Energy (MeV)	Beta Decay Energy (2) (MeV)	Gamma Decay Energy (2) (MeV)
U238	α	4.51×10^9 γ	4.268	---	0.048
Th-234	β	24.1 d	0.263	0.191 0.100	0.029 0.093
Pa-234	β^- IT.	1.17 m 6.75 h	2.23	0.23 1.02	0.043 0.153
U-234	α	2.5×10^5 y	4.856	---	0.053 0.580
Th-230	α	8×10^4y	4.77	---	0.068 0.206
Ra-226	α	1600 y	4.97	---	0.186 0.610
Rn-222	α	3.82 d	5.59	---	0.510
Po-218	α β	3.05 m	α 6.111 β 0.28	---	---
Pb-214	β^-	27 m	1.04	0.59 1.04	0.053 0.777
Bi-214	α β	19.7 m	α 5.62 β 3.28	0.082	0.063 0.806
Po-214	α	1.64×10^{-4} s	7.84	---	0.800
Ti-210	β^-	1.3 m	5.5	1.3 2.3	0.097 2.4
Pb-216	α β^-	21 y	α 3.72 β 0.061	0.015	0.046
Bi-210	β^- α	5 d 3×10^6 y	β 1.16 α 5.04	---	---
Pb-206	Stable	Stable	---	---	---

(1) From Lederer, et al. (1967), CRC (1973), Goldman and Yaniv (1978).

(2) Among several energy levels, the lowest and highest are shown.

7.2.2.4 Product Disposal

Disposal of prescription lenses is generally by way of the local waste disposal site. There are some philanthropic organizations that try to distribute old discarded prescription lenses, but the glasses they can use are generally only the magnifying type used for reading. Prescription glasses are, in a sense, a fingerprint of their owner; each is ground to fit the unique optical requirements of each individual and for this reason few, if any, can be worn again by another person. Since most ophthalmic glasses are made with hardened materials, they cannot generally be reground. Because of this, more than 99 percent of used ophthalmic lenses are eventually disposed of in the local waste disposal system.

Collection

There are an estimated 175,000 workers and 138,000 vehicles involved in solid waste collection (see Appendix D). Based on a disposal rate of 60 million lenses per year there would be an average of one pair of glasses per vehicle collected daily. This would amount to a daily exposure of about three hours and results in a whole body dose value much below that of the ophthalmic glass user (1.8×10^{-5} rem/year). Because of this, a detailed exposure assessment will not be performed for this oocupation.

Landfills

The leach rate for ophthalmic glass in unknown in regards to the minute amount of background thorium found as contaminant in these lenses. The impact of disposal in landfills can be estimated utilizing data from Appendix D. If equilibrium is assumed for ophthalmic glass distribution, 60 million individual lenses are being disposed of per year. This would amount to approximately 3200 lenses per waste site per year, assuming 18,500 disposal sites. For an individual lens the estimated activity is 1.7×10^{-9} Ci. This means that the individual disposal site is receiving approximately 5.4×10^{-6} Ci per year from the disposal of prescription glass lenses.

For the reference site, the time of first appearance of thorium at the point of water withdrawal is 480,000 years. This is too long a time to be meaningful in terms of dose calculations and, therefore, these calculations will not be included.

For a worst case situation, the collective fifty-year dose commitments are calculated as outlined in Appendix D. The dose commitments from ingestion are listed in Table 7.5.

Incineration

Sections 5 and 6 assess the individual population doses resulting from the incineration of other products containing similar amounts of thorium. These doses are found to be extremely low, and the same would be the case with respect to ophthalmic glass.

7.2.2.5 Accidents

Accidents related to the thorium content of ophthalmic lenses either for inhalation or ingestion cannot be considered significant simply due to the nature of the glass. The maximum exposure situation considered is that of the glass blank manufacturer where one million lenses may be stored at one time. Based on the methodology assumed in Section 5.2.2.5, a fifty-year dose commitment to a firefighter that inhaled thorium from a ten percent release of activity from the lenses would be 32 rem to the lungs. Since it is unlikely that circumstances could occur such that the firefighters could receive this dose, the further development of dose commitments will not be conducted.

7.2.3 Non-Radiological Impact

Non-radiological impact would appear to be of little, if any, significance in relation to the manufacture, distribution, use, or disposal of ophthalmic glass. Some of the oxides used in the manufacture of the lens blanks may be of consequence to personnel in that industry, but levels of use are not known (not made public by manufacturers).

Three chemicals that are used in glass lens manufacture are barium oxide, zinc oxide, and silicon. Many other oxides have been reported to be used but only these three most popular elements will be considered. A benign form of pneumoconiosis is associated with the inhalation of barium sulfate and silicosis

Table 7.5. Fifty-year Organ Dose Commitments from Worst Case Scenario of Landfill Disposal of 60 Million Thoriated - Glass Lenses.

	Collective Fifty-year Dose Commitment (Organ-rem)	
Organ	Th-232(1)	Ra-228(2)
Total Body	3.1E+5	3.1E+3
Liver	5.7E+2	5.2E-1
Bone	3.1E+5	2.9E+3
Lungs	3.1E+5(3)	3.1E+3(3)
G.I. Tract	3.2E+2	3.1E+1
Gonads	3.1E+5(3)	3.1E+3(3)

(1) Dose commitments from ingestion of 2.6E+3 uCi of Th-232 in equilibrium with its daughters

(2) Dose commitments from ingestion of 2.6E+1 uCi of Ra-228 in equilibium with its daughters

(3) Dose conversion factor (DCF) data unavailable; DCF assumed to be same as for total body

is a well-documented pneumoconiosis associated with silicon. No respiratory problems have been associated with the inhalation of zinc oxide, although a local granuloma has been reported associated with its topical application (USDHEW 77). All of the precautions associated with these elements in terms of their occupational hazard are related to doses much higher than expected in the glass manufacturing process. As such, no non-radiological hazard associated with glass lens manufacturing is expected. As a result the non-radiological impact of the manufacturing, distributing, using, and discarding of this material is considered minimal.

7.2.4 Summary of Impact

The environmental impact due to the disposal of glass ophthalmic lenses containing thorium has not been found to constitute a significant hazard. The values calculated for all consequences are very conservative to allow for the worst possible situations. The highest doses calculated are the fifty-year dose commitments due to all lenses being disposed of in a worst case scenario, a situation which is totally unrealistic. The total population dose resulting from the wearing of ophthalmic glass lenses is about 950 person-rem per year.

7.3 ALTERNATIVES

7.3.1 Radiological Alternatives

For ophthalmic glass, no alternatives exist in relation to the thorium or uranium content. These elements are included in the glass only as impurities in the raw material that goes into the development of the lens blanks. Since these materials are not included for purposes of lens quality, there is no need to consider any other radionuclides as their replacements.

7.3.2 Non-Radiological Alternatives

For ophthalmic glass, the only non-radiological alternative is to go to hyper-pure materials. This would drive the cost of the lens above that which can be afforded by consumers, and this industry would stand the chance of complete replacement by the plastic lens industry. At this point, it would appear that other areas of glass use by the consumer would have to be considered. This would include ordinary window glass as well as auto, industrial, and a host of other applications.

Plastic lenses are a non-radiological alternative and have currently gained at least 45 percent of the market that could be occupied by glass lenses. Plastic contact lenses are another alternative, but the costs of these consumer items are high and, thus, unavailable to a large segment of the consumer market.

7.4 COST AND RISK VERSUS BENEFIT

7.4.1 Cost

The cost to the consumer for a pair of glasses depends upon characteristics of the lens, the frame, the establishment supplying the service, and whether or not the customer has some kind of insurance that covers all or some of these costs. The lenses themselves can have the options of being either treated, hardened, photochromic, single lens, bifocal, or trifocal. The individual requirements of each consumer's specific needs also put demands on what is required in surfacing the lens to fit the prescription.

Due to the range of options available to the consumer, the cost for a pair of lenses can have considerable range. For a "standard" pair of lenses, one with no frills other than a simple prescription, the average cost is approximately $30.00. Adding tints or other treatments can increase the cost of a basic pair of lenses by $6.00 to $50.00. These costs are for the lenses only and do not include the frames, which can again increase cost considerably and which also have a wide range of prices.

7.4.2 Product Analysis

The various grades of silicon that are available, as well as different grain sizes, play an important part in the manufacture of ophthalmic glass lenses. These characteristics lead to desired optical differences in the final ophthalmic lens product. By carefully controlling the additives to the silica, as well as the silica base material, a very large array of ophthalmic lenses are produced.

The ophthalmic glass supplied for the 1974 USAEC Study (USAEC 74, Tobias 74, Casarett 74) consisted of 441 samples. Based on similar chemical compositions, the glass samples were segregated into 28 groups. The number of different types of glass per group varied from one to ten. Table 7.6 is from the 1974 USAEC study and lists the 28 groups of lenses. From Table 7.7 it appears that all lens material, including the clear crown base material (Group one) displays some radioactive disintegrations. The radioactivity being measured in the study was that of gamma emissions originating from daughter nuclides of either the thorium series (Pb-212) or uranium series (Pb-214). Measurements of gamma emissions were employed in these studies as a means of determining the thorium or uranium content of the glass. The validity of this approach is based on the assumption that the short lived radionuclides of these series are in secular equilibrium with the parent material.

The questions of "background" activity for uncontaminated ophthalmic glass was an unknown when the USAEC study began. By a succession of evaluations it was decided by the study group that background would be disintegration rates below 20 dpm/g. Of the 441 glass samples, 351 (80 percent) met this criterion. Nine groups of the 28 listed in Table 7.6 contained samples in excess of 20 dpm/g. Of the 90 samples in these nine groups, seven exhibited activity in excess of 50 dpm/g. Table 7.7 lists the total number of lens samples in each group and the percent of "contaminated" lenses in each group.

The highest count rates for contaminated lenses occurred in group 19 of Table 7.6. Here, within a single type of glass, the count rate varied among four samples from 7.78 to 350 dpm/g. This apparently demonstrates, as explained by manufacturers, problems that can occur due to differences in batch materials, as well as a need for more stringent controls.

7.4.3 Alternatives Analysis

The consumer that requires prescription glasses to correct faulty vision has a choice of three items: glass lenses, plastic lenses, or contact lenses. Due to cost, physical discomfort, and/or possibly medical reasons, many consumers cannot consider contact lenses as an alternative form of corrective

Table 7.6. Results of Gamma Measurements, Ophthalmic Glass Samples(1).

	Lead-212 Disintegration per minute per gram		Lead-214 Disintegration per minute per gram		
Group No.	Mean*	Standard** Deviation	Mean*	Standard** Deviation	Glass Description
1	1.84	1.28	3.71	2.00	Clear crown
2	3.27	1.28	13.15	2.28	High index segments for fused bifocals, contains 3-6% ZrO_2
2	2.55	0.97	5.46	1.62	High index segments for fused bifocals, contains ∿1% ZrO_2
4	3.12	1.57	2.87	1.92	Crown glasses - different tints
5	7.37	1.66	24.15	2.78	Barium segment - 5-7% ZrO_2
6	2.16	1.19	5.13	1.84	Flint segments; high lead content - 0-2% ZrO_2
7	2.19	1.24	4.40	1.85	Clear crown
8	15.39	3.11	6.34	2.27	Pink crowns - 2-3% rare earth oxides
9	18.82	2.35	9.90	2.42	Crookes crown - ∿6% rare earth oxides
10	2.11	1.31	1.78	1.89	Green crown - no rare earth oxides
11	1.93	1.26	2.45	1.83	Neutral crown
12	2.18	1.28	3.01	1.89	Tan crown
13	1.73	1.34	3.40	2.08	Blue crown
14	2.24	1.25	3.75	1.96	Yellow crown
15	2.08	1.28	2.81	2.00	Special glass for strengthening
16	10.00	1.80	27.48	2.80	Barium segments ∿7% ZrO_2
17	1.62	1.32	2.52	1.95	Clear crown
18	23.17	10.81	6.70	2.88	Tinted glasses - 1 to 10% rare earth oxides
19	64.56	23.38	16.37	7.14	Crooks glasses <10% rare earth oxides
20	1.65	0.88	3.02	1.30	Tinted glasses - no rare earth oxides
21	1.93	0.99	2.73	1.46	Tinted glasses - no rare earth oxides
22	2.60	1.15	4.14	1.91	Tinted glasses - no rare earth oxides
23	4.55	1.51	8.69	2.37	Photocromic glass - 1-10% ZrO_2
24	2.35	1.48	3.26	1.97	Welding and industrial glasses
25	2.72	0.89	8.28	1.67	Flint segments - 1-10% ZrO_2
26	1.29	0.73	2.09	1.15	Flint segments no ZrO_2
27	5.79	1.27	18.60	3.01	Clear barium segments - 1-10% ZrO_2
28	7.04	1.16	22.44	2.61	Tinted barium segments - 1-10% ZrO_2

*Background included.
The background pooled from 6 individual determinations is: for ^{212}Pb — 1.32 ± 1.30 dpm/g
for ^{214}Pb — 1.45 ± 1.85 dpm/g

**Standard Deviation includes both measurement error and the within group variability.

(1) From USAEC, 1974.

Table 7.7. Percent (P) of "Contaminated" Lens Pressings in a Group and the Upper 80 Percent Confidence Limit (CL) for the Percent.(1)

Group No.	Total Lens in Group	P	C.L.
1	15	0.00	1.18
2	22	13.64	20.35
3	16	0.00	1.10
4	10	0.00	1.76
5	20	90.00	94.91
6	16	0.00	1.10
7	15	0.00	1.18
8	16	37.50	47.89
9	16	68.75	78.02
10	16	0.00	1.10
11	18	0.00	0.98
12	16	0.00	1.10
13	18	0.00	0.98
14	15	0.00	1.18
15	12	0.00	1.47
16	15	100.00	100.00
17	16	0.00	1.10
18	16	25.00	34.60
19	16	75.00	83.50
20	15	0.00	1.18
21	15	0.00	1.18
22	11	0.00	1.60
23	15	0.00	1.18
24	16	0.00	1.18
25	16	0.00	1.10
26	15	0.00	1.18
27	16	50.00	60.45
28	18	72.22	80.61

---

(1) From USAEC, 1974.

lenses. Because of this, for the general consumer, plastic lenses on conventional frames will be considered to be the only alternative available.

The additional cost to the consumer for a pair of plastic lenses as opposed to glass lenses can range from $5.00 to $20.00. For this added expense the customer has the convenience of lenses that weigh approximately half that of equivalent glass lenses. This is especially important when large or thick lenses are desired or required.

Plastic lenses that have a photochromic quality are only available in Europe at the present time. As the photochromic quality is considerd a desirable feature (for some consumers) of glass lenses, the plastic lens industry has been actively seeking a similar feature. The ability of the plastic lens to change color density on exposure to specific photons of light is not as persistent as in glass. Apparently, the chemicals used in the plastic lenses fatigue with age resulting in a longer time required for the color change to occur.

Plastic lenses give the customers a wide range of color choices and are less expensive to stock by the wholesale optical lab since the colors are applied to the lens by dipping in a dye solution. The dye in this case is an organic substance and can be bleached from the plastic if the color is not correct.

There are basically three types of plastics used in optical systems: polycarbonate, methyl methacrylate (acrylic), and a monomer sold under the trade name CR-39. None of these materials can come near glass in its ability to resist scratching. Part of this problem has been overcome by adding harder organic materials to the basic plastic material. A second development has been to fuse a thin layer of quartz glass to the plastic lenses.

Plastic lenses do not have the scratch resistance of glass lenses but they do surpass glass in terms of impact resistance. The impact resistance factor for plastic lenses is especially true when considering sharp pointed objects coming into contact with the lens. As the Federal Government continues to regulate the safety factor of consumer items, the impact resistance factor for plastic lenses may prove to be a marketing edge of greater proportion than it already is for this product.

The risk to the consumer associated with the choice of plastic lenses is primarily that of cost. This product is initially sold to the consumer at a higher cost than glass lenses and, due to its inherent lower scratch resistance, it may need to be replaced sooner than a glass lens.

The result of the above analysis is that plastic lenses, as an alternative to the consumer, are more expensive and are, thus, not expected to aquire more than approximately 50 percent of the prescription lens market.

7.5 FUTURE USE

Ophthalmic glass is expected to continue to be a major material in the manufacture of prescription glasses. The analysis of the market position of plastic lenses indicated that glass will probably keep a 50 percent share of this market in the foreseeable future. For the total number of lenses marketed, the only major decline that could occur is considered to be associated with contact lenses. For contact lenses to cause a major change in the future use of glass lenses in prescription glasses, the product will need to be reduced considerably in cost to the consumer. Therefore, the future market for ophthalmic glass lenses is expected to remain around 30 million pairs per year.

7.6 SUMMARY

The materials from which glass ophthalmic lenses are manufactured contain a minute quantity of thorium and/or uranium. These radionuclides are present as contaminants and to remove them would make glass opthalmic lenses beyond the reach of the consumer in regards to cost.

The radioactivity in glass ophthalmic lenses emits alpha particles which potentially could be damaging to critical tissues. For lenses with maximum permissible concentration levels, the dose to the corneal germinal layer was calculated to be four rem per year.

No definitive calculations were made for the exposure received by workers in the ophthalmic glass industry. The geometries associated with their exposure to the stored lenses made these calculations too unreliable at this time.

There is currently an annual average of 3200 glass ophthalmic lenses being disposed of per waste site. This amounts to 5.4×10^{-6} Ci per year per disposal sets. Leach rates from this glass are unknown, but for a reference case, the first appearance of thorium at a point of water withdrawal due to this disposal would be 480,000 years from now.

7.7 REFERENCES FOR SECTION 7

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8. GLASSWARE AND CERAMIC TABLEWARE

8.1 PRODUCT DESCRIPTION AND BACKGROUND

Uranium compounds have been used for many decades to produce pigmented glazes for glassware and ceramic pottery and tableware. Examples of products containing uranium compounds are platters, pitchers, tumblers, vases, mugs, and bowls.

Sodium uranyl carbonate was commonly used to produce fluorescent and irridescent glass in the past. Uranium in the form of oxides and as sodium uranite was used to produce ceramic glazes in the past, with concentrations ranging from one to 20 percent by weight. Today, no manufacturer is known to use uranium as a glaze for dinnerware, although at least one company is still involved in the manufacture and distribution of decorative glassware containing depleted U_3O_8. Another firm was involved in the distribution of drinking glasses containing a uranium coloring agent as recently as 1972.

Several authors have evaluated the public health aspects of uranium in glassware and tableware (Menczer 65, BRH 72, NCRP 77, Simpson 78). These studies have been plagued by difficulties in estimating the relative quantities of uranium items distributed in the past. It is the intent of this section to briefly summarize the radiological health impact associated with the distribution of these products.

8.2 DISTRIBUTION

As mentioned above, a scarcity of information exists with respect to uranium glassware or ceramics. Table 8.1 contains an estimation of the past distribution of specific products. These estimates are by no means complete; rather, they represent estimates made by a limited number of manufacturers concerning past and, in the case of decorative glassware, current production.

Table 8.1. Estimated Distribution of Glassware and Ceramic Dinnerware Containing Uranium(1)

Items	Years	Number of Pieces	Uranium Content (lb)	
			Per Item	Total
Dinnerware	1959-69	2,000,000	0.01	22,000
Decorative Glassware	1958	10,000	0.002	20
	1959-68	170,000	0.002	340
	1969-72	1,300,000	0.002	2,600
	1973-77	2,200,000	0.002	4,400
	1978	480,000	0.002	950
Drinking Glasses	1968-72	15,000	Unknown	Unknown

(1) Incomplete data; numbers reflect estimates by some of the major distributors over the period indicated

8.3 RADIOLOGICAL IMPACT

Menczer (1965) has performed numerous measurements of the beta and gamma radiation levels from a variety of glazed ceramic items, most of which were believed to have been manufactured prior to the 1940's (Table 8.2). Using the measurement results and assuming an exposure time of 1.5 hours per day, Menczer estimated that radiation exposure to the hands ranged from 2 to 10 rem per year. The 1.5 hour daily exposure period would probably produce an upper level exposure estimate, since under conditions of normal use the duration of exposure would be shorter.

A radiological hazards evaluation performed on pottery samples by the Bureau of Radiological Health has indicated the following gamma exposure rates from a variety of ceramic products (BRH 72):

Pottery - 5 mR/hr at 0.5 inches
Plates - 3.5 to 7.8 mR/hr at face of plate
5.0 mR/hr at three inches from surface
8.0 mR/hr at contact
3.0 mR/hr at one inch
15 mR/hr at contact
0.5 to 1 mR/hr at surface

The BRH evaluation concluded that:

(1) The use of uranium in pottery or tableware should be classified as a product of small or no value to the health or well-being of the public.

(2) Suitable substitutes appear to be available.

(3) Radiation levels from such products are greater than background radiation levels which are usually encountered. Depending upon the circumstances, products of this type may lead to a dose equivalent which does not fulfill the present accepted philosophy of "as low as practicable."

(4) The concentration found in the leach solution is greater than the allowable concentration (MPC) for chronic intake. If one assumes that uranium is leached out in solutions used in the cooking and preparation of food, then it is conceivable that an intake of uranium would result. This fact should be considered along with the other toxicants found in the leach solution, e.g., lead.

(5) It is evident that the use of source material in pottery, tableware, and glassware is of small or no benefit to the health or well-being of the public (and may be detrimental). The application results in

Table 8.2. Film Badge Readings Obtained from Surface Contact with Ceramic Glazes(1)

Color	Type of Earthenware	Location of Badge	24-Hour Reading (mrad)				Maximum Gamma Plus Beta (mrad/hr)
			Minimum(2) Gamma	Maximum(2) Gamma	Beta	Total Maximum Plus Beta	
Chocolate Brown	Dinner dish round-scalloped edge	Center of plate	0	70	140	210	9
Orange	Same	Same	0	60	140	200	8
Orange	Dinner dish square-scalloped corner	Same	0	50	150	200	8
Orange	Dinner dish square-scalloped corner	Same	0	40	150	240	10
Orange	Dinner dish round	Same	0	40	90	130	5
Orange	Saucer round	Same	0	60	160	220	9
Orange	Cereal bowl	Same	27	140	330	470	20
Orange	Saucer round-scalloped edge	Same	0	85	160	245	10
Orange and Blue	Coffee cup	Lip portion outside	0	70	160	230	10
Orange	Coffee cup	Lip portion inside Lip portion outside	0	80	210	290	12
Orange	Water mug	Lip portion inside Lip portion outside	0	80	260	340	14
Orange	Coffee cup	Lip portion inside Lip portion outside	0	110	240	350	15
Orange	3-sectional dinner plate	Center	0	40	110	150	6
Green, Orange, and Blue	Matching dish and bowl	Dish, edge orange bowl, on orange base	0	90	280	370	15
Lemon Yellow	Dinner dish	Center of plate	0	3	10	13	0.5

(1) From Menczer, 1965

(2) Since some of the soft gamma components are difficult to distinguish from high energy beta particles, gamma components are expressed in terms of "minimum" or "maximum" gamma

unnecessary radiation doses and accordingly such use should not be allowed.

A joint study by ORNL and the NRC was performed in 1974 to estimate individual and population doses resulting from the distribution, use, and disposal of ceramic tableware (O'Donnell 74). It was conservatively assumed that the tableware contained 20 percent natural uranium in the glaze, and that such radioactive tableware was used in feeding establishments. Following these assumptions, it was calculated that dishwashers, waiters, and patrons would be the only recipients of measurable doses in this hypothetical situation. Estimated doses were about 34 mrem/yr to dishwashers, 8 mrem/yr to waiters, and 0.2 mrem/yr to patrons.

8.4 SUMMARY

The use of uranium compounds in glassware and ceramic tableware has been practiced for several decades. Most of this application was halted by the demand for uranium in the 1940's, causing substitute materials to be applied. At this time, it is believed that no dinnerware containing uranium glazing is currently being manufactured in the U. S. The application of uranium to produce fluorescence or irridescence in decorative glassware, such as vases or candleholders, is still being practiced. Since these products are ornamental in nature and not involved in food handling, they are exempt from Food and Drug Administration (FDA) regulatory controls, and are likely to represent much less of a public health risk than dinnerware.

In the case of both decorative glassware and glazed ceramic products, some exposure of the public to ionizing radiation will result. In some instances, these exposures may be significant. The currently promulgated philosophy is that all exposures should be kept to levels which are "as low as reasonably achievable" (ALARA), and that no exposures should be incurred without a commensurate benefit (NCRP 75). Since the use of uranium in these products results in little or no benefit in terms of the health or well-being of the public, and since suitable substitutes for uranium exist for this application, it can be concluded that the exposures incurred as a result of product distribution and use, while low in most cases, are unnecessary and unwarranted.

8.5 REFERENCES FOR SECTION 8

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9. ELECTRON TUBES

9.1 PRODUCT DESCRIPTION AND BACKGROUND

9.1.1 General Discussion

Radioactive materials are currently in widespread use in many electronic applications. Faster and more reliable operation is achieved through preionization of the gases encased in electron tubes. Among the most common uses of these electron tubes are voltage regulation, current surge protection, and as indicator lights.

Many difficulties exist which make an accurate and comprehensive radiological assessment of electron tubes containing radionuclides a formidable task. These difficulties arise as a direct result of the wide range of electron tube types and applications, and radionuclide types and quantities. In addition, little information is available regarding such aspects as occupational exposures and the distribution of tubes containing thorium. For these reasons, the assessments performed in this section will be general in nature, and will be limited to the exposure pathways for which sufficient information exists. This is justifiable since it can be shown that the individual and collective doses resulting from electron tube life span activities are very low.

9.1.2 Product Description

Indicator lights are provided in many appliances on the market, such as clothes washers and dryers, stereos, coffee makers, and pinball machines. Such lights typically contain tritium or Kr-85 with activities in the range of 1 to 5 uCi per unit. These tubes are constructed of glass, with typical dimensions of 0.2 inches in diameter and 1 inch in length. Some of these tubes are designed for a service life of 25,000 hours. Several hundreds of millions of such tubes have been distributed in the last few years.

Voltage regulators and current surge protectors are used in almost every type of consumer electronic equipment, especially integrated circuit devices and digital equipment. These tubes are constructed of glass and are on the order of 0.25 inches in diameter and 1 inch in length. The isotopes used in these devices are usually Cs-137, Ni-63, and Kr-85. Thoriated-tungsten is also frequently used in such devices. Activities in voltage regulator and current surge protector tubes are usually below 1 uCi per unit.

Spark gap tubes or glow lamps are used as starters for fluorescent lamps (see Section 10), and in electric blanket thermostats and other specialty devices. These tubes typically contain less than 1 uCi of either Co-60, Ni-63, Kr-85, Pm-147, or some other radionuclide. Several millions of these type devices are manufactured annually.

9.1.3 Distribution

The total distribution of electron tubes in the U. S. for the period 1970 to 1978 is presented in Table 9.1. As can be seen from these data, tubes containing Kr-85, H-3, and Pm-147, respectively, are the most predominant types, with several hundred million Kr-85 tubes having been distributed in recent years. The average distribution of these products on a yearly basis, as well as the average activity per unit, is presented in Table 9.2.

9.2 ENVIRONMENTAL IMPACT

9.2.1 Benefits

In general, electron tubes containing radioactive materials provide benefits related to the faster and more reliable operation, and prolonged life of electronic circuits. Benefits of this nature are certainly important from an economic or resource conservation point of view. Since some of these products are employed in systems which are safety-related, the public health or safety may also benefit to some extent from product usage. The primary benefits, however, are those related to savings in cost.

Table 9.1. Total Distribution (1970-78) of Electron Tubes Containing Radionuclides(1).

Radionuclide	1970	1971	1972	1973	1974	1975	1976	1977	1978
Kr-85									
Units	34,300,000	82,900,000	70,300,000	95,500,000	168,000,000	114,000,000	139,000,000	58,200,000	120,000,000
Total Activity(2)	18,100	31,800	19,000	25,300	7,680	10,400	17,400	21,900	10,400
H-3									
Units	3,280,000	9,840,000	10,100,000	23,200,000	13,800,000	6,400,000	8,030,000	3,210,000	56,100,000
Total Activity	37,200	115,000	133,000	225,000	235,000	403,000	445,000	515,000	733,000
Co-60									
Units	184,000	145,000	118,000	44,400	53,000	37,900	43,400	37,200	41,300
Total Activity	5.92	15.3	20.1	19.5	14.1	13.2	12.5	11.3	8.1
Cs-137									
Units	377	23,000	11,400	13,700	22,400	30,100	24,800	13,300	12,300
Total Activity	0.494	20.8	10.1	10.5	19.1	48.0	29.1	15.8	16.4
Ni-63									
Units	61,000	57,300	83,300	59,600	70,100	103,000	69,600	158,000	169,000
Total Activity	18.5	23.6	20.3	25.5	45.4	77.9	93.7	161.0	237.0
Pm-147									
Units	1,690,000	1,470,000	2,390,000	3,910,000	3,860,000	1,540,000	2,210,000	3,180,000	1,530,000
Total Activity	5,990	5,240	20,800	35,500	47,800	10,300	14,700	16,400	31,000
C-14									
Units	--	--	--	--	52	--	10	--	--
Total Activity	--	--	--	--	2.44	--	0.470	--	--

Totals

Totals	Activity (mCi)
Kr-85	162,000
H-3	2,840,000
Co-60	121
Cs-137	170
Ni-63	703
Pm-147	188,000
C-14	2.91

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(1) Does not include tubes containing thorium

(2) All radionuclide activities are in units of mCi

Table 9.2. Annual Average Distribution of Electron Tubes Containing Radionuclides. (1)

Radionuclide	Average Annual Distribution		
	mCi/yr	units/yr	mCi/unit
H-3	3.2E+5	1.5E+7	2.1E-2
C-14(2)	3.2E-1	6.9E+0	4.7E-2
Co-60	1.3E+1	7.8E+4	1.7E-4
Ni-63	7.4E+1	8.1E+4	9.1E-4
Kr-85	1.8E+4	9.8E+7	1.8E-4
Cs-137	1.9E+1	1.7E+4	1.1E-3
Pm-147	2.1E+4	2.4E+6	8.7E-3

(1) Data averaged over period of 1970 to 1978

(2) Only distributed in 1974 and 1976

9.2.2 Radiological Impact

Members of the general population receive a small radiation dose as a result of the widespread use of radioactive materials in electron tubes. As mentioned previously, this section will present the results of assessments which are necessarily general in nature with a substantial degree of uncertainty. Conservatism has been applied in these assessments to ensure that the radiological impact of product life span activities does not underestimate the actual radiological health impact.

9.2.2.1 Occupational Exposures

Conversations with many of the manufacturers or distributors of electron tubes containing radionuclides has led to the conclusion that a useful estimate of the occupational exposures incurred as a result of this life span activity cannot be made at this time. In many instances, workers are not monitored for exposure, while in others workers are monitored but the company representatives contacted were either unwilling or unable to supply the required information. However, the relatively small quantities of radioactive materials used, together with the fact that most of the assembly processes are highly automated, indicate that these exposures are probably very low.

9.2.2.2 Exposures During Distribution

Exposures to the populations groups involved in the transportation and wholesale and retail sales of these products were not assessed. As a result of the small amounts of radioactive materials involved, short exposure times, and smaller population, the collective doses from these lifespan activities are probably considerably less than those associated with normal product usage.

9.2.2.3 Exposure To Users

Three general scenarios are considered for the evaluation of external exposures from radioactive materials in electron tubes. Only gamma-emitting radionuclides are considered in this evaluation. Although some external radiation doses will result from bremsstrahlung production within the tube, the magnitude of these doses will certainly be much lower than those from gamma emissions.

The three exposure scenarios which are considered are as follows:

(1) Each tube exposes one individual at an average unshielded distance of three meters for a duration of 2920 hours per year.

(2) Each tube exposes two individuals at an average unshielded distance of two meters for 4380 hours per year.

(3) Each tube exposes four individuals at an average unshielded distance of two meters for 8760 hours per year.

The results of these assessments are presented in Table 9.3. The highest whole body doses (Scenario 3) are 2300 person-rem to the population and 0.74 mrem to the average individual. Although Cs-137 is the largest contributor to the individual doses, the relatively small number of tubes distributed reduces its importance regarding the collective dose. Kr-85 is the largest contributor to the population dose, followed by Co-60 and Cs-137, respectively. Even though the external radiation levels from Kr-85 tubes are much lower than those from Co-60 or Cs-137, their abundance is much greater, and this widespread distribution is the predominant factor.

Another mode of exposure which should be evaluated is inhalation or absorption through the skin of HTO leaking from tritium electron tubes. This can be done by comparing this exposure scenario with that performed by McDowell-Boyer and O'Donnell (1978) for tritium luminous clocks. If it is assumed that tritium leaks at about the same rate from both clocks and tubes, and that these products are located similarly within personal environments, the collective dose from clock usage can be scaled to estimate the dose from electron tube usage. McDowell-Boyer and O'Donnell calculated a dose of 520 person-rem from a distribution of 5500 Ci of tritium in clocks. Since about 3200 Ci of tritium in electron tubes are currently in use (assuming an average product lifetime of ten years), a population dose of 300 person-rem is estimated.

9.2.2.4 Product Disposal

Electron tubes, like other consumer products containing radioactive material, can be disposed of with household or commercial refuse. This section assesses the radiological impact of product disposal by incineration and by

Table 9.3. External Total Body Doses from Three Exposure Scenarios Involving Electron Tubes Containing Gamma-Emitting Radionuclides.

	Dose (rem or person-rem)					
	Average Individual			Total Population		
Radionuclides	1	2	3	1	2	3
Co-60	4.3E-5	1.5E-4	3.0E-4	3.4E+1	2.3E+2	9.2E+2
Kr-85	4.3E-8	1.4E-7	2.8E-7	4.2E+1	2.8E+2	1.1E+3
Cs-137	6.7E-5	2.2E-4	4.4E-4	1.1E+1	7.4E+1	3.0E+2
Total	1.1E-4	3.7E-4	7.4E-4	8.7E+1	5.8E+2	2.3E+3

Scenarios:

(1) Each tube exposes one individual at an average unshielded distance of 3 m for 2920 hr/yr

(2) Each tube exposes two individuals at an average unshielded distance of 1 m for 4380 hr/yr

(3) Each tube exposes four individuals at an average unshielded distance of 2 m for 8760 hr/yr

disposal in landfills. The dose to the waste collection population is not assessed, as this dose is certainly very low and not worthy of detailed evaluation.

The lack of data concerning such matters as the loss of radioactivity during incineration or by leaching in landfills has necessitated the utilization of assumed conditions. Table 9.4 presents a summary of the disposal parameter values assumed for the reference incineration and landfill disposal cases (see Appendix D for additional assumptions and methods).

Landfill Disposal

The results of the reference case landfill disposal assessment (Table 9.5) indicate that tritium is the main contributor to the population dose. This dose, however, is low (120 person-rem) and far below the level at which any health effects may be expected to result. Table 9.6 presents the results of the worst case landfill scenario. As can be seen from this table, resultant population doses are still much lower than the health hazard level, and as in the reference case, tritium is the radionuclide primarily responsible for the dose.

Incineration

The estimated dose to the entire U.S. population resulting from the incineration of electron tubes containing radioactive material is summarized in Table 9.7. The total body dose is estimated at 0.74 person-rem, while the dose to the bone of the population is about 1.1 rem. As in the case of landfill disposal, tritium is the radionuclide primarily responsible for the resultant dose, even though this dose is very small.

9.2.2.5 Accidents

Three accidental cases of exposure are considered here: a warehouse fire involving an inventory of ten percent of the average annual distribution of either tritium, Kr-85, or Pm-147 tubes. This amounts to 320 Ci of tritium, 1.8 Ci of Kr-85, and 2.1 Ci of Pm-147. Assuming the same set of conditions as in Section 2.2.2.4, and the fact that all of the involved radioactivity is released, the following doses (per involved individual) are calculated:

Tritium - 0.2 rem (total body)
Kr-85 - 4.3 rem (lungs)

Table 9.4. Summary of Reference Case Disposal Parameter Values for Electron Tubes Containing Radionuclides.

Radionuclide	Annual Distribution (mCi)	Annual Disposal (mCi)	Annual Incineration (mCi)	fs(1)	Annual Activity Buried Intact (mCi)	R_t(2) (yr^{-1})	R_c(3) (yr^{-1})	t_m(4) (yr)
H-3	3.2E+5	1.8E+5	1.8E+4	1.0E+0	1.6E+5	1.0E-1	NA(5)	9.6E+0
C-14	3.2E-1	3.2E-1	3.2E-2	1.0E+0	2.9E-1	1.0E-1	NA	9.6E+0
Co-60	1.3E+1	3.5E+0	3.5E-1	5.0E-1	3.2E+0	5.0E-2	5.0E-1	2.2E+4
Ni-63	7.4E+1	6.9E+1	6.9E+0	5.0E-1	6.2E+1	5.0E-2	5.0E-1	2.2E+4
Kr-85	1.8E+4	9.5E+3	9.5E+2	1.0E+0	8.6E+3	1.0E-1	NA	9.6E+0
Cs-137	1.9E+1	1.5E+1	1.5E+0	5.0E-1	1.4E+1	5.0E-2	5.0E-1	9.9E+3
Pm-147	2.1E+4	1.5E+3	1.5E+2	5.0E-1	1.4E+3	5.0E-2	5.0E-1	1.4E+4

(1) f_s = fraction of activity released by the incineration process

(2) R_t = leach rate of radionuclides in products buried intact (fraction/yr)

(3) R_c = leach rate of radionuclides in products previously incinerated (fraction/yr)

(4) t_m = total migration time required for activity to reach groundwater withdrawal point

(5) NA = not applicable

Table 9.5. Fifty-year Dose Commitments Resulting from Reference Case Landfill Disposal of Electron Tubes Containing Radionuclides.(1)

Radionuclide	Amount Ingested (μCi)	Dose Commitment (person-rem or organ-rem)				
		Total Body	Liver	Bone	GI Tract	Gonads(2)
H-3	1.5E+6	1.2E+2	1.2E+2	1.2E+2	1.2E+2	1.2E+2
C-14	8.0E+0	5.6E-2	5.6E-3	8.8E-3	5.6E-3	5.6E-3
Kr-85(3)	7.0E+4	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0
All Others	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0

(1) See Table 9.4 and Appendix D for assumptions and methodology

(2) Dose assumed to be same as that for total body

(3) Because of its inert properties, the dose conversion factor for ingestion of Kr-85 is usually taken to be zero, even though some very small dose would result

Table 9.6. Fifty-year Dose Commitments Resulting from Worst Case Landfill Disposal of Electron Tubes Containing Radionuclides.

Radionuclide	Amount Ingested (μCi)	Dose Commitment (person-rem or organ-rem)				
		Total Body	Liver	Bone	GI Tract	Gonads
H-3$^{(2)}$	3.8E+6	3.0E+2	3.0E+2	3.0E+2	3.0E+2	3.0E+2
C-14$^{(2)}$	8.0E+0	5.6E-3	5.6E-3	8.8E-3	5.6E-3	5.6E-3
Co-60$^{(3)}$	6.7E+0	3.1E-2	1.4E-2	3.1E-2	2.7E-1	3.1E-2
Ni-63$^{(3)}$	1.5E+3	6.6E+0	1.4E+1	2.0E+2	2.9E+0	6.6E+0
Kr-85$^{(2)}$	2.0E+5	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0
Cs-137$^{(3)}$	2.7E+2	1.9E+1	3.0E+1	2.2E+1	5.4E-1	1.9E+1
Pm-147$^{(3)}$	1.1E+3	3.2E-3	7.8E-3	8.3E-2	9.8E+0	3.2E-3
Total		3.3E+2	3.4E+2	5.2E+2	3.1E+2	3.3E+2

(1) See Appendix D for assumptions and methodology

(2) Assumes R_t value of 10 times the reference case values

(3) Assumes R_t and R_c values of twice the reference case values

Table 9.7. Fifty-year Dose Commitments Resulting from the Incineration of Electron Tubes Containing Radionuclides. (1)

Radionuclide	Amount Inhaled (μCi)	Dose Commitment (person-rem or organ-rem)					
		Total Body	Liver	Bone	Lungs	GI Tract	Gonads(2)
H-3	9.1E+3	7.3E-1	7.3E-1	7.3E-1	7.3E-1	7.3E-1	7.3E-1
C-14	1.7E-2	1.2E-5	1.2E-5	1.9E-5	1.2E-5	1.2E-5	1.2E-5
Co-60	8.7E-3	1.7E-5	1.2E-5	1.7E-5	6.5E-3	3.1E-4	1.7E-5
Ni-63	1.9E-1	3.4E-4	7.4E-4	1.0E-2	4.2E-3	3.2E-4	3.4E-4
Kr-85	4.8E+2	0.0E+0	0.0E+0	0.0E+0	1.2E-3	0.0E+0	0.0E+0
Cs-137	3.9E-2	2.1E-3	3.0E-3	2.3E-3	3.7E-4	4.3E-5	2.1E-3
Pm-147	3.9E+0	1.2E-2	3.1E-2	3.3E-1	2.6E-1	2.1E-2	2.1E-3
Total		7.4E-1	7.6E-1	1.1E+0	1.0E+0	7.5E-1	7.4E-1

(1) See Table 9.4 and Appendix D for assumptions and methodology

(2) Dose assumed to be same as that for total body

Pm-147 - 0.2 rem (total body)
4.3 rem (bone)

9.2.2.6 Summary of Radiological Impact

The total radiological impact resulting from both normal life span activities and accidents involving radioactive material in electron tubes is very low, although the level of uncertainty associated with the estimated impact is large. The life span activity which is the largest contributor to this dose is normal product usage. Doses associated with product disposal are very small, and due mainly to the presence of tritium.

9.3 ALTERNATIVES

The types and applications of electron tubes are numerous and no attempt has been made here to identify and evaluate possible alternatives to the products currently in use. This is justifiable in light of the fact that individual and population doses resulting from the distribution, use, and disposal of these products are insignificantly low.

9.4 COST AND RISK VERSUS BENEFIT

The cost associated with these products is, of course, widely variable depending on the type of tube and application. The benefits gained by incorporating radioactive material into these products are similarly variable, ranging from increased operating efficiency for non-essential items such as appliance indicator lights to current surge protection and voltage regulation for important electronic systems. In general, it can be stated that the degree of benefit gained is at least commensurate with cost. If this was not the case, these products would not persist as they have in the competitive open market.

It has been shown in the previous assessments that the risk to the public health from radioactive material in electron tubes is very low, if not negligible. The benefits gained from the use of these products, while difficult to quantify, are significant. Thus, it is reasonable to conclude that electron tubes containing radioactive material offer benefits which are at least commensurate with the level of risk, and probably far in excess of that level.

9.5 FUTURE USE

The use of radionuclides in electron tubes has been in common practice over the last few decades. As can be seen from Table 9.1, only tritium and Kr-85 have shown an increase in use over the last few years. Since these radionuclides are commercially available in gaseous form at a moderate price, their popularity will probably continue to persist in the near future. The use of Co-60 and Cs-137 seems to have declined somewhat in recent years, although this trend is tenuous. Thoriated-tungsten tubes, which are not assessed here, will probably also enjoy widespread application in the near future. In addition, Ni-63 is currently being evaluated for a number of new applications.

9.6 SUMMARY

Radioactive materials are commonly used in various types of electron tubes for such purposes as current surge protection or voltage regulation, and as indicator lights or fluorescent lamp starters. The most prevalent types of radionuclides used for these purposes are tritium, Pm-147, and Kr-85. Assessments performed for the normal use and disposal phases of the product life span, as well as for accidents, indicate that the level of risk associated with these products is very low. Since some degree of benefit is gained by the incorporation of radionuclides into electron tubes, and since this benefit is not achieved at a large increase in cost, these products can be considered acceptable from both a cost- and risk-effectiveness point of view.

9.7 REFERENCES FOR SECTION 9

McDowell-Boyer, L. M. and F. R. O'Donnell, *Radiation Dose Estimates from Timepieces Containing Tritium or Promethium-147 in Radioluminous Paints*, ORNL/NUREG/TM-150, Oak Ridge National Laboratory, Oak Ridge, TN, 1978b.

10. FLUORESCENT LAMP STARTERS

10.1 PRODUCT DESCRIPTION AND BACKGROUND

10.1.1 General Discussion

A fluorescent lamp starter is a device which functions to preheat the filaments of a fluorescent lamp. When fluorescent lighting was first designed and marketed there were no fluorescent starters in their circuit and the functional life of these lamps was a few hundred hours. The design of circuitry, which includes some method of preheating the filaments of the fluorescent lamps, has managed to extend the functional life of a fluorescent lamp to 7000 hours or more.

In the late 1940's and early 1950's uranium was used in some of the more popular models of fluorescent lamp starters. The uranium was handled under license from the Atomic Energy Commission (AEC). In the mid 1950's, the industry switched to thorium in place of uranium, again under license from the AEC. Eventually it was realized that the amount of thorium being used in the manufacturing of fluorescent lamp starters was of such a small quantity annually that the industry was allowed to continue to handle the radioactive material under a license exempt status.

Industry spokespersons estimate that ten to 15 years ago, the U. S. fluorescent lamp starter market amounted to an annual volume of 75 to 80 million units. The U. S. Census Report, S.I.C. 3643096, for 1978 indicates an annual U. S. sales volume for fluorescent lamp starters of 18.4 million. This drop in sales volume is the reflection of functional changes in fluorescent lamp circuitry. Today the consumer market is the main area of use for the 40 watt fluorescent lamp and it is this device that still requires the aid of a starter circuitry.

10.1.2 Product Description

The basic circuitry for one type of fluorescent lamp and starter is illustrated in Figure 10.1 (Baumeister 1978). Inside the fluorescent lamp starter, a bimetal contact is in a normally closed position prior to the flow of electricity. Electrons are capable of leaving the surface of the contacts if enough electromotive force is applied. The ability of these electrons to leave the surface of the contact is a measure of the work function of the metal. The lower the work function of the metal, the easier it is for the electrons to leave the surface. One method of lowering the work function is to have an ionized gas surrounding the metal. This is accomplished by surrounding the metal contacts with an atmosphere of neon or argon gas. To ionize gas prior to the delivery of electric current, a small amount of radioactive material is included. In the case of fluorescent lamp starters found in the U. S. the radioactive material is mostly Th-232. This radioactive material supplies just sufficient energy to ionize enough gas molecules to accomplish the necessary lowering of the work function. When current is applied through the starter circuit the bimetal heats and contracts until eventually the contact in this circuit is opened. By this time the filaments of the fluorescent lamp have heated sufficiently to commence ionization of mercury molecules within the lamp. When the starter circuit opens it throws a potential across the lamp which causes it to discharge. At this point both the starter and the lamp have sufficient ionizations within them to keep the bimetal of the starter warm, its circuit open, and to maintain the mercury in the lamp in an ionized state, leading to the production of visible light.

A lead glass container surrounds the bimetal filaments of the fluorescent starter and creates an air tight seal which restricts the contamination of the argon or neon gas that is the atmosphere inside the container. To further maintain the gas in a pure state a getter material is applied to the inside surface of the lead glass container. The getter material is composed of a mixture of barium, magnesium, and thorium. The mixing ratio of the getter chemicals changes depending on specified needs.. In some cases strontium metal is also added to the mixture. The thorium concentration in the getter mixture ranges from three to six percent by weight.

In the early days of manufacture of fluorescent starters the getter material was applied to the inside surface of the glass container in the form of a paint. These early applications of getter material were applied by hand and

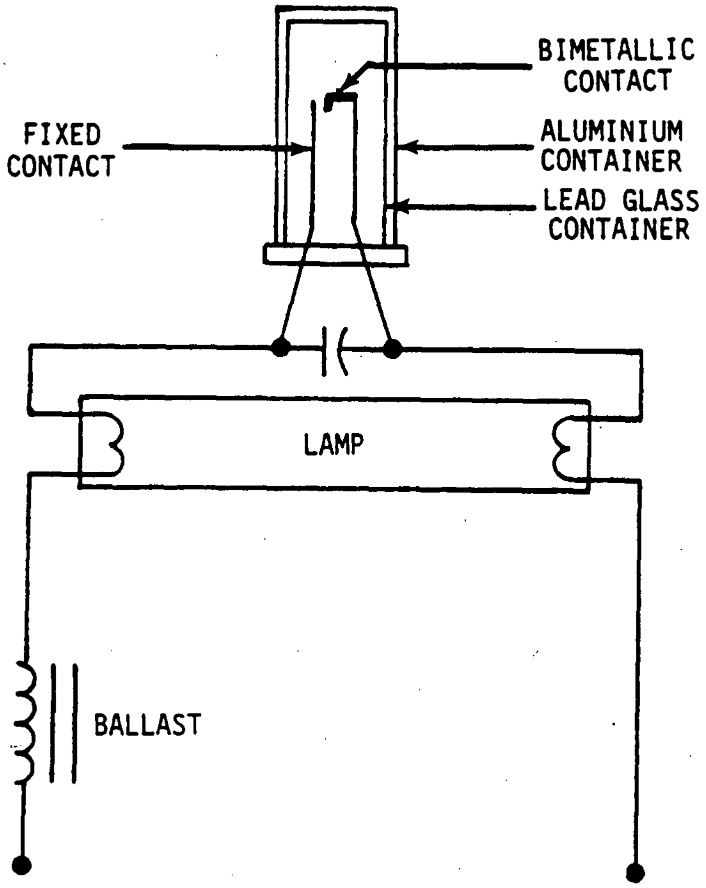

Figure 10.1. Fluorescent Lamp and Starter Circuitry.

contained uranium. Today the getter material represents about 0.75 mg of material and initially is a small bead on the surface of a 0.1 cm^2 piece of metal. The metal is mounted on one of the filaments of the starter and the device is encapsulated within the lead glass container. When the starter is subjected to a radio frequency transmission the getter material is vaporized and deposited on the inside surface of the glass container. The getter material, including the thorium, is contained within the sealed lead glass container. A second seal occurs when this unit is then sealed within an aluminum outer casing.

10.1.3 Distribution

Fluorescent lamp starters are distributed throughout the U. S. and are available to the consumer at most retail outlets that supply electrical lighting products. The 1980 editions of the U. S. Industrial Product Directory (USIPD-80) lists some 15 companies as wholesale distributors of fluorescent lamp starters. Few of these distributors consider themselves to be manufacturers of this product. There is presently estimated to be no more than six and possibly as few as four companies in the U. S. that profess to be manufacturers of fluorescent lamp starters

Fluorescent lamp starter manufacturers in the U. S. do not manufacture their product in the U. S. They either purchase the items from a foreign manufacturer or have foreign subsidiaries that manufacture the starters for them. Areas of the world from which these products reach the U. S. market are Costa Rico, Brazil, Korea, Japan, and the Philippines.

The getter material that is the source of the thorium in units sold in the U. S. is primarily manufactured by one company in the U. S. This getter manufacturer may have on hand, at any one time, a maximum of 500,000 getter sources (see Table 10.1). These sources are sent either to the U. S. manufacturer of the fluorescent starters or directly to manufacturers' assembly plants outside the U. S.

In the U. S., the manufacturer of fluorescent lamp starters may have on hand in a single warehouse a maximum of 50,000 units. Inside this warehouse there may be a maximum of 25,000 units stored in a single location. Distributors of the starters may stock a maximum of 2000 units at one time, while in the retail outlet the maximum storage at a single time would be no more than 100 items. In reference to the specific type that contains thorium, it is unlikely

Table 10.1. Distribution of Fluorescent Lamp Starters Containing Thorium.

	Getter Manufacturer	Starter Manufacturer	Wholesale Distributor	Retail Distributor
Number of Companies	1	6	15	12,000
Average Number Workers Per Company	10	300	250	25
Total Number Workers Per Category	10	1,800	3,750	300,000
Average Number Starters Per Location	500,000	50,000	2,000	20
Total Number Starters Per Category	500,000	300,000	30,000	240,000

that more than 20 of these would be stored at any one time. For the consumer it is estimated that a total of ten fluorescent lamp starters, spread throughout the living space, is the maximum that could be found.

10.2 ENVIRONMENTAL IMPACT

10.2.1 Benefits

The benefit gained from the use of fluorescent lamp starters that contain radioactive material is principally one of cost to the consumer. Environmentally speaking, the benefit derived from these starters is in their end use, that is, the production of artificial light. That these starters are inexpensive and that their end use produces an equivalent light source at half the watt consumption of conventional tungsten lamps is very important in today's world of limited available electricity and rising energy prices.

In terms of energy savings, a hypothetical case may be enlightening in comparing benefits. A power plant that manifests 30 percent efficiency in burning oil to generate electricity realizes approximately 609 kilowatt-hours of delivered electricity per barrel of oil consumed. If 100 million tungsten lamps averaging 100 watts each are burned ten hours per day for one year, the oil consumed will amount to approximately 60 million barrels. If all these lamps were fluorescent, the oil consumption requirement could be reduced one-half, thus, saving approximately 30 million barrels of oil per year. The significance of this savings is left for the reader to judge.

Environmentally speaking, in terms of waste disposal, the benefit of the fluorescent lamp starter can be considered at least two ways. Although the fluorescent lamp associated with the starter lasts some ten times that of equivalent tungsten lamps, the volume of waste will be similar due to the bulkier fluorescent lamps. But, the energy requirements to transport and dispose the tungsten lamps may be greater since ten trips are required rather than a single trip for fluorescent lamps.

10.2.2 Radiological Impact

No study has been located on the actual measurement of the radioactivity, if any, that can be detected from a fluorescent lamp starter. One manufacturer reported trying to determine gamma emissions but reported no readings could be detected even when the outer aluminum casing was removed from the starter. What measuring equipment was used and how these measurements were taken is not known.

For a 0.06 MeV gamma ray such as that from Th-232 the absorption coefficient (u) is approximately 0.6 cm^{-1} (Casarett 68). This u value can be used to theoretically determine the fraction of gamma rays remaining after passage through the aluminum casing of the fluorescent starter. The relationship, $I/I_0 = e^{-ud}$, yields a value of 94 percent as the amount of gamma rays remaining. This value is not for a broad beam application such as exists in our model but serves to indicate that most of the gamma rays will be exiting the container. To absorb one-half the 0.06 MeV gammas a thickness of 1.2 cm of aluminum would be required. Absorption coefficient values for the lead glass surrounding the source are not given but this glass, combined with the aluminum, is not expected to develop a significant absorption coefficient. Also, since the Th-232 is assumed to have been recovered at least 20 years previously, the gamma emissions from daughters contribute even more. Because of this, the gamma emissions from the fluorescent lamp starters will be considered to be 100 percent of the thorium content.

10.2.2.1 Occupational Exposure

Getters manufactured in the U. S. are composed of barium, magnesium, and thorium. The getters weigh approximately 0.75 mg and the thorium content is from three to six percent by weight, depending on manufacturers' requirements. Six percent is being used for a conservative figure which gives a thorium content of 4.5×10^{-5} grams. This amount of thorium is equivalent to 5 pCi of Th-232 per starter. The annual distribution in the U. S. of five million starters with thorium getters gives an annual distribution of 25 uCi of Th-232. Based on the distribution data of Table 10.2, there is a total of 5.2 uCi distributed throughout the fluorescent lamp starter industry at any one time.

The thorium oxide used in the manufacture of getters is physically blended with magnesium, barium, and, in some cases, strontium. The mixture is processed into small beads which are mounted on approximately one square centimeter of metal backing. These items are then stored for a period of time until the starter manufacturer requires them. At this point, the amount of finished getters being stored can amount to 500,000 items.

The getters that are produced for domestic manufacture are all made with thorium as their radiation source. So far as can be determined there appears to be a sole source for U. S. manufactured getters and in this operation probably no more than ten workers are in an area of contact with the getter material. An estimation of the maximum number of workers coming into contact with fluorescent lamp starters is given in Table 10-1. This table indicates the distribution of workers throughout various categories of the industry and the number of starters that can be found at a single location during a single period of time. The number of workers indicated may not in every case share equally in the duration of exposure to the starters but each will be considered to be within one meter of the starters for two hours per working day or 500 hours per year.

For the individual workers within the distribution system of starter manufacturers, the maximum exposure possible occurs to the ten workers in the manufacturing area of the getters. A storage of 500,000 getters would amount to an equivalent of 2.5 uCi of thorium contained within the workplace. These sources are manufactured by these employees and, thus, constitute the major exposure potential for any step in the manufacturing of fluorescent lamp starters.

Without considering any protection factors, the following represents calculations for the "worst case" (getter manufacture) situation. (Worst case based on per individual exposure). At 5 pCi per getter (six percent thorium) and using 1.3 R/hr at 1 m/Ci as the gamma-ray constant (see Appendix A) the gamma exposure per unit would be

$$(1.3 \text{ R/hr-Ci})(5\text{x}10^{-12} \text{ Ci}) = 6.5\text{x}10^{-12} \text{ R/hr.}$$

Extrapolating from the dose factor curves of Appendix A, the whole body dose factor for gamma rays from thorium is 0.67 rem/R. Multiplying the derived gamma exposure of $6.5\text{x}10^{-12}$ R/hr by the dose factor of 0.67 rem/R, the maximum whole body dose rate from a single getter would be

$$(6.5\text{x}10^{-12}\ \text{R/hr})(0.67\ \text{rem/R}) = 4.4\text{x}10^{-12}\ \text{rem/hr}.$$

To illustrate the resulting low exposure rate, even if a worker could be exposed to all 500,000 getters for 2000 hr/yr (impossible situation), the resulting dose would only be

$$(4.4\text{x}10^{-12}\ \text{rem/hr})(2000\ \text{hr/yr})(500{,}000) = 4.4\text{x}10^{-3}\ \text{rem/yr}.$$

For this reason no occupational doses have been calculated.

10.2.2.2 Exposures During Distribution

Since the external exposure from the units was found to be insignificant in the preceding section and since no internal contamination is assumed to occur during normal distribution, the radiological impact during distribution is assumed to be essentially zero.

10.2.2.3 Exposures to Users

There is no existing record that gives any indication as to the number of functioning fluorescent lamps that require thoriated starters. The Census Bureau (USBOC 79) estimates approximately 75 million occupied housing units existed in the U. S. in 1972. If we assume one-third of these units are each equipped with two fluorescent lamps requiring thoriated starters we have 25 million housing units and 50 million thoriated starters. The average occupancy of housing units in 1977 was 2.8 persons per unit. This yields an exposured population of 70 million persons.

To give some justification to the population estimates we shall look at some functional values for fluorescent lamp starters. It is estimated by some manufacturers that a fluorescent lamp starter has a functional life equal to that of two fluorescent lamps. At 7000 hours of functional life per fluorescent lamp, an average starter has a functional life of 14,000 hours. At four hours of use per day the life time of the starter would be approximately ten years. At the present time, five million thoriated fluorescent lamp starters are estimated to be sold in the U. S. per year. With 50 million starters in use this gives an annual turnover rate of ten percent. Some manufacturers have indicated this to be a reasonable estimation.

A single thoriated fluorescent lamp starter contains an estimated 5 pCi of Th-232 (20 years since thorium recovery). Assuming an average of two hours per day exposure for consumers that own starters and using the external dose equation and conversion factors as in Section 10.2.2.1 , the dose rate to the whole body from a single starter, at one meter, is 4.4×10^{-12} rem/hr. With two starters the maximum dose rate at one meter is 8.8×10^{-12} rem/hr. If we assume two hours per day exposure, the individual consumer receives an annual dose of 6.4×10^{-9} rem or 3.2×10^{-9} rem per starter. For the consumer population of 70 million persons the dose rate is 0.45 person-rem per annum.

10.2.2.4 Product Disposal

The very low activity associated with individual fluorescent lamp starters makes much of the disposal analysis an insignificant issue. With a total of five million thoriated starters being disposed of annually, there are only 270 starters entering each waste disposal site (assuming 18,500 sites per Appendix D) per year on the average. For waste incineration, the levels of radiation associated with starter disposal are similar or lower than those calculated for gas mantles and welding rods. These calculations indicated negligible amounts for all incineration dose calculations (see Sections 5 and 6).

Landfills

With only 270 starters entering each landfill annually, there is no meaningful calculation that can be performed for a single site. For a worst case estimation, all starters will be considered disposed of at "worst case" sites. Table 10.2 is the result of these calculations based on methodologies found in Appendix D.

10.2.2.5 Accidents

A "worst case" accident has been calculated for a warehouse fire in which 500,000 fluourescent lamp starters were stored. For this scenario, all thorium activity is assumed released during the fire and the firefighters are exposed for a total of eight hours. Fifty-year dose commitments were calculated for the firefighter and the maximum dose was found to be 0.47 rem to the bone.

Table 10.2. Fifty-Year Organ Dose Commitments from Worst Case Scenario of Landfill Disposal of Five Million Thoriated Fluorescent Lamp Starters.

Organ	Collective Fifty-year Dose Commitment (Organ-rem)	
	Th-232(1)	Ra-228(2)
Total Body	3.0E+3	3.0E+1
Liver	5.5E+0	5.0E-3
Bone	3.0E+3	2.8E+1
Lungs	3.0E+3(3)	3.0E+1(3)
G.I. Tract	3.1E+3	3.0E-1
Gonads	3.0E+3(3)	3.0E+1(3)

(1) Dose commitments from ingestion of 2.5E+1 uCi of Th-232 in equilibrium with its daughters

(2) Dose commitments from ingestion of 2.5E-1 uCi of Ra-228 in equilibium with its daughters

(3) Dose conversion factor (DCF) data unavailable; DCF assumed to be same as for total body

10.2.3 Non-Radiological Impact

Barium, thorium, and magnesium are the basic materials of the getters found in a fluorescent lamp starter. Lead glass and aluminum are the main encasement components. A minute amount of neon or argon gas is encapsulated within the getter. Some copper, tin, or other material makes up the bimetal wire construction.

The Federal standard for magnesium oxide fumes is 15 mg/m^3 (USDHEW 77). The permissible exposure level for soluble barium compounds is 0.5 mg/m^3. Both items show little, if any, toxic effects in humans at these concentrations. For the worst case situation, a firefighter in a warehouse with 500,000 starters (see Section 10.2.2.5), the atmospheric levels of barium and magnesium will both have a possibility of exceeding Federal standards during a transient phase of the fire. No health consequences are expected since the volume of the warehouse will quickly reduce this concentration.

Based on the above arguments, a non-radiological consequence is not expected to be of significance in relation to the contents of fluorescent lamp starters.

10.2.4 Summary of Impact

The benefit related to thoriated fluorescent lamp starters is not directly related to them but to the light available at low cost because of their inclusion in the lamp circuit.

No major environmental impact was found for any manufacturing, distribution, use, or disposal operation related to the thoriated fluorescent lamp starters. The population dose was found to be 0.45 person-rem to the whole body. For an accident scenario, the worst possible case resulted in a fifty-year dose commitment to a firefighter and amounted to 0.47 rem to the bone.

10.3 ALTERNATIVES

10.3.1 Radiological Alternatives

At least five radionuclides have been or are being used in fluorescent lamp starters: U-238, Th-232, promethium-147 (Pm-147), Kr-85, and tritium. Of these nuclides only thorium and possibly Pm-147 are found in starters sold for use within the U. S. consumer market. Of the two radionuclides used in the U. S. consumer products, conversations with manufacturers indicate that thorium is by far the dominant radionuclide found in these products.

No replacement of thorium with a different radionuclide is anticipated within the conceivable near future. There appears to be a sole-source manufacturer for the thoriated getter material in the U. S. and the market for this product has been declining so much in recent years that the cost effectivenes of changing to a different radionuclide seems certain to prevent this.

Pm-147 is a radionuclide used in fluorescent lamp starters manufactured in Japan. The Japanese government is presently encouraging the manufacturers of these starters to do research into the possibility of replacing Pm-147 with a material of lower radiotoxicity and thus reduce the potential dose burden of workers exposed to this material.

The radionuclide most likely to replace Pm-147 is nickle-63 (Ni-63). Like Pm-147, Ni-63 is a beta emitter but the average beta particle energy is less than one-third that of Pm-147 (Pm-147 - 0.225 MeV, Ni-63 - 0.067 MeV). Although it is not certain what quantity of Ni-63 would be required, the amount of Pm-147 in each starter is reported by manufacturers to be about 0.1 uCi.

10.3.2 Non-Radiological Alternatives

Fluorescent lamps that have filament electrodes which are heated by passing current through them are subdivided into two groups: preheat lamps and rapid start lamps. Preheat lamps are the type which require a starter in their circuit and would be of the type the present study is concerned with. There has been a trend, for a number of years, to get away from the preheat circuit type fluorescent lamps and toward the installation of bulkier more expensive instant start circuits. Manufacturers have indicated that not more than 20 percent of the U. S. fluorescent lamp market is of the preheat circuit type.

For small installations, such as a unit that fits on top of a table lamp, the heavy instant start lamps are not as good a choice as the lighter preheat circuit type. There is an indicated change in the circuitry for fluorescent lamps that could eliminate the need of both starter and ballast (Scott 80). This change is the development of an electronic starter which claims not only to replace starter and ballast but to reduce current drain by 20 percent. A check with Underwriters Laboratories indicated no units have yet been submitted to them for their testing and evaluation (Underwriters Laboratories 80). Whether or not this electronic starter, which is a product developed in Korea, will ever reach the U. S. market and replace the preheat and rapid start circuit types of lamps can not be judged at this time.

10.4 COST AND RISK VERSUS BENEFIT

10.4.1 Cost

Fluorescent lamp starters are utilized by the consumer in what is basically two manners. The consumer either personally purchases a starter for placement in a fluorescent lamp fixture or the consumer purchases a fluorescent lamp fixture (or lamp) that has the starter already incorporated into it and in which case the starter is not a singularly replaceable item. In the second case, the consumer may be totally unaware of the existance of, or need for, the starter.

If a consumer has a fluorescent lamp fixture that requires a starter there is no option for this consumer other than to have a starter in the lamp circuit. If the lamp circuitry is designed for a preheat type fluorescent lamp, then the consumer must purchase a starter for this circuit and this means a starter that contains a small amount of radioactivity.

Replaceable fluorescent lamp starters are available to the consumer in most hardware stores, lighting fixture stores, and "electrical needs" departments of major general merchandising stores. These items vary in price according to the pricing structure of the individual supplies. For purposes of estimating costs, an average price of $1 per unit will be used.

For fluorescent lamp fixtures that have the starter incorporated in such a way that it is not a replaceable item, the consumer has no options as to what price they must pay for the starter. In this case the starter will be estimated to cost the consumer approximately three percent of the total fixture price. This then puts the starter in a price range of approximately $0.50

The most significant benefit to the consumer from the use of fluorescent lamps, especially during times of increasing energy costs, is that fluorescent lamps have an average watt loadage of about 0.4 times that of incandescent tungsten lamps of a similar light output. For the consumer, the initial benefit is not immediately apparent due to the initial increased cost for fluorescent lighting caused by the need of ballast and starters in this system. When the initial costs and the operating costs are combined and analyzed on a daily or yearly basis, the cost of producing equal amounts of luminosity strongly favor fluorescent lighting.

There currently exists three relatively new products available to the consumer that are of benefit in regards to convenience. One is a completely self-contained plug-in fluorescent lamp unit, disposable at end of life, and needing no further adjustments or additional hardware. A second product is a screw-in adaptor with replaceable circular fluorescent lamp that converts an ordinary tungsten table lamp fixtrue to a fluorescent lamp fixture. A third product is a completely self-contained item consisting of discharge tubes, starter, and other controls, which, as a unit, screws into a conventional lamp socket. This unit is currently only available in Europe. Since the two self-contained units have their starters built in, these items are discarded with the lamps during replacement.

As with any fluorescent lamp, the initial costs associated with these new consumer devices are considerably higher than those for incandescent lamps of a similar luminosity. Table 10.3 is a cost comparison of two of these fluorescent lamps to an incandescent lamp of similar luminosity. Even when costs are compared on a per usable hours-of-life basis, the fluorescent lamps cost from two to three times that of the tungsten lamp. But, when the cost of electricity to operate these lamps is included in the analysis, as in Table 10.4, there is a considerable change in the total costs of these units. On an annualized basis the tungsten lamp will cost approximately 2.2 times that of the fluorescent lamp when each is used for a period of six hours per day, 365 days per year. When initial lamp costs are factored in with the operating costs, the tungsten lamp

Table 10.3. Life, Luminosity and Initial Unit Cost Comparison for Fluorescent Lamp and Tungsten Lamp.

Lamp Type		Rated Average(1) Life(hr)	Lumins	Lamp Cost $	Lamp Cost Per Usable Hour
Fluorescent Lamp					
Fully Disposable	33W	5000 hr	1500	16.00	0.0032
Circular	38W	7500 hr	1750	18.00	0.0024
Incandescent	100W	750 hr	1740	0.75	0.0010

(1) Based on three hours per use.

Table 10.4. Operational and Total Cost Comparison for Fluorescent and Tungsten Lamps.

Lamp or Component		Operational Cost at 7¢ per KWhr-6 hr per day		Total Lamp Use Cost (Initial, Operational, Ballast)	
		Per Day	Per Year	Per Day	Per Year
Fluorescent					
Fully Disposable	33W	0.014	5.06	0.037	13.50
Circular	38W	0.016	5.83	0.034	12.41
Ballast	10W	0.004	1.53	—	—
Incandescent	100W	0.042	15.33	0.048	17.52

still costs the consumer approximately 30-40 percent more to operate than the fluorescent lamp. It is this benefit, which occurs in an indirect manner, that the consumer receives due to the fluorescent starter containing radioactive material.

10.4.2. Product Analysis

Risk to the consumer associated with thoriated fluorescent lamp starters is so low as to be insignificant. For a single lamp starter, the annual dose is calculated to be 3.2×10^{-9} rem. The benefit to the consumer is quality lighting at a lower cost than that received from tungsten lamps. If the annual dose to the consumer from the thoriated fluorescent lamp starter were compared to the dose received from the human body's natural concentration of potassium-40 (K-40), there would be a three million to one ratio in favor of the starter. This assumes a value of 19 mrem/yr whole body due to K-40 (Altman 77). Even for a population dose to 70 million consumers the dose associated with the thoriated starters is only 0.45 person-rem. Although thoriated fluorescent lamp starters are not an essential item, their benefit far outweighs their risk to the consumer.

10.4.3 Alternative Analysis

Two alternatives exist in regards to thoriated fluorescent lamp starters. The first is fluorescent lamps of different ballast design that eliminates this type of starter and a second is a Korean-made solid state product that has not yet been made available in the U. S.

The cost to the consumer is rapidly elevated as the lamp design is switched to the type requiring no thoriated starters. Frequently, this type requires a larger ballast and this results not only in higher costs but also noise problems. To overcome the noise problem as much as possible, these ballasts are packed with sand. It must, therefore, be concluded that this option is not one the general consumer would consider.

The Korean solid state fluorescent lamp starter was not available for study. A check with the Underwriters Laboratories indicated that they had not yet seen the item or knew of any evaluation related to it.

10.5 FUTURE USE

The use of thoriated fluorescent lamp starters has been in decline in recent years. At present the maximum number being sold is 18 million units annually. There are a few new consumer products that are being marketed which require these starters but there are also new concepts in solid state starters which can replace the thoriated starters. It would appear that the future for the thoriated fluorescent lamp starter is a sales volume of no more than its current volume of 18 million units annually and, quite possibly, a much lower volume.

10.6 SUMMARY

The thoriated fluorescent lamp starter contains approximately 5 pCi of Th-232 per unit. Based on the current consumer market of five million units annually this amounts to a distribution of 25 uCi of thorium per year. The consumer population was estimated to be currently 70 million people. This is higher than the current distribution of starters and reflects the declining market that is occurring and the ten-year life span of the starter. Based on these data, the consumer is estimated to receive 3.2×10^{-9} rem per starter used annually. For the consumer population this is an annual dose of 0.45 person-rem.

The cost benefit question concerning thoriated fluorescent lamp starters would appear to be answered strongly in favor of the starters. Whether or not there is any point of concern here may be of no consequence if the future use of these units continues to decline as the market appears to indicate.

10.7 REFERENCES FOR SECTION 10

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11. IONIZATION CHAMBER SMOKE DETECTORS

11.1 GENERAL

A detailed environmental assessment of ionization chamber smoke detectors containing americium-241 (Am-241) has been performed (Belanger 79), and the results of that assessment will be summarized here. For more detailed information, the reader is referred to the referenced work.

11.2 SUMMARY

Residential fires continue to pose a major problem in the U. S., ranking as the second most frequent cause of accidental death in the home. Between 7500 and 12,000 lives are lost in fires every year with 70 percent of these occurring in residential fires. Residential property loss due to fires is over one billion dollars annually. A key element in reducing loss of life and property is early detection of fires. Various devices are available that detect flames, heat, or products of combustion. In recent years devices that detect products of combustion have been introduced to the consumer market and their sales have increased dramatically. These so-called "smoke detectors" are of two types. The majority have been of the ionization chamber smoke detector (ICSD) type which contains small amounts of radioactive material. The remainder have been of the photoelectric type which contains no radioactive material. A combination detector has been recently marketed which contains both an ICSD and a photoelectric detector. Based on theoretical studies and case histories, the estimated percent of residential fire-related deaths that smoke detectors could save is between 41 and 89 percent.

Most ICSDs contain Am-241, a radioactive substance that emits ionizing radiation. Other radionuclides could be used in place of Am-241. In the past, radium-226 (Ra-226) has been used in detectors, but at present it is not employed in the consumer product. Nickel-63 (Ni-63) is presently being studied for use in consumer smoke detectors, having already been used in commercial applications, such as in large warehouses.

The user of an Am-241 ICSD is exempt from regulations, but the manufacturer must apply for and obtain a specific license from the NRC to distribute the product. The license is granted only after the applicant has demonstrated that the product is designed and will be manufactured in a manner such that specific requirements and safety criteria are met.

Since 1972-1973, the residential market for Am-241 ICSD's has increased rapidly. Fourteen million units containing a total of 41 curies of Am-241 were distributed in 1978, and 26 million units have been distributed since 1972. Based upon present projections, about 90 million units will have been distributed by 1986. The peak year of distribution should be 1978. The average activity per unit in 1978 was 3 uCi and projections indicate this will become lower in the future.

The distribution of photoelectric detectors has been minimal compared to ICSD's. Early photoelectric units were not as reliable as comparable ICSD's because of various reasons, the most notable of which was the use of incandescent bulbs. The introduction of low-power light emitting diodes has improved their reliability considerably. The cost of photoelectric units is greater than the cost of comparable Am-241 ICSD's, basically because of slightly more complex circuitry. The photoelectric units have been found to react quicker to smoldering or slow-burning fires while the ICSD's react quicker to fast-burning fires. Both appear to have responses comparable in terms of saving lives. However, it is felt the ICSD units may have a slight edge over the photoelectric units since response to fast-burning fires is more critical.

The non-radiological impacts due to use of both ICSD's and photoelectric detectors have been determined to be slight and indistinguishable from one another. For a typical residential installation, the risk of an occupant being exposed to non-radiological toxic products, due to burning, for example, is judged to be extremely small.

The use of Am-241 ICSD's does result in exposure of people to low levels of radiation. Analysis shows that the manufacture, distribution, normal use, and disposal of 14 million Am-241 ICSD's each containing 3 uCi of Am-241 will result in a collective total body dose of about 1100 person-rem. The useful life is assumed to be ten years and disposal is by either sanitary landfill or incineration. Fourteen million ICSD's will service about 21 million people. Analysis also shows the risk to the exposed population is less than one fatal cancer (0.1). The normally-occurring cancer mortality rate for the total population of the U. S. is about 370,000 per year (with estimates for 1980 as high as 405,000 per year), or about 35,000 per year for a group of 21 million people.

A comparison of these numbers illustrates the relatively small risk involved in using an Am-241 ICSD. It should be noted the 370,000 cancer deaths per year are actual deaths while the 0.1 fatal cancer over ten years of ICSD use was calculated using conservative assumptions. The ratio of the potential lives saved to the possible fatal cancers due to use of ICSD's ranges from 15,000 to 51,000.

Analysis of potential accidents with Am-241 ICSD's showed the dose commitments received by maximally exposed individuals to be significantly within the safety criteria standards required by the NRC.

Comparative analyses based upon cost, risk, and benefits found present day Am-241 ICSD's to be preferable to ICSD's with other radionuclides and preferable to present day photoelectric detectors. The best available unit for fire protection was found to be the recently marketed combination detector. The cost of a combination unit is considerably more than the Am-241 ICSD, and data is lacking on how much more sensitive it is. Therefore, the cost-effectiveness of combination units compared to Am-241 ICSD's is unknown.

11.3 REFERENCES FOR SECTION 11

* Belanger, R., D. W. Buckley and J. B. Swenson, "Environmental Assessment of Ionization Chamber Smoke Detectors Containing Am-241," NUREG/CR-1156, Science Applications, Inc., La Jolla, CA, 1979.

12. OTHER PRODUCTS

Various types of consumer products containing deliberately incorporated radionuclides are in everyday use throughout the world. The preceding ten sections of this report dealt with the most common and most widely distributed of the consumer products in the U. S. Other products have existed in the past and some products exist that have not already been discussed. The following discussion focuses on these other consumer products.

12.1 MISCELLANEOUS RADIOLUMINOUS PRODUCTS

A variety of radioluminous products have existed at one time or another. Included in this category are compasses, light switches, gun sights, marine instruments, automobile lock illuminators, automobile shift quadrants, instrument push buttons and switches, simple markers, map readers, thermostat dials, and telephone dial lights. At present, only the compasses and marine instruments appear to be distributed in the U. S. One company manufactures marine compasses with byproduct material, but the number distributed is very small and the compasses are not considered to be consumer products. The United Nations Scientific Committee on the Effects of Atomic Radiation reported on sources of ionizing radiation (UNSCEAR 77). Included in the report is a section on radiation exposures from consumer products. Numerous other radioluminous products are listed in the report, however, none are available to the consumer in the U. S. Table 12.1 indicates maximum allowable values for radionuclides and external radiation for miscellaneous radioluminous products in the U. S. Obviously, all the listed products are not consumer products (aircraft luminous safety devices, for example).

Table 12.1. Maximum Allowable Values for Radionuclides and External Radiation for Miscellaneous Radioluminous Products

Item	H-3 (mCi)	Pm-147 (uCi)	Other Nuclides (uCi)	External Dose Rate (mrad/hr)
Automobile Lock Illuminator	15	2	NS(1)	1 @ 1 cm
Automobile Shift Quadrant	25	NS	NS	NS
Marine Compass	750(2)	NS	NS	NS
Other Marine Instruments	250(2)	NS	NS	NS
Thermostat Dials and Pointers	25	NS	NS	NS
Aircraft Luminous Safety Devices	10,000	300,000	NS	0.5 @ 10 cm

(1) NS = Not specified

(2) Tritium Gas

12.2 PIEZOELECTRIC CERAMICS

Piezoelectric ceramics are used in a variety of consumer products that require an electromechanical coupling device. Products that contain piezoelectric ceramic devices include pilot light ignitors in gas heaters, cigarette lighters, flash-bulb ignitors, ultrasonic cleaning equipment, sonar projectors, phonographic pick-up cartridges, bell clappers, microphones, ultrasonic components, strain gauges, accelerometers, precipitators, intermediate-frequency bandpass filters in components, such as automobile FM radios, audible signal components in watches, and many others.

Piezoelectric ceramics essentially generate electricity when subjected to mechanical stress. The same materials exhibit the reverse effect in that they change form under influence of an electric field. Piezoelectric ceramic materials are hard, chemically inert, and immune to humidity or any other atmospheric condition. Because of their ceramic nature, the piezoelectric materials can be given almost any shape or size. By slightly changing the chemical composition of the materials it is possible to emphasize one or more specific properties. Hence, the long list of uses given above is possible.

A commonly used substance for making piezoelectric ceramics is lead zirconate titanate. Various additives are used to make piezoelectric ceramics. One additive used is uranium oxide. A typical formulation is 65 wt percent PbO, 33 wt percent $ZrO_2 + TiO_2$, and 2 wt percent UO_2. A typical device may contain about 2 mg of uranium.

Because of the lack of information on piezoelectric ceramics containing uranium, an assessment of their impact was not performed. Communication with O'Donnell at ORNL indicates about ten million piezoelectric devices may be produced annually. However, the amount of uranium used is unknown at this time.

12.3 MILITARY SURPLUS COMMODITIES

Military surplus commodities containing radioactive materials, primarily radium, have been released by Department of Defense facilities through the General Services Administration and the Defense Supply Agency to commercial retail channels of trade in the past. At the time of release, the commodities became available for consumer use and may still be used by or possessed by consumers.

The main radioactive material used was radium and its use by the military services has taken place for over 60 years. The primary uses included illumination of gunsights, watch dials, and compass cards and other similar functions. It is estimated that several hundred grams of radium were used by the military during World War II (Halperin 66).

In September of 1964, the Bureau of Radiological Health of the California State Department of Public Health undertook a study to determine the types of surplus military articles containing radioactive material available in California. Of 37 selected stores surveyed, 20 offerred items containing radioactive material for sale. Dose rates were measured with a portable instrument and readings as high as 5 mr/hr were recorded at the faces of bins containing radioactive material. Conclusions of the survey indicated the radioactivity appeared to mainly be limited to aircraft panel instruments, switches, knobs and buttons, electron tubes, dials, gauges and meters, portable radio transmitters, ship clocks, and pocket compasses. The electron tubes contained Cobalt-60 (Co-60) for the most part.

Since the time of the above survey, the release of commodities containing radioactive materials by Department of Defense facilities has been under more control, requiring that potential purchasers or recipients be advised of the radioisotopic content and activity. However, past released commodities still exist in the consumer sector and for purposes here, such products are considered consumer products.

12.4 EDUCATIONAL PRODUCTS

Numerous devices containing radioactive materials have been available in the past for use in educational institutions. Some of the products have been available to the general consumer, but the quantities have been very small. Included were alpha ray sources containing Ra-226 or lead-210 (Pb-210), cloud chambers with Ra-226 or Pb-210, spinthariscopes with Ra-226 or Pb-210, radioactivity demonstrator rate meter systems with Ra-226, thallium-204 (Tl-204), or Co-60, and various other apparatus.

Of the identified products, spinthariscopes were also available in home "chemistry sets" which could be purchased in retail stores. The spinthariscope was used to demonstrate scintillation. The device contained a radioactive material such as Ra-226 in a mixture of scintillation material on a small screen.

An eyepiece with a magnifying glass was located next to the screen. By viewing through the eyepiece, scintillation could be observed.

12.5 PHOTOGRAPHIC TONER

Uranium nitrate, UNO_3, has been used in photographic toner for photographic prints. The uranium nitrate was used in a process as a constituent of a mordant which was poured or brushed over the surface of black and white prints. The mordant was then combined with dyes to form an insoluble compound that gave the black and white prints the appearance of being color prints. The uranium toner by itself would give the prints a sepia coloration. Although prior to the 1950's this was a method employed by both amateur and professional photographers, it probably was not used extensively since it was a difficult and involved process. The chemicals used in the process were sold in both pre-mixed liquid form and dry chemical form. Restrictions imposed by the Atomic Energy Commission (AEC) on the use of uranium in toner caused the product to be "discontinued till a later date" in 1947. It is mentioned in this report since old prints using the process still exist and because the process could still be employed by the individual photographer.

An apparent misconception still exists concerning the availability of photographic toners containing uranium nitrate. A recent article published by the Center for Occupational Hazards (Rossol 80) is an example of this. In this article it was indicated that the uranium nitrate product was still available. Part of this misconception stems from the continued publication of certain annual handbooks that persist in publishing information that was originally developed many years ago but has since become outdated. To verify the nonavailability of uranium nitrate toner, major photographic chemical manufacturing houses were contacted and all have indicated that the uranium nitrate toner does not exist. In most cases, the spokesperson for the manufacturers had not even heard of such a product.

12.6 ION GENERATORS

Part 31 of the Code of Federal Regulations under Title 10, Energy, addresses the issuance of general licenses for ion generating tubes containing byproduct material. Since a number of consumer products are presently marketed as "ion generators," the possibility of the use of radioactive materials in these products was investigated. The products were found to be solid state, high-voltage devices that produce negatively charged ions. Manufacturers and distributors who were contacted concerning this consumer product, confirmed that the ion generators are electronic in principle and contain no intentionally added radioactive materials.

12.7 URANIUM IN POTTERY PRODUCTS AND GLASS ENAMEL FRITS

There exists in the United States a considerably large number of supply houses that offer for sale the materials used in porcelain enamel and pottery products. Table 12.2 illustrates price and size-packaging ranges for uranium oxides from the price lists of four such distributors.

A check with several suppliers indicated that the purchase of uranium oxide in pound lots or greater is a rare, if ever, event. In fact, distributor 4 in Table 12.2 was found to be no longer supplying uranium oxide of any lot size. A spokesperson for this distributor indicated no knowledge of how the oxide would be used by an artist and indicated that they never received a request for the items. A major drawback to the sale of uranium oxide appears to be cost. This was the indication of all suppliers contacted.

The uranium oxide used by pottery artists is purchased in a powdered form. For the enamel-ware artist, the uranium oxide is purchased either as a powder or in the form of glass frits and subsequently ground into a powder of about 80 mesh. In any case, the uranium oxide takes on the form of a powder and is thus capable of becoming, to some extent, airborne.

There exists concern among some people (Rossol 80) that the airborne powdered uranium oxide may pose a health hazard of greater significance than that of other powders normally used by persons working in these artistic endeavors. Specifically, it has been pointed out that errors in handling the powdered product could pose a health hazard to children or young people incapable of understanding the significance of such an event.

Table 12.2. Retail Costs of Uranium Oxide.

	Packaging and Price					
	Price per Package ($)			Price per lb ($)		
	Ounce	¼ lb	½ lb	1 lb	10 lb	25 lb
Distributor 1	-	-	25.00	43.00	-	-
Distributor 2	-	3.50	6.50	12.00	-	-
Distributor 3	1.25	-	-	16.00	-	-
Distributor 4	-	-	-	18.00	17.00	26.00

It is extremely difficult to make a determination on the health hazard questions in terms of exposure of artists to uranium oxide. Suppliers of artists materials, art instructors, and artists have all stated that little, if any, uranium of any kind is used in their industry. The art department of a major university indicated they have never used uranium because of the radioactive aspect of the material. A spokesperson for an enamelists guild stated that they knew of no one currently supplying uranium oxide materials. The purchasing agent for one of the ten largest public school districts in the U. S. has indicated that all materials of this nature must be certified and labeled non-toxic before a school could use them and that, in at least the last ten years, no known use of uranium oxide by that district's schools had occurred.

Based on the inability of detecting any significant amount of uranium oxide in the enamel or pottery art industry, no risk analysis is performed in this section. However, it would behoove any primary or secondary school art instructor to withhold this material and any material that could be a health hazard until the student is of such an age that an educated choice could be made as to whether or not this material should be used. The apparent danger, if any, can be overcome by proper handling of the material. This safety factor should be well within the reach of an adult artist. It is the choice of the adult artist to determine whether or not to use the material.

12.8 SUMMARY

As can be seen, numerous products containing radioactive material have been and are being produced in the U. S. This short discussion of "other products" acts to simply give the reader a perspective on the range of products available. Other products likely exist, but they will remain relatively obscure unless their impact via sales increases in the future.

12.9 REFERENCES FOR SECTION 12

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APPENDIX A - DOSE CALCULATIONAL METHODS

A.1 EXTERNAL DOSE

Many of the radionuclides used in consumer products emit levels of gamma radiation which, while low, are of a magnitude sufficient to warrant careful assessment. This section describes the general method used to calculate total body and organ doses from external sources of gamma radiation.

The basic equation used for estimating the external dose rate from a product containing a point source of radionuclide i emitting gamma radiation is:

$$D_{e,T} = \frac{(\Gamma_i A_i F_s) DCF'_{i,T}}{d^2}$$

where

$D_{e,T}$ = total body or organ dose rate (rem/hr)

Γ_i = total gamma exposure rate constant for radionuclide i ($R\text{-}m^2/hr\text{-}Ci$)

A_i = activity of radionuclide i per product (Ci)

F_s = shielding or other modifying factor (dimensionless)

$DCF'_{i,T}$ = external dose conversion factor for absorption of mean weighted photon energy i in tissue T (rem/R)

d = source-to-subject distance (m)

The mean weighted photon energy, E_i for radionuclides emitting multiple gamma rays is determined by:

$$E_i = \sum_{j=1}^{n} \frac{E_j Y_j}{n}$$

Table A.1 Gamma-Ray Exposure Constants for Point Sources of Selected Radionuclides

Nuclide	Γ (R/hr @ 1 m/Ci)
Co-60	1.3E+0
Kr-85	1.3E-3
Cs-137	3.3E-1
Po-210	4.9E-6
Ra-226+D	8.3E-1
Th-232+D	1.5E+0(1)
U-238+D	4.0E-2
Am-241	1.2E-2

(1) Value given for equilibrium condition or at least 100 years after thorium recovery. For 20-year condition after recovery the value is 1.3E+0.

where

E_j = energy of photon j (MeV)

Y_j = yield of photon j (photons/dis)

n = total number of photons per disintegration

The use of the mean weighted photon energy is not an exact and precise method for estimating the external dose conversion factor if the gamma rays from the radionuclide of interest span a very broad energy range. For the radionuclides of interest to this study, however, this method represents a sufficiently accurate approximation for obtaining DCF values.

Total gamma exposure rate constants for radionuclides of interest are listed in Table A.1. These values represent gamma exposure rates in roentgens per hour at a distance of one meter from an unshielded one-curie point source of radioactivity.

O'Brien (1978) has calculated total body and organ dose conversion factors for males and females exposed to an isotropic radiation field. The results of his calculations are used in these assessments and are presented in Figures A.1 and A.2. Genetic doses are estimated by use of the dose factors for testes and ovaries depicted in Figure A.3.

Exposure rates from extended sources, such as lines or discs, are assessed according to the methods described by Rockwell (1956). In many cases, the composition of the product is such that calculation of the exposure rate is a complex and difficult task. Such cases are discussed in the sections dealing with specific products. Whenever available, applicable measurement data will be used in place of calculated exposure rate values. The general method for assessing exposure rates from extended or non-point sources is presented below.

The geometry for calculation of the photon flux from a line source is given in Figure A.4. The flux is calculated by:

$$\phi = \frac{S_L}{4\pi a} \left(\int_0^{\theta_2} e^{-ut \sec\theta} \, d\theta - \int_0^{\theta_1} e^{-ut \sec\theta} \, d\theta \right)$$

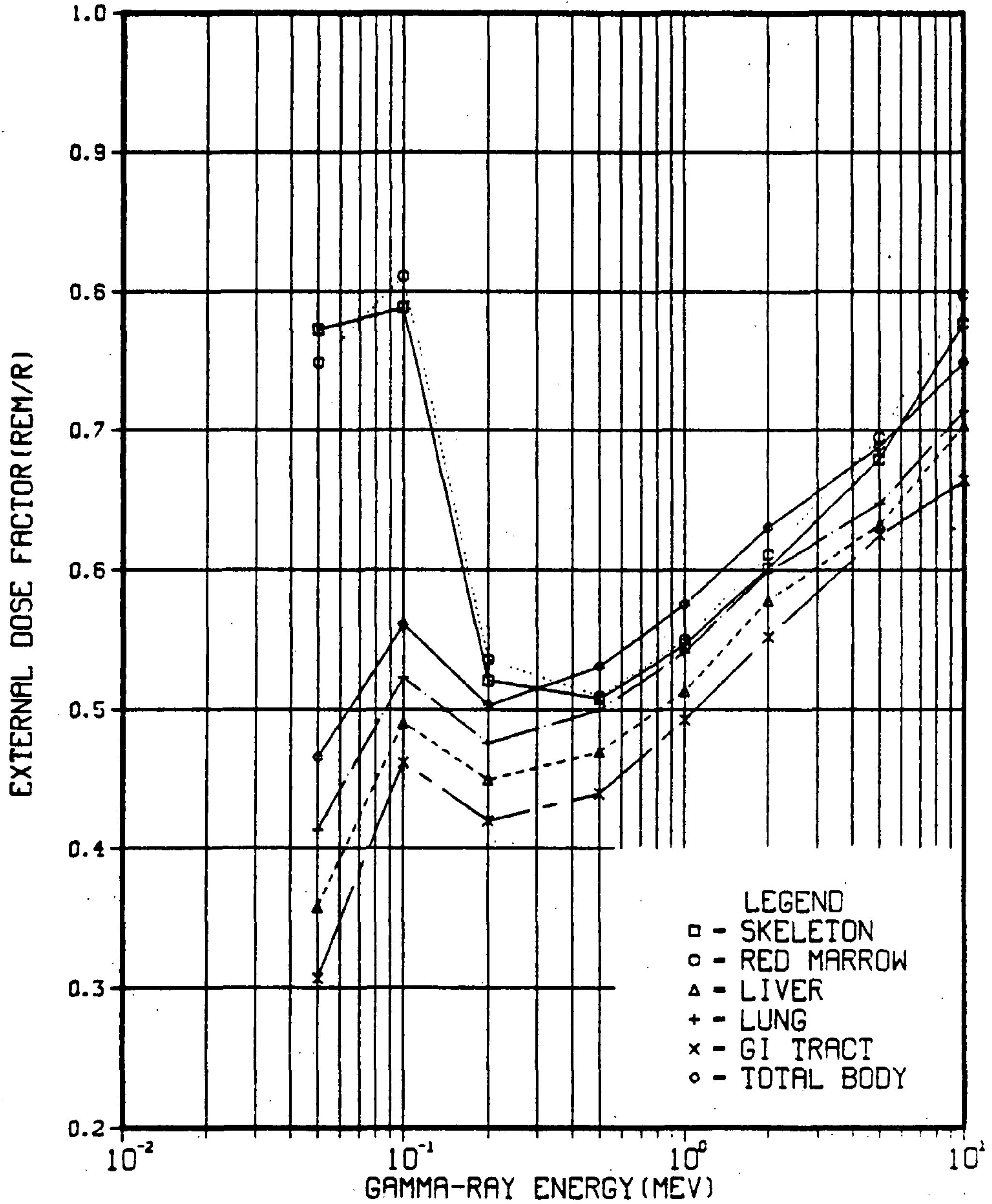

Figure A.1. Dose Factors for Male Organs Exposed to External Gamma Radiation.

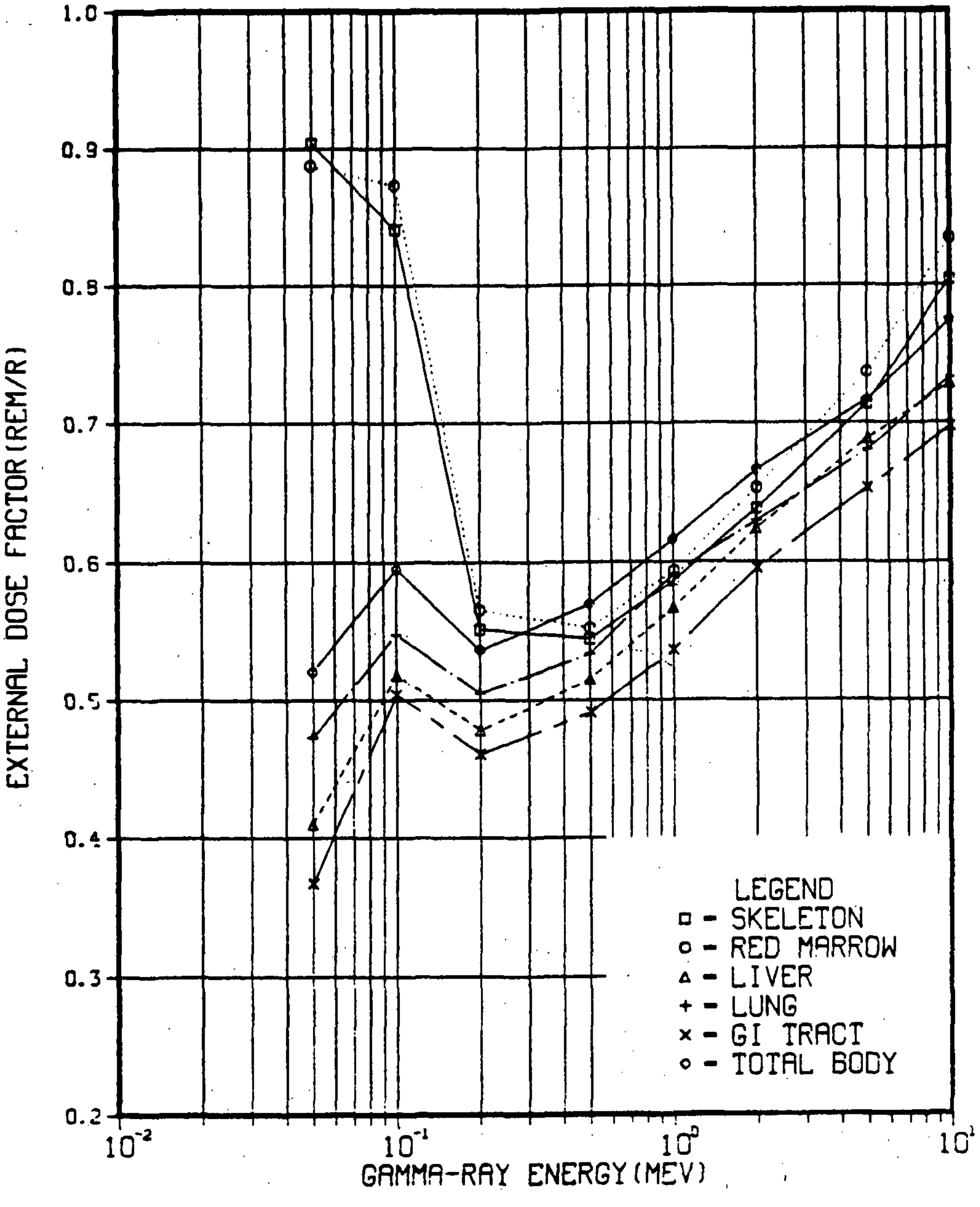

Figure A.2. Dose Factors for Female Organs Exposed to External Gamma Radiation.

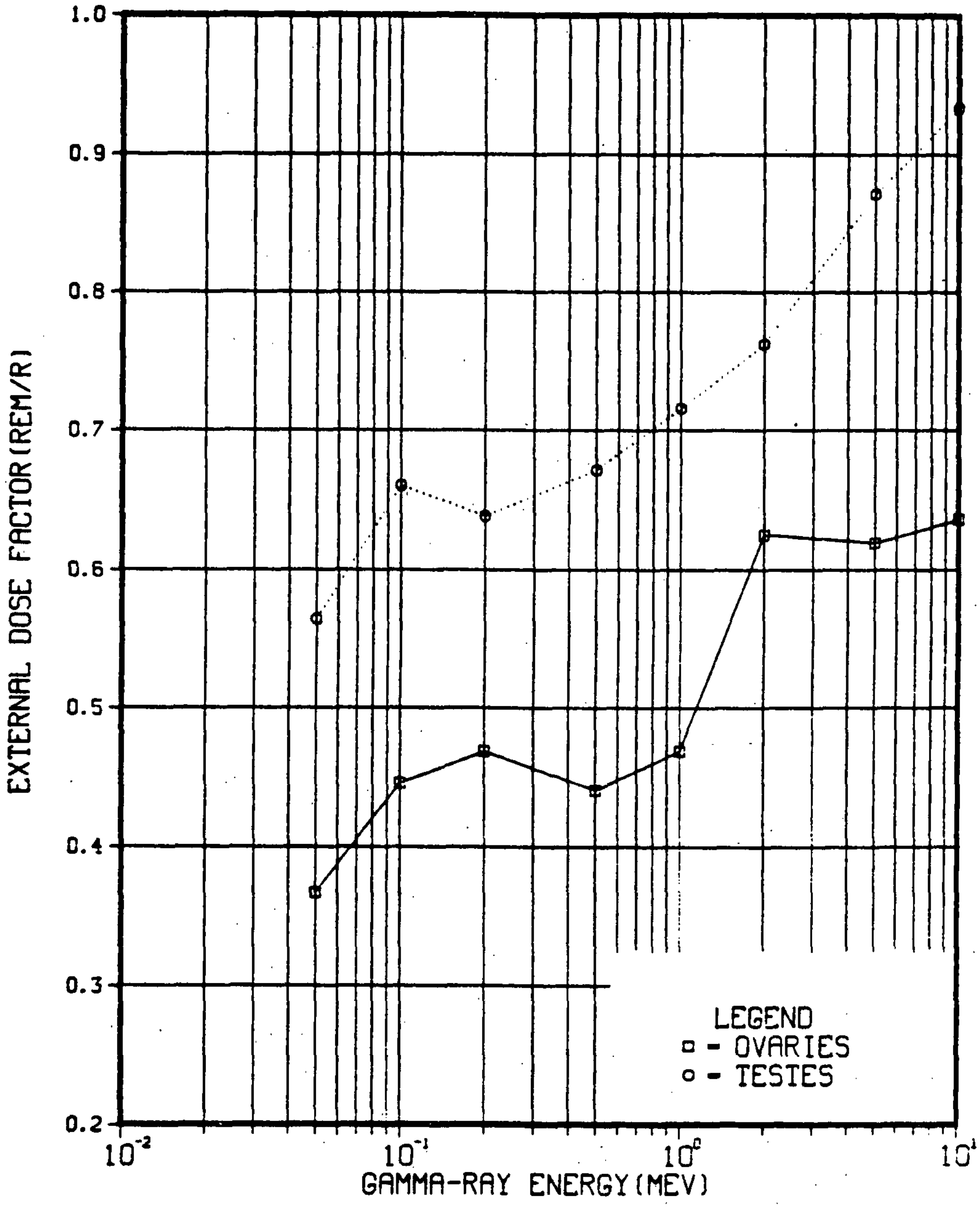

Figure A.3. Dose Factors for Human Sex Organs Exposed to External Gamma Radiation.

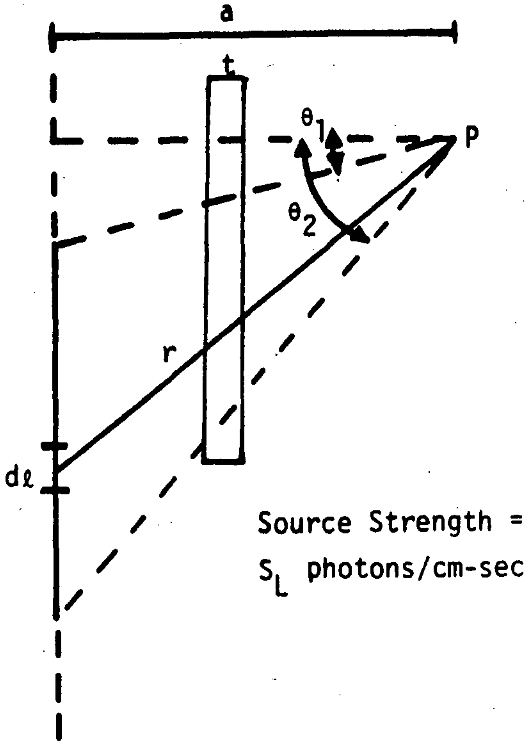

Figure A.4. Geometry for Calculation of Photon Flux from a Line Source.

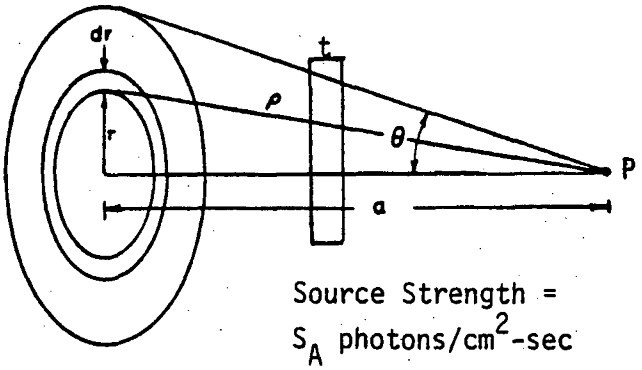

Figure A.5. Geometry for Calculation of Photon Flux from a Disc Source.

where

ϕ = photon flux (photons/cm^2-sec)

S_L = source strength (photons/cm-sec)

u = linear attenuation coefficient of absorber (cm^{-1})

t = thickness of absorber (cm)

The solution can be expressed in terms of the Sievert function ($F_{\theta,ut}$):

$$\phi = \frac{S_L}{4\pi a}\left(F_{\theta_2,ut} - F_{\theta_1,ut}\right)$$

The geometry for calculation of the flux from a disc source is given in Figure A.5. The flux is calculated by:

$$\phi = \frac{S_A}{2}\left[E_1(ut) - E_1(ut \sec \theta)\right]$$

Values for the functions $F_{\theta,ut}$ and $E_1(ut)$ are given in the referenced work of Rockwell.

Exposure rates are calculated from photon fluences according to the following expression:

$$R = K\phi E(u_{en}/p)$$

where

R = exposure rate (R/hr)

K = correction factor for unit conversions (6.57x10^{-5} R-s-g/MeV-hr)

ϕ = photon flux (γ/cm^2-s)

E = gamma ray energy (MeV/γ)

u_{en}/ρ = mass energy absorption coefficient (cm^2/g)

The total body or organ dose rate is calculated from the exposure rate by:

$$D_{e,T} = (DCF'_{i,T})\ R$$

where

$D_{e,T}$ = external dose rate to tissue T (rem/hr)

$DCF'_{i,T}$ = external dose conversion factor for mean weighted photon energy i and tissue T (rem/R)

R = exposure rate (R/hr)

A.2 INTERNAL DOSE

Unless specified otherwise, doses to the total body and to specific organs as a result of radionuclide inhalation or ingestion were calculated according to the following formula:

$$D_{i,T} = (DCF_{i,T})(I_i)$$

where

$D_{i,T}$ = dose-equivalent commitment averaged throughout a tissue T over 50 years following an intake of radionuclide i (rem)

$DCF_{i,T}$ = dose commitment factor relating dose to tissue T per unit intake of radionuclide i (rem/uCi)

I_i = amount of intake of radionuclide i (uCi)

The dose commitment factors used in most cases were those calculated by Hoenes and Soldat (1977) according to the methodology recommended in ICRP Publication 2 (ICRP 59). Where necessary, corrections were made in the effective energy values to account for the ingrowth of radioactive daughters (indicated by the "+D" nomenclature). Since recent recommendations by the NCRP and ICRP advocate an increase in the quality factor for alpha radiation by a factor of two, corrections have been made in the dose commitment factors for alpha-emitting radionuclides to account for this change. Values for the inhalation and ingestion dose conversion factors used in this report are listed in Tables A.2 and A.3.

A.2.1 Tritium

Tritium can enter the human body by inhalation, ingestion, or absorption. Elemental tritium gas (HT or T_2) or tritiated water (HTO or T_2O) can be absorbed by the respiratory tract. In general, 98 to 99 percent of inhaled tritiated water is absorbed, while the value for absorption of elemental tritium is orders of magnitude lower. In fact, submersion in tritium gas is limited solely by consideration of the dose-equivalent rate to the lung from the tritium gas contained within it (ICRP 79). Tritiated water which is absorbed through the skin, lungs, or gut is rapidly distributed throughout the body via the blood. Maximum blood concentrations are reached within a few minutes of absorption through the lung or gut, and within two hours of absorption through the skin (NCRP 79).

Reasonably good data concerning human retention and elimination of tritium are available. The range of observed biological half-lives of tritium in accidentally contaminated individuals has been found to be between 2.4 and 18 days. The NCRP (NCRP 59) has adopted a value of 12 days for practical application of the biological half-life of tritium, while the ICRP currently uses a value of ten days (ICRP 79). This assessment uses the former value (12 days) since it is the more conservative of the two. Thus, the fraction of tritium taken into the body as water which is retained at t days after the intake is described by

$$R(t) = \exp(-0.693t/12)$$

An initial intake of 1 mCi will result in a maximum concentration in body water of 0.023 uCi/ml. However, since tissue exclusive of bone and fat contains about 70 percent water (compared to 60 percent for entire reference man), the tritiated water concentration in tissue will be about 0.016 uCi/g per mCi taken in. Tritiated water is excreted in the urine, sweat, breath, and feces. It has been shown that tritiated water in urine has the same specific activity as that of blood (Feinendegen 67, Seelentag 73, Pinson 57). Thus, if

Table A.2. Fifty-year Inhalation Dose Conversion Factors for Radionuclides of Interest

Radionuclide	Dose Conversion Factor (rem/uCi)				
	Total Body	Liver	Bone	Lungs	G. I. Tract
H-3	8.0E-5	8.0E-5	8.0E-5	8.0E-5	8.0E-5
C-14	7.0E-4	7.0E-4	1.1E-3	7.0E-4	7.0E-4
Co-60	1.9E-3	1.4E-3	1.9E-3(1)	7.5E-1	3.6E-2
Ni-63	1.8E-3	3.9E-3	5.4E-2	2.2E-2	1.7E-3
Kr-85	0.0E+0	0.0E+0	0.0E+0	2.4E-6	0.0E+0
Cs-137+D	5.4E-2	7.8E-2	6.0E-2	9.4E-3	1.1E-3
Pm-147	3.2E-3	7.9E-3	8.4E-2	6.6E-2	5.5E-3
Po-210	1.9E-1	1.7E+0	7.9E-1	6.3E+1	8.4E-2
Pb-212(2)	1.1E-2	3.6E-2	1.5E-1	3.3E-1	7.0E-3
Bi-212(2)	1.9E-3	1.2E-2	2.8E-3	6.4E-2	7.2E-3
Ra-226+D	1.8E+2	4.8E-3	2.5E+2	2.4E+2	5.9E-1
Ra-228+D	4.8E+1	1.2E-3	4.4E+1	1.6E+2	5.0E-2
Th-228+D	1.4E+1	6.8E+0	4.0E+2	2.0E+3	7.0E-1
Th-232+D	1.8E+2	2.2E+2	5.2E+3	1.2E+3	6.4E-2
U-238+D	1.1E+0	1.1E+0(1)	1.9E+1	9.2E+1	6.8E-2
Am-241	1.3E+2	7.2E+2	2.0E+3	1.2E+2	9.2E-2

(1) No data; DCF for this organ assumed to be equal to that for total body.

(2) Data taken from Killough (1976).

Table A.3. Fifty-year Ingestion Dose Conversion Factors for Radionuclides of Interest

Radionuclide	Dose Conversion Factor (rem/uCi)			
	Total Body	Liver	Bone	G. I. Tract
H-3	8.0E-5	8.0E-5	8.0E-5	8.0E-5
C-14	7.0E-4	7.0E-4	1.1E-3	7.0E-4
Co-60	4.7E-3	2.1E-3	4.7E-3	4.0E-2
Ni-63	4.4E-3	9.0E-3	1.3E-1	1.9E-3
Kr-85	0.0E+0	0.0E+0	0.0E+0	0.0E+0
Cs-137 +D	7.1E-2	1.1E-1	8.0E-2	2.1E-3
Pm-147	2.9E-6	7.1E-6	7.5E-5	8.9E-3
Po-210(1)	1.7E-1	1.5E+0	7.2E-1	1.2E-1
Ra-228+D	1.2E+2	3.1E-3	1.1E+2	5.6E-2
Th-228+D	3.4E-2	1.7E-2	9.9E-1	1.1E+0
Ra-226+D	4.4E+2	1.1E-2	6.0E+2	6.6E-1
Th-232+D	3.0E-1	2.0E-1	4.6E+0	1.0E-1
U-238+D	9.0E-2	9.0E-2	1.5E+0	1.1E-1
Am-241	1.1E-1	5.8E-1	1.6E+0	1.5E-1

(1) For Po-210, the kidneys conversion factor is 5.0E+0 rem/uCi

the concentration is relatively constant over the entire year, the annual tissue dose can be calculated from the measured urine concentration as follows:

$$D = C\,D'\,K_1\,K_2$$

where

D = annual tissue dose (mrad/yr)

C = urine concentration (uCi/ml)

D' = tissue dose rate (12.14 mrad/hr per uCi/g)

K_1 = correction factor to account for the fraction of water in soft tissue, the fraction of H-3 in organic constituents relative to water, and the fraction of organic solids in soft tissue (0.76 ml/g)

K_2 = hours per year (8760 hours)

A continuous urine concentration of 1 uCi/ml will result in an annual tissue dose of:

$$(1\ \text{uCi/ml})(12.14\ \frac{\text{mrad/hr}}{\text{uCi/g}})(0.76\ \text{ml/g})(8760\ \text{hr/yr})$$

$$= 8.1\text{x}10^4\ \text{mrad/yr}$$

Of course, in most cases, the intake of tritium and the resultant urine concentrations will not remain constant. Figure A.6 illustrates the change in urine concentration with time after a single intake of 1 mCi. If exposure to tritium is not relatively constant, the elapsed time between acute intake and urine sample collection must be known in order to accurately assess the dose. Since urine concentrations decrease with time after intake, dose assessments made using these data would tend to underestimate the true dose. Moghissi, et al. (1970) has suggested using a value of 0.1 mrad/yr per nCi/l ($1.0\text{x}10^5$ mrad/yr per uCi/ml), a value which would compensate somewhat for the tendency to underestimate the dose.

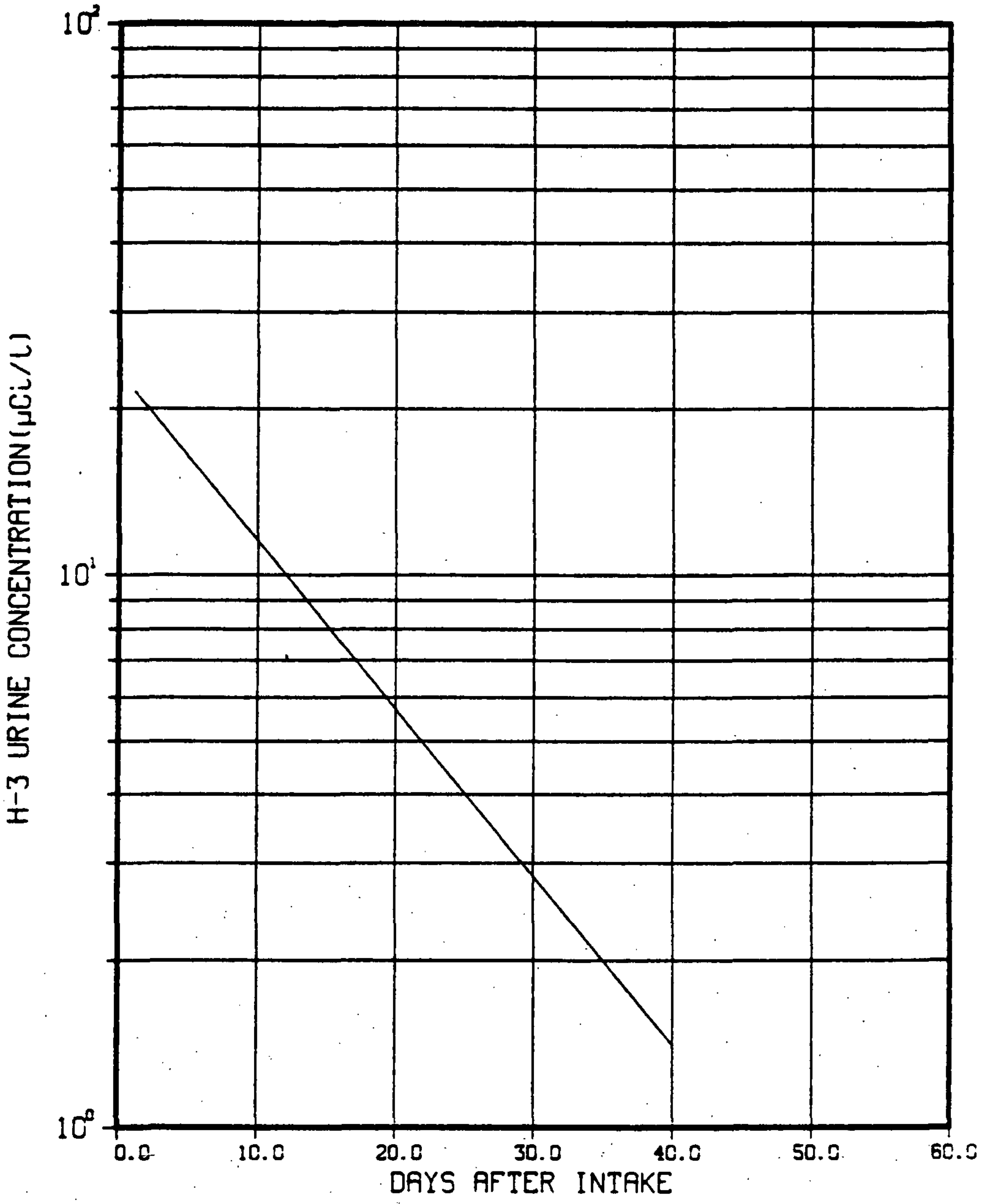

Figure A.6. H-3 Concentration in Urine as a Function of Time After Intake of 1 mCi.

For assessing the dose to individuals for single intakes, or where excreta data are not available, the following calculation is used:

$$D = \frac{K\,A\,E\,T_e}{m}$$

where

D = total dose (over all time) to soft tissue (rads)

K = correction factor for unit conversions (73.8 rads-g-dis/uCi-MeV-d)

A = activity taken in (uCi)

E = average energy per disintegration (0.0057 MeV/dis)

T_e = effective half-life (12 d)

m = mass of soft tissue (6.3×10^4 g)

The total dose to soft tissue for each mCi of tritium taken in is:

$$D = \frac{(73.8\ \frac{\text{rad-g-dis}}{\text{uCi-MeV-d}})(1000\ \text{uCi})(0.0057\ \text{MeV/dis})(12\ \text{d})}{(6.3 \times 10^4\ \text{g})}$$

$$= 8.0 \times 10^{-2}\ \text{rads or about 80 mrad}$$

This compares reasonably well with recently recommended values of 81 mrad/mCi (NCRP 79) and 62 mrad/mCi (ICRP 79), and, assuming a quality factor (Q) of one, is somewhat lower than the dose conversion factor of 158 mrem/mCi used by the NRC (Hoenes 77). One reason that the NRC dose factor is higher is the use of 1.7 for the quality factor of the 18 keV beta particles emitted by H-3. Both the ICRP and the NCRP currently recommend a value of one for Q (ICRP 77, NCRP 79). A discussion of this aspect is included in Appendix B. In this report, a value of unity will be used, so the dose-equivalent conversion factor is:

$$(80\ \text{mrad/mCi})(1\ \text{mrem/mrad}) = 80\ \text{mrem/mCi}$$

A.2.2 Carbon - 14

The metabolism of C-14 in the human body varies considerably as a function of the compound into which it is incorporated. Most of the available data relate to compounds such as bicarbonate-CO_2, glycine, and acetate. It can be concluded from the available data that in humans and animals the major portion of radioactive carbon dioxide will be eliminated by exhalation three hours after it has entered the blood. However, there is evidence that up to 30 percent of the carbon dioxide may be deposited in bone where it is eliminated with a half-life of 0.4 days (ICRP 67).

For a transportable C-14 (carbonate) intake of 1 mCi, the dose equivalent to bone (m = 7,000g) can be: calculated by

$$D\text{ (rem)} = \frac{(73.8\ \frac{\text{rad-g-dis}}{\text{uCi-MeV-d}})(1000\text{ uCi})(.054\text{ MeV/dis})(0.4\text{ d})(5\text{ rem/rad})}{(7.0\times 10^{3}\text{ g})}$$

$$= 1.1\text{ rads}$$

To assess the total body dose for 1 mCi of transportable C-14 taken into the body ($m = 7.0\times 10^{4}$ g) the following dose conversion factor is used:

$$D = (73.8\ \frac{\text{rad-g-dis}}{\text{uCi-MeV-d}})\ \frac{(1000\text{ uCi})(0.054\text{ MeV/dis})(12.5\text{ d})}{(7.0\times 10^{4})}$$

$$= 0.7\text{ rads}$$

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APPENDIX B - RADIOLOGICAL HEALTH EFFECTS

B.1 INTRODUCTION

The estimation of serious health effect risks resulting from exposure to low levels of ionizing radiation is made difficult by a lack of specific data. This limitation persists despite the fact that in recent years much effort has been expended in an attempt to quantitatively link exposure and health effects. The purpose of this appendix is to briefly summarize the state of knowledge regarding internal and external radiation effects. This knowledge provides the technical basis used in this report to estimate the radiological health impact, in terms of cancer mortality and serious genetic diseases, which would result from the manufacture, use, and disposal of consumer products.

B.2 CARCINOGENIC EFFECTS

Much literature in recent years has concerned itself with estimating the extent of cancer incidence and mortality resulting from low levels of ionizing radiation. These studies have drawn primarily from observations of carcinogenic effects following human exposure to comparatively high levels of radiation which were delivered at high dose rates. The externally exposed population groups which are the most widely studied are the atomic bomb survivors of Hiroshima and Nagasaki, a group of British patients who were treated by spinal irradiation for ankylosing spondylitis, and a group of Israeli children who received radiation for treatment of thyroid disorders. In addition, attention has been recently directed at relating cancer mortality to the relatively low levels of exposure incurred by workers at nuclear facilities, such as the Hanford Works (Mancuso 77, Sanders 78). Results from these studies, however, are of limited usefulness in quantitatively relating dose and effects. Two groups of people that have been subjected to relatively large amounts of internal contamination (and have been thoroughly studied as a result) are uranium miners and the population of women who used radium paint in making luminous timepieces.

Excessive cancer death rates observed in externally exposed populations are primarily due to leukemia and cancers of the breast, lung, and gastrointestinal tract, a pattern which follows fairly closely the estimated normally-occurring cancer mortality rates (Table B.1). It should be emphasized, however, that the increased mortality was observed at doses and dose rates which are many orders of magnitude greater than those encountered in this report. Thus, the estimation of risk at very low total doses and dose rates must be extrapolated, and this is usually performed on the basis of a linear, non-threshold relationship, a practice which has been seriously questioned by some organizations. For example, the National Council on Radiation Protection and Measurements (NCRP 75) has commented that:

> "The NCRP continues to hold the view that risk estimates for radiogenic cancers at low doses and low dose rates derived on the basis of linear (proportional) extrapolation from the rising portions of the dose-incidence curves at high doses and dose rates. . . cannot be expected to provide realistic estimates of the actual risks from low level, low-LET radiations, and have such a high probability of overestimating the actual risk as to be of only marginal value, if any, for purposes of realistic risk-benefit evaluation."

More recently, a "Dissenting Report" of the National Academy of Sciences (NAS) Subcommittee on Somatic Effects concluded that for low-LET radiation at doses of between a few and about 100 rad the dose-effect relation for total carcinogenic effect is very likely to be linear in this range (NAS 79). The dissenting group went on to say:

> "We conclude furthermore that risk estimates for whole body irradiation that are based on individual organ risk estimation in BEIR-III are overestimates of incidence at low doses. We do not believe that there is adequate information to determine accurately the magnitude of the error. It seems likely, however, that it is as much as an order of magnitude and possibly more."

Table B.1. Spontaneous Cancer Mortality Rates for the United States.(1)

Cancer Type	Mortality (Per 10^6 Persons Per Year)
Leukemia	71
Respiratory system	483
Stomach	64
Rest of GI tract	278
Pancreas	95
Breast	163
Bone	8
Thyroid	5
All others	674
Total	1841

(1) American Cancer Society, 1979

Conversely, Brown (1976, 1977) has summarized some recent human data which suggest that estimates of risk at low doses or dose rates are either the same as, or in some cases higher than, risk estimates derived from high doses and dose rates.

While the controversy regarding the estimation of effects of exposure to low levels of lightly-ionizing radiation is not likely to be settled in the near future, the currently practiced policy of most responsible organizations is to linearly extrapolate risk estimates derived at high dose and dose rates to lower levels of exposure, adding the proper caveats. This is done in the belief that it is more prudent to overestimate the risk than to underestimate it. Table B.2 summarizes the coefficients of risk for whole body irradiation estimated by various responsible organizations.

There seems to be more general agreement regarding the acceptance of a linear dose-effect relationship for high-LET radiation, such as that produced by alpha irradiation of internal organs from inhaled or ingested actinides. Lack of specific human data makes the task of estimating cancer mortality from inhaled or ingested alpha-emitting radionuclides difficult except for the case of radium ingestion. Estimates made in this study for internal deposition of alpha emitters are, therefore, highly uncertain. Table B.3 summarizes the currently available risk factors which can be applied to internal exposure to alpha-emitting radionuclides.

B.3 GENETIC EFFECTS

Genetic effects occur as a result of changes or alterations of the genetic material in germ cells. They may result from exposure to a variety of environmental factors including heat, ionizing and nonionizing radiation, and chemicals. These genetic alterations can arise from changes within one or more of of the purine or pyrimidine bases comprising the gene or from changes in the number or structure of chromosomes.

Genetic effects can be induced by exposure of the gonads (testes and ovaries) to either external radiation or to internally deposited radionuclides. Ingestion or inhalation of transuranics, for example, may result in the translocation of transuranics to the gonads. Once deposited in these organs, the radionuclides can cause a significant dose to sensitive stem cells. Tests involving plutonium deposition in various animal species have shown, however,

Table B.2. Comparison of Cancer Mortality Risk Estimated Associated with Collective Dose of 10^6 Organ-rem.

Cancer Type	BEIR 72	UNSCEAR 77	ICRP 77	NRC 75
Leukemia (red marrow)	20	20	20	28.4
Breast	45(1)	50	25	25.6
Lung	16-100(2)	25-50	20	22.2
Bone	2-17(2)	2-5	5	NS(3)
Liver	1-7(2)	10-15	NS	NS
Thyroid(4)	1.6-9.3(4)	10(5)	5(5)	4.3(6)
GI Tract(7)	30	10-15	NS	13.6
All Fatal Cancers	50-78(2),(8) 92-165(2,)(9)	100	100	121.6

(1) Derived from observed incidence in females by correcting for 50% cure rate and inclusion of males in population.

(2) Lower value represents absolute risk model with 30 year plateau following latent period; higher value represents relative risk model with lifetime plateau.

(3) NS-Not specified

(4) Values represent incidence of thyroid cancers per year per 10^6 person-rem in irradiated children. The risk for adults is somewhat lower.

(5) Mortality risk for general population.

(6) Incidence of thyroid cancers per year per 10^6 person-rem in irradiated children and adults.

(7) Includes stomach.

(8) Based on atomic bomb survivor data.

(9) Based on ankylosing spondylitis patient data.

Table B.3. Comparison of Cancer Mortality Risk Estimates Associated with Internally-Delivered Organ Dose of 10^6 Rem.

Cancer Type	BEIR 72	UNSCEAR 77	ICRP 77	Nenot 79	MRC 75	Mays 76
Lung Tumors	16 - 100(1)	25 - 50	20	20	25	20
Bone Tumors	2 - 17(1)	2 - 5	5	5	5	4
Liver Tumors	1 - 7(1)	10 - 15	NS(2)	5	20	10

(1) Lower value represents absolute risk model with 30-year plateau period; higher value represents relative risk model with lifetime plateau.

(2) NS = not specified

that the total deposition of inhaled plutonium in the gonads is low enough such that even allowing for preferential exposure of stem cells, the dose to these cells would not be expected to exceed the total body average (Thompson 77, Richmond 75). It is likely that the same can be said for other transuranics.

One conventional approach to assess the genetic health impact of ionizing radiation is to determine a "doubling dose," which is an estimate of the radiation dose that results in a twofold increase in the spontaneous mutation rate. Given the doubling dose, which is determined on the basis of experimental evidence, and assuming that the burden of human ill health attributable to normally occurring mutations is known (and is the same for radiation-induced mutations), one can directly estimate the detrimental genetic effects for any specific radiation dose. The 1977 UNSCEAR Report reviews and discusses the state of knowledge on this topic and arrives at a doubling dose of 100 rem, while the BEIR-III report ascribes to a range of 50 to 250 rem (UNSCEAR 77, NAS 79).

The types of mutagenic disorders induced by ionizing radiation can be placed in four catagories: (1) autosomal dominant disorders, (2) multifactional disorders, (3) chromosomal aberrations, and (4) spontaneous abortions. Autosomal dominant disorders are those caused by the alteration of a single gene, whereas multifactorial disorders have a more complex and ill-defined pattern of inheritance. Chromosomal aberrations include disorders characterized by changes in the number or structure of chromosomes present. Spontaneous abortions, while known to occur as a result of chromosomal effects (often so early in pregnancy as to go undetected), are generally not considered as a radiological health effect (USDOE 79).

Table B.4 summarizes the genetic effect risk estimates of various agencies and individuals. These estimates are for total effects to be experienced over all future generations.

B.4 HEALTH EFFECT RISK COEFFICIENTS

The coefficients of risk used for the purposes of estimating cancer mortality and genetic detriment are listed in Table B.5. These risk estimates were chosen since they represent a concensus of the best currentlv available literature values (as shown in previous tables) and are generally considered to be upper level estimates. Thus, although a lower level of risk - or even zero risk - cannot be entirely ruled out on the basis of currently available data, the

Table B.4. Genetic Disorder Risk Estimates (Cases per 10^6 Organ-Rem).

Type of Effect	BEIR 72	UNSCEAR 77	Nenot 79	Newcombe 75	NRC 75	MRC 75
Autosomal Dominant Disorders	50 - 500(1)	100	42	10	NS(2)	NS
Chromosomal Disorders	NS	40	7	NS	NS	NS
Multifactorial Disorders	10 - 1000	45	8	NS	NS	NS
Total	60 - 1500	185	57	10	95 - 165(3)	300

(1) Includes chromosomal aberrations

(2) NS = not specified

(3) Includes spontaneous abortions

Table B.5. Health Effect Risk Estimates Used in This Report.

Cancer	Tumor Deaths or Serious Genetic Defects Per 10^6 Organ-Rem
Liver	15
Bone	5
Leukemia (Red Marrow)	20
Lung	25
G.I. Tract	15
All Other Fatal(1) Cancers	70
Total	150
Genetic Defects(2)	200

(1) Primarily cancers of the breast and thyroid

(2) Total genetic detriment over all future generations

use of upper level estimates based on the linear, non-threshold hypothesis is considered to represent the most prudent approach to the low-level risk assessment performed in this report.

B.5 RADIONUCLIDE TOXICITY

Since certain radionuclides can have unique physical, chemical, or biological properties, special attention must be directed toward assessing the toxicological aspects of radionuclide deposition. This section summarizes the toxicology of some of the radionuclides which are capable of causing biological damage beyond that which would be expected solely on the basis of the radiation dose.

B.5.1 Tritium

Tritium is of special concern in that it, like Carbon-14, is a radionuclide which can become incorporated into the genetic material, deoxyribonucleic acid (DNA). Since damage to DNA is known to be the principal cause of radiation-induced cell death, mutations, and other harmful effects, the subject of radioactive decay within the nucleus, in which most of the cellular DNA is located, has important relevance to radiation protection standards and their bases, as well as to medical or clinical applications. The radiological health implications of DNA-incorporated radionuclide decay is the subject of a recent report by the National Council on Radiation Protection and Measurements (NCRP 79).

Damage to DNA from the decay of incorporated tritium can result either from the effects of ionizing radiation or from a number of events which occur in the immediate vicinity of the decaying atom. Examples of such local events are:

(1) Chemical change due to atomic number change (transmutation). Beta decay of tritium results in the formation of a He-3 atom at that site. Since helium is inert, it is a particular characteristic of tritium that its decay will always result in the breakage of the chemical bond between the daughter nuclide and its bonding partner (in this case, carbon).

(2) Production of an excited or ionized daughter. The electron which is emitted in beta decay usually has an energy which is high relative to that of the outer orbital electrons, and it leaves the atom before these electrons have time to adjust to the energy change. Thus, a residual amount of available energy remains after beta emission, leaving the new atom in an excited electronic state. Although the electronic excitation of the newly-formed helium atom can result in a transfer of up to 11 eV to the neighboring carbon atom prior to separation, the effect on the bonded partner is thought not to play a significant role in causing biological damage (Feinendegen 71).

(3) Nuclear recoil. According to the law of conservation of momentum, if an atom begins the beta decay process at rest, the vector sum of the momenta of all resulting particles must be zero. Since beta particles are usually emitted with a high velocity, considerable kinetic energy may be imparted to the residual atom, causing the nucleus to recoil. Of course, the maximum recoil energy is always very low in comparison to that of the emitted electron since the nucleus is so much more massive. Nevertheless, the energy of the recoiling nucleus is sufficient to cause molecular bond breakage in some instances.

Since tritium emits a beta particle of such low energy (about 6 keV average), a large fraction of the beta energy is deposited within the cell of origin resulting in a very large radiation dose. Several studies have been performed in an effort to determine whether the deleterious effects produced by tritium are a direct result of the radiation dose or from transmutation events. (For purposes of discussion, transmutation events will be considered to include nuclear recoil and daughter excitation, as well as the atomic number change.) These studies have been thoroughly reviewed by Krisch and Zelle (1969) and, more recently, by the NCRP (1979), and have focused on three main effects: cell killing, leukemogenesis, and mutagenesis.

Under normal growth conditions (as opposed to the storage of cultured cells at cryogenic temperatures to allow them to accumulate damage), the calculated dose to the cell nucleus from incorporated tritium required to produce a lethal effect is comparable to the amount of external radiation which produces the same effect. Thus, it is generally conceded that nearly all lethal effects can be accounted for on the basis of beta radiation effects alone. With respect to leukemogensis, the NCRP has concluded that it is of no consequence whether the cells at risk (the hemopoietic stem cells) are exposed as a result of partial or total body external exposure, or by the decay of radionuclides incorporated in stem cell DNA (NCRP 79). There is considerable evidence, however, that in some instances, transmutation events play a major role in inducing genetic mutations. This is apparent from the fact that the probability for inducing specific types of mutations in bacteria is highly dependent both on the type of DNA or RNA (ribonucleic acid) precursor which contains the tritium atoms and the specific location of the radionuclide on the precursor (Person 64, Person 68, Kieft 68). This difference in mutagenecity cannot be explained on the basis of beta radiation since the radiation dose to the cell nucleus was equal for each labelled precursor. Thus, it is apparent that transmutation effects play a key role in mutagenesis, the degree of which varies according to the type of precursor and the position of the tritium atom within the precursor. It should be noted, however, that the above studies used labelled DNA precursors, as opposed to tritiated water, for cell treatments.

It is the position of the NCRP that, although there is clear-cut evidence that transmutation effects do play a role in mutagenesis, the relative abundance of hydrogen atoms attached to the three specific DNA positions for which transmutation effects have been demonstrated among all DNA hydrogens is relatively low. Thus, it appears reasonably conservative to assume, for the purpose of practical hazards evaluation, that there is no significant transmutation effect for tritium incorporated into DNA, and that one may estimate hazards solely on the basis of absorbed beta dose (NCRP 79). For this reason, an increase in the relative biological effectiveness (RBE) or quality factor (Q) for tritium is not considered appropriate for the assessments performed in this report.

B.5.2 Carbon-14

The long half-life of C-14 (5730 years) has proven to be a major obstacle to research efforts directed at determining its biological effects. Labelled precursors with a high enough specific activity to be useful in studies of this nature are very difficult to synthesize. As an example, consider the experiments performed by Apelgot (1968), who incubated *E. coli* cells in a growth medium containing thymine labelled with C-14 at the two position of the pyrimidine ring. In order to observe a survival fraction of only 0.1, it was necessary to store the cells at -196°C for nearly two years. It was found that the lethal efficiency per decay of a C-14 atom in DNA was 0.01 (that is, it took 100 C-14 decays, on the average, to kill a cell).

Lee and Sega (1973) measured the mutagenic activity of C-14 decay on D. melanogaster (fruit fly) sperm cells. It was reasoned that since these cells were so small, only about 4 x 10^{-4} of the radiation dose associated with the beta particle would be absorbed within the cell. Thus, any observed genetic effects would probably be due to localized events associated with decay. The large number of nucleotide pairs in fruit fly DNA enabled incorporation of C-14 in amounts sufficient to produce several decays per cell. Their data indicated that eight to 80 disintegrations per X chromosome did not result in either broken chromosomes or a significant induction of lethal mutations. The authors concluded that, for all practical purposes, only the beta radiation produced by C-14 decay need by considered in evaluating the mutagenic or toxic effects of this nuclide.

Cleaver and Burki (1974) have determined the efficiency of DNA strand breakage resulting from C-14 decays originating from labeled thymidine in mammalian cells. Since the C-14 label was in the methyl group position of thymidine, it is very doubtful that decay would cause much strand breakage by transmutation and related local events. Thus, the authors concluded that most of the damage probably results from the beta radiation itself.

Conversely, other researchers have shown that the location of the C-14 atom within the labeled pyrimidine molecule largely influences the effectiveness for inducing chromosomal aberations in onion root tip cells (McQuade 56a, 56b, 60), and that the mutagenic effect of chronic C-14 decay is much greater than the effect produced by chronic external irradiation (Kuzin 64). Both of these findings suggest the existence of a transmutation effect.

Although it appears that the RBE for C-14 decay may, at least in some instances, be greater than unity, it must be remembered that a very small fraction of C-14 taken into the body eventually becomes incorporated into DNA. Thus, it is considered reasonable to expect that damage from the beta radiation dose will far outweigh that from transmutation effects. This report, therefore, uses a value of unity for the quality factor for the beta radiation dose produced by C-14 decay.

B.5.3 Uranium

Aside from being a radiological hazard, uranium displays properties of chemical toxicity. In fact, the chemical hazard to the kidneys represented by transportable compounds of uranium in the bloodstream is likely to be more important than the radiological hazard. On the other hand, the amounts of non-transportable uranium compounds in the lung or gastrointestinal tract are limited by the radiation dose which they deliver (ICRP 67). Table B.6 presents a summary of the development of standards for exposure to uranium, the basis for the standards and the postulated hazards.

Transportable uranium compounds are carried by the bloodstream and distributed between the kidney, bone, and other organs. Chemical toxicity in the kidney results from a decreased solubility in the urine due to the reabsorption of the complexing bicarbonate ions and increased acidity (Wills 49). Free uranium in the urine affects the tubular epithelium and causes cell death by interfering with normal membrane respiration (Hodge 56). Renal tissue damage is usually localized to the proximal convoluted tubule with total repair possible in 14 days following a mild intoxication (Barnett 49). The nephrotoxic limit of 3 ug U/g kidney can be related to a single deposition of a few milligrams (Alexander 74). Damage to the kidney is the limiting factor for body burdens of uranium of enrichments up to about five percent (i.e., uranium in which the relative abundance of the isotope U-235 has been increased from 0.7 to five percent). At greater enrichments, the radiation dose to the bone is the limiting consideration (Ford 64).

Minimum doses of uranium which have been shown to produce histological changes in the kidneys of dogs after one year of continuous inhalation (36 hours per week) are listed in Table B.7. As can be seen, no excess mortality and little or no kidney damage results from chronic exposure to uranium concentrations in air up to 2 mg/m^3 for soluble compounds and 10 mg/m^3 for insoluble compounds. The incidence of chronic renal damage in rabbits has been shown to be about the same as that in exposed dogs (Durbin 75). On the basis of these observations, Hodge and co-workers (1949) recommended a maximum allowable concentration for soluble uranium compounds in air of 50 $ug\ U/m^3$. However, the results of these experiments cannot be considered as positively demonstrating a threshold value, below which no damage is expected to result. Unfortunately, the test animals were killed at the end of the exposure period and no pertinent data

Table B.6. Historical Development of the Standard for Natural Uranium in Air(1).

MPC ug/m^3	Solubility(2)	Critical Organ	Postulated Hazard	Basis	Reference
150	NS(3)	NS	NS	Set at Same level as MPC for lead	Voegtlin, 1949
613	NS	Lung	Radiological	Calculation(4)	Morgan, 1947
100	I	Lung	Radiological	Calculation(5)	Hodge, 1949a
50	S	Kidney	Chemical	Animal Experiments	Hodge, 1949b
146	S	Kidney	Chemical	Animal Experiments	Tripartite Conference, 1949
73	I	Lung	Radiological	Calculated	NCRP, 1953
73	S	Kidney	Chemical	Animal Experiments	NCRP, 1953
210	I	Lung	Radiological	Calculation	ICRP, 1955
180	I	Lung	Radiological	Calculation	NCRP, 1959 ICRP, 1959
210	S	Kidney	Chemical	Calculation	NCRP, 1959 ICRP, 1959
70	I	Lung	Radiological	Calculation	ICRP, 1979

(1) From Hursh, 1975

(2) Solubility refers to the relative ease by which a compound can be transferred across body membranes. S = soluble (or, more apprortately, transportable); I = insoluble (non-transportable).

(3) NS = not specified

(4) Assumed 0.1 rem/day limit, 25% retention and effective half-life of 60 days.

(5) Assumed 0.045 rem/day limit and 25 ug U per gram lung.

Table B.7 Mortality and Kidney Damage in Dogs after One Year of Continuous (36 hr/wk) Inhalation.(1)

Compound	Dose Level (mg U/m^3)	U Concentration in Tissue (2)	Observations
$UO_2(NO_3)_2 \cdot 6H_2O$	0.15 - 2.0	Kidney, 0.5 Lung, 0.5 Bone, 2.2	No excess mortality No kidney changes
UF_6	0.05 - 0.2	Kidney, 0.3 Lung, 0.4 Bone, 0.5	No excess mortality Mild changes in alveolar bone from F^- Occasional mild kidney changes
UCl_4	0.05 - 0.2	Kidney, 0.2 Lung, 0.4 Bone, 0.1	No excess mortality Minimal kidney changes
UF_4	0.5 - 3.0	Kidney, 0.5 Lung, 5.7 Bone, 1.6	No excess mortality No changes in teeth or alveolar bone Minimal kidney changes
UO_2	1.0 - 10	Kidney, 1.7 Lung, 953 Bone, 4.9	No excess mortality No kidney changes observed

(1) Data of Stokinger, 1949

(2) Concentrations in kidneys and lungs are expressed as ug U/g wet tissue, and in bone as ug U/g bone ash.

are available on their ability to withstand stress, their susceptibility to renal disease, or shortening of the life span (Durbin 75).

The uranium exposure assessments performed in this report make no attempt at estimating the number of cases of renal damage which may result from consumer product use. This is not to infer that no such cases are possible. However, on the basis of current knowledge, the incidence of renal disease in humans with body burdens of less than a milligram or so of natural uranium appears to be very low. Therefore, the impact on human health will be estimated solely on the basis of radiological effects.

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APPENDIX C - REGULATIONS AND SAFETY CRITERIA

C.1 INTRODUCTION

The regulation of the use of radioactive materials in consumer products is outlined in this appendix. Presently, the possession, utilization, and transfer of both byproduct and source materials are controlled by the NRC directly or by the 26 Agreement States. Naturally occurring and accelerator-produced radioactive materials (NARM) are not controlled by the NRC, their regulation being at the option of the individual states. In 1977, it was determined that all 25 Agreement States (Rhode Island became the 26th Agreement State) and five of the 13 Non-Agreement States (of 28 Non-Agreement States) responding to information requests, had licensing programs covering NARM use (Nussbaumer 77).

"The regulation of naturally occurring and accelerator-produced radioactive material (NARM) is fragmented, non-uniform, and incomplete at both the Federal and State level." This was a major conclusion of an NRC-established Task Force (Nussbaumer 77) that reviewed the matter of regulation of these materials. The Task Force recommended "that the NRC seek legislative authority to regulate naturally occurring and accelerated-produced radioactive materials for the reason that these materials present signficiant radiation exposure potential."

Along these lines the NRC plans to reevaluate the adequacy of the existing policy pertaining to the distribution of consumer products containing byproduct and source material to the public on a license-exempt basis. The present NRC policy was published in the Federal Register on March 16, 1965 (30FR3462). A copy of the policy is attached at the end of this appendix. The NRC policy as established in 30FR3462, states, "it appears unlikely that the total contribution to the exposure of the general public to radiation from the use of radioactivity in consumer products will exceed small fractions or limits recommended for exposure to radiation from all sources ... If radioactive materials are used in sufficient quantities in products reaching the public, so as to raise any question of population exposure becoming a signficant fraction of the permissible dose to the gonads, the Commission will, at that time, reconsider its policy on the use of radioactive materials in consumer products." 30FR3462 goes on to state the principal considerations with respect to evaluation of products.

Licensing and regulatory policy and procedures for environmental protection as related to the NRC are stated in the Code of Federal Regulations under Title 10, Energy, Part 51 (10CFR51). The National Environmental Policy Act (NEPA) of 1969 requires all agencies of the Federal Government to prepare detailed environmental statements on proposals for legislation and other major Federal actions signficiantly affecting the quality of the human environment. The main objective is to take into careful consideration the environmental aspects of proposed actions. 10CFR51 states the NRC policy for the preparation and processing of environmental impact statements and related documents.

C.2 REGULATION OF BYPRODUCT AND SOURCE MATERIAL.

The extent of regulation of byproduct and source material is stated in the Code of Federal Regulations under Title 10, Energy, Chapter 1. Parts 20 and 30 of Title 10 (10CFR20 and 30) define byproduct material as "any radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation incident to the process of producing or utilizing special nuclear material." 10CFR40 defines byproduct material as "... the tailings or wastes produced by the extraction or concentration of uranium or thorium from any ore processed primarily for its source material content, including discrete surface wastes resulting from uranium solution extraction processes. Underground ore bodies depleted by such solution extraction operations do not constitute 'byproduct material' within this definition."

Source material is defined as "(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 percent) or more of: (i) uranium, (ii) thorium or (iii) any combination thereof. Source material does not include special nuclear material."

C.2.1 Regulation of Byproduct Material

10CFR30 states the rules applicable to all persons in the U. S. governing domestic licensing of byproduct material under the Atomic Energy Act of 1954, as amended, and under Title II of the Energy Reorganization Act of 1974, and exemptions from domestic licensing requirements permitted by Section 81 of the Act. Certain items or products containing byproduct material are discussed in 10CFR30.15. Those considered to be consumer products are included. A copy of 10CFR30.15 is included at the end of this appendix. 10CFR30.19 deals explicitly with self-luminous products containing tritium, krypton-85, or promethium-147. 10CFR30.20 deals explicitly with gas and aerosol detectors containing byproduct material.

In order to manufacture, process, produce, or initially transfer these products, a license is required. 10CFR32 covers specific domestic licenses to manufacture or transfer certain items containing byproduct material. General domestic licenses "to transfer, receive, acquire, own, possess, and use byproduct material incorporated" in certain products such as static elimination devices containing not more than 500 microcuries of polonium-210 are covered in 10CFR32. The manufacturers of such products, however, require a specific license.

Self-luminous products containing tritium, krypton-85, or promethium-147 are covered in 10CFR32.22 to 25. An applicant for a licenses is required to demonstrate that a product is designed and will be manufactured in a manner such that certain listed safety criteria are met. 10CFR32.22 to 25 are included at the end of this appendix.

Gas and aerosol detectors are covered in 10CFR32.26 to 29. An applicant for a license is required to demonstrate that a product is designed and will be manufactured in a manner such that certain listed criteria are met. 10CFR32.26 to 29 are included at the end of this appendix.

C.2.2 Regulation Of Source Material

10CFR40 states the regulations for the "issuance of licenses to receive title to, receive, possess, use, transfer, deliver, or import into or export from the U. S. source and byproduct materials." Certain items or products containing source material are discussed in 10CFR40.13. Those considered to be consumer products are included. A copy of 10CFR40.13 is included at the end of this appendix.

C.3 REGULATION OF NARM

The regulation of naturally occurring or accelerator-produced radioactive materials (NARM) on a state level is non-uniform and incomplete. The Conference of Radiation Control Directors, which is an organization whose membership is comprised of all directors of radiation control programs in the 50 states, the territories, and some large municipal agencies, established a Task Force to develop guidance for the individual state's evaluation of NARM products. Representatives from State Radiation Control Programs, the Bureau of Radiological Health (BRH), the NRC, and the U. S. Environmental Protection Agency (EPA) were on the Task Force. The major output of the Task Force was a set of guides for NARM materials (HEW 77).

NARM Guide 3 deals with gas and aerosol detectors and "provides criteria for the evaluation of gas and aerosol detectors containing radioactive material which are to be distributed to persons exempt ..." NARM Guide 3 applies essentially the same standards to Ra-226 smoke detectors as the NRC applies to smoke detectors with byproduct material. NARM Guide 6 deals with static elimination devices. NARM Guide 7 deals with radioluminous products, and NARM Guide 8 deals with electronic and electrical devices. The rationale for each guide is included in the report of the Task Force (HEW 77).

C.4 REFERENCES FOR APPENDIX C

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U. S. ATOMIC ENERGY COMMISSION

(Reprint from Federal Register) 30 F.R. 3462, March 16, 1965

ATOMIC ENERGY COMMISSION

USE OF BYPRODUCT MATERIAL AND SOURCE MATERIAL

Products Intended for Use by General Public (Consumer Products)

Criteria for the approval of products intended for use by the general public containing byproduct material and source material. This notice sets forth the essential terms of the Commission's policy with respect to approval of the use of byproduct material and source material in products intended for use by the general public (consumer products) without the imposition of regulatory controls on the consumer-user. This is accomplished by the exemption, on a case-by-case basis, of the possession and use of the approved items from the licensing requirements for byproduct and source material of the Atomic Energy Act of 1954, as amended, and of the Commission's regulations "Licensing of Byproduct Material", 10 CFR Part 30 and "Licensing of Source Material", 10 CFR Part 40.

1. At the present time it appears unlikely that the total contribution to the exposure of the general public to radiation from the use of radioactivity in consumer products will exceed small fractions of limits recommended for exposure to radiation from all sources. Information as to total quantities of radioactive materials being used in such products and the number of items being distributed will be obtained through recordkeeping and reporting requirements applicable to the manufacture and distribution of such products. If radioactive materials are used in sufficient quantities in products reaching the public so as to raise any question of population exposure becoming a significant fraction of the permissible dose to the gonads, the Commission will, at that time, reconsider its policy on the use of radioactive materials in consumer products.

2. Approval of a proposed consumer product will depend upon both associated exposures of persons to radiation and the apparent usefulness of the product. In general, risks of exposure to radiation will be considered to be acceptable if it is shown that in handling, use and disposal of the product it is unlikely that individuals in the population will receive more than a small fraction, less than a few hundredths, of individual dose limits recommended by such groups as the International Commission on Radiological Protection (ICRP), the National Council on Radiation Protection and Measurements (NCRP), and the Federal Radiation Council (FRC), and that the probability of individual doses approaching any of the specified limits is negligibly small. Otherwise, a decision will be more difficult and will require a careful weighing of all factors, including benefits that will accrue or be denied to the public as a result of the Commission's action. Factors that may be pertinent are listed in paragraphs 9 and 10, below.

3. It is considered that as a general rule products proposed for distribution will be useful to some degree. Normally the Commission will not attempt an extensive evaluation of the degree of benefit or usefulness of a product to the public. However, in cases where tangible benefits to the public are questionable and approval of such a product may result in widespread use of radioactive material, such as in common household items, the degree of usefulness and benefit that accrues to the public may be a deciding factor. In particular, the Commission considers that the use of radioactive material in toys, novelties, and adornments may be of marginal benefit.

4. Applications for approval of "off-the-shelf" items that are subject to mishandling especially by children will be approved only if they are found to combine an unusual degree of utility and safety.

5. The Commission has approved certain long standing uses of source material, most of which antedate the atomic energy program. These include:

(1) Use of uranium to color glass and glazes for certain decorative purposes;

(2) Thorium in various alloys and products (gas mantles, tungsten wire, welding rods, optical lenses, etc.) to impart desirable physical properties; and

(3) Uranium and thorium in photographic film and prints.

6. The Commission has also approved the use of tritium as a substitute luminous material for the long standing use of radium for this purpose on watch and clock dials and hands.

7. The Commission has approved additional uses of byproduct and source material in consumer products. These include the following:

(1) Tritium in automobile lock illuminators;

(2) Tritium in balances of precision;

(3) Uranium as shielding in shipping containers; and

(4) Uranium in fire detection units.

8. In approving uses of byproduct and source materials in consumer products, the Commission establishes limits on quantities or concentrations of radioactive materials and, if appropriate, on radiation emitted. In some cases other limitations, such as quality control and testing, considered important to health and safety are also specified.

PRINCIPAL CONSIDERATIONS WITH RESPECT TO EVALUATION OF PRODUCTS

9. In evaluating proposals for the use of radioactive materials in consumer products the principal considerations are:

(a) The potential external and internal exposure of individuals in the population to radiation from the handling, use and disposal of individual products;

(b) The potential total accumulative radiation dose to individuals in the population who may be exposed to radiation from a number of products;

(c) The long-term potential external and internal exposure of the general population from the uncontrolled disposal and dispersal into the environment of radioactive materials from products authorized by the Commission; and

(d) The benefit that will accrue to or be denied the public because of the utility of the product by approval or disapproval of a specific product.

10. The general criteria for approval of individual products are set forth in paragraph 2, above. Detailed evaluation of potential exposures would take into consideration the following factors together with other considerations which may appear pertinent in the particular case:

(a) The external radiation levels from the product.

(b) The proximity of the product to human tissue during use.

(c) The area of tissue exposed. A dose to the skin of the whole body would be considered more significant than a similar dose to a small portion of the skin of the body.

(d) Radiotoxicity of the radionuclides. The less toxic materials with a high permissible body burden, high concentration limit in air and water, would be considered more favorably than materials with a high radiotoxicity.

(e) The quantity of radioactive material per individual product. The smaller the quantity the more favorably would the product be considered.

(f) Form of material. Materials with a low solubility in body fluids will be considered more favorably than those with a high solubility.

(g) Containment of the material. Products which contain the material under very severe environmental conditions will be considered more favorably than those that will not contain the material under such conditions.

(h) Degree of access to product during normal handling and use. Products which are inaccessible to children and other persons during use will be considered more favorably than those that are accessible.

(Sec. 161, 68 Stat. 948; 42 U.S.C. 2201. Administrative Procedure Act, sec. 3, 60 Stat. 238; 5 U.S.C. 1002)

Dated at Washington, D.C., this 8th day of March 1965.

For the Atomic Energy Commission.

W. B. McCool,
Secretary.

[F.R. Doc. 65-2616; Filed, Mar. 15, 1965; 8:45 a.m.]

owned vehicle or vessel. In addition to the foregoing exemptions and subject to the requirement for licensing of Administration facilities and activities pursuant to section 202 of the Energy Reorganization Act of 1974, any prime contractor or subcontractor of the Administration or the Commission is exempt from the requirements for a license set forth in sections 81 and 82 of the Act and from the regulations in this part to the extent that such prime contractor or subcontractor manufacturers, produces, transfers, receives, acquires, owns, possesses, or uses byproduct material under his prime contract or subcontract when the Commission determines that the exemption of the prime contractor or subcontractor is authorized by law; and that, under the terms of the contract or subcontract, there is adequate assurance that the work thereunder can be accomplished without undue risk to the public health and safety.

(Sec. 161, as amended, Pub. L. 83-703, 68 Stat. 948 (42 U.S.C. 2201); sec. 201, as amended, Pub. L. 93-438, 88 Stat. 1243 (42 U.S.C. 5841))

[40 FR 8784, Mar. 3, 1975, as amended at 43 FR 6921, Feb. 17, 1978]

§ 30.13 Carriers.

Common and contract carriers, freight forwarders, warehousemen, and the U.S. Postal Service are exempt from the regulations in this part and Parts 31-35 of this chapter and the requirements for a license set forth in section 81 of the Act to the extent that they transport or store byproduct material in the regular course of carriage for another or storage incident thereto.

(Sec. 161, as amended, Pub. L. 83-703, 68 Stat. 948 (42 U.S.C. 2201); sec. 201, as amended, Pub. L. 93-438, 88 Stat. 1243 (42 U.S.C. 5841))

[37 FR 3985, Feb. 25, 1972, as amended at 43 FR 6921, Feb. 17, 1978]

§ 30.14 Exempt concentrations.

(a) Except as provided in paragraphs (c) and (d) of this section, any person is exemtpt from the requirements for a license set forth in section 81 of the Act and from the regulations in this part and Parts 31-35 of this chapter to the extent that such person receives, possesses, uses, transfers, owns or acquires products or materials containing byproduct material in concentrations not in excess of those listed in § 30.70.

(b) This section shall not be deemed to authorize the import of byproduct material or products containing byproduct material.

(c) A manufacturer, processor, or producer of a product or material in an agreement State is exempt from the requirements for a license set forth in section 81 of the Act and from the regulations in this part and Parts 31, 32, 33, and 34 to the extent that he transfers byproduct material contained in a product or material in concentrations not in excess of those specified in § 30.70 and introduced into the product or material by a licensee holding a specific license issued by an agreement State, the Commission, or the Atomic Energy Commission expressly authorizing such introduction. This exemption does not apply to the transfer of byproduct material contained in any food, beverage, cosmetic, drug, or other commodity or product designed for ingestion or inhalation by, or application to, a human being.

(d) No person may introduce byproduct material into a product or material knowing or having reason to believe that it will be transferred to persons exempt under this section or equivalent regulations of an Agreement State, except in accordance with a license issued pursuant to § 32.11 of this chapter or the general license provided in § 150.20 of this chapter.

(Sec. 161, as amended, Pub. L. 83-703, 68 Stat. 948 (42 U.S.C. 2201); sec. 201, as amended, Pub. L. 93-438, 88 Stat. 1243 (42 U.S.C. 5841))

[30 FR 8185, June 26, 1965, as amended at 40 FR 8785, Mar. 3, 1975; 43 FR 6921, Feb. 17, 1978]

§ 30.15 Certain items containing byproduct material.

(a) Except for persons who apply byproduct material to, or persons who incorporate byproduct material into, the following products, or persons who initially transfer for sale or distribution the following products containing byproduct material, any person is exempt from the requirements for a license set forth in section 81 of the Act and from the regulations in Parts 20 and 30-35 of this chapter to the extent that such person receives, possesses, uses, transfers, owns, or acquires the following products:

(1) Timepieces or hands or dials containing not more than the following specified quantities of byproduct material and not exceeding the following specified levels of radiation:

(i) 25 millicuries of tritium per timepiece,

(ii) 5 millicuries of tritium per hand,

(iii) 15 millicuries of tritium per dial (bezels when used shall be considered as part of the dial),

(iv) 100 microcuries of promethium 147 per watch or 200 microcuries of promethium 147 per any other timepiece,

(v) 20 microcuries of promethium 147 per watch hand or 40 microcuries of promethium 147 per other timepiece hand,

(vi) 60 microcuries of promethium 147 per watch dial or 120 microcuries of promethium 147 per other timepiece dial (bezels when used shall be considered as part of the dial),

(vii) The levels of radiation from hands and dials containing promethium 147 will not exceed, when measured through 50 milligrams per square centimeter of absorber:

(*a*) For wrist watches, 0.1 millirad per hour at 10 centimeters from any surface,

(*b*) For pocket watches, 0.1 millirad per hour at 1 centimeter from any surface,

(*c*) For any other timepeice, 0.2 millirad per hour at 10 centimeters from any surface.

(2) Lock illuminators containing not more than 15 millicuries of tritium or not more than 2 millicuries of promethium 147 installed in automobile locks. The levels of radiation from each lock illuminator containing promethium 147 will not exceed 1 millirad per hour at 1 centimeter from any surface when measured through 50 milligrams per square centimeter of absorber.

(3) Balances of precision containing not more than 1 millicurie of tritium per balance or not more than 0.5 millicurie of tritium per balance part.

(4) Automobile shift quadrants containing not more than 25 millicuries of tritium.

(5) Marine compasses containing not more than 750 millicuries of tritium gas and other marine navigational instruments containing not more than 250 millicuries of tritium gas.

(6) Thermostat dials and pointers containing not more than 25 millicuries of tritium per thermostat.

(7) [Reserved]

(8) Electron tubes: *Provided*, That each tube does not contain more than one of the following specified quantities of byproduct material:

(i) 150 millicuries of tritium per microwave receiver protector tube or 10 millicuries of tritium per any other electron tube;

(ii) 1 microcurie of cobalt-60;

(iii) 5 microcuries of nickel-63;

(iv) 30 microcuries of krypton-85;

(v) 5 microcuries of cesium-137;

(vi) 30 microcuries of promethium-147;

And provided further, That the levels of radiation from each electron tube containing byproduct material do not exceed 1 millirad per hour at 1 centimeter from any surface when measured through 7 milligrams per square centimeter of absorber.[1]

(9) Ionizing radiation measuring instruments containing, for purposes of internal calibration or standardization, a source of byproduct material not exceeding the applicable quantity set forth in § 30.71, Schedule B.

(10) Spark gap irradiators containing not more than 1 microcurie of cobalt-60 per spark gap irradiator for use in electrically ignited fuel oil burners having a firing rate of at least 3 gallons per hour (11.4 liters per hour).

(b) Any person who desires to apply byproduct material to, or to incorporate byproduct material into, the products exempted in paragraph (a) of this section, or who desires to initially transfer for sale or distribution such

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[1] For purposes of this paragraph "electron tubes" include spark gap tubes, power tubes, gas tubes including glow lamps, receiving tubes, microwave tubes, indicator tubes, pickup tubes, radiation detection tubes, and any other completely sealed tube that is designed to conduct or control electrical currents.

products containing byproduct material, should apply for a specific license pursuant to § 32.14 of this chapter, which license states that the product may be distributed by the licensee to persons exempt from the regulations pursuant to paragraph (a) of this section.

(Secs. 81, 161, Pub. L. 83-703, 68 Stat. 935, 948 (42 U.S.C. 2211, 2201); sec. 201, Pub. L. 93-438, 88 Stat. 1242 (42 U.S.C. 5841))

[31 FR 5316, Apr. 2, 1966, as amended at 31 FR 14349, Nov. 8, 1966; 32 FR 785, Jan. 24, 1967; 32 FR 6434, Apr. 26, 1967; 32 FR 13921, Oct. 6, 1967; 34 FR 6651, Apr. 18, 1969; 34 FR 19546, Dec. 11, 1969; 35 FR 6427, Apr. 22, 1970; 35 FR 8820, June 6, 1970; 43 FR 2387, Jan. 17, 1978; 43 FR 6921, Feb. 17, 1978]

§ 30.16 Resins containing scandium 46 and designed for sand-consolidation in oil wells.

Any person is exempt from the requirements for a license set forth in section 81 of the Act and from the regulations in Parts 20 and 30-35 of this chapter to the extent that such person receives, possesses, uses, transfers, owns, or acquires synthetic plastic resins containing scandium 46 which are designed for sand-consolidation in oil wells, and which have been manufactured or initially transferred for sale or distribution, in accordance with a specific license issued pursuant to § 32.17 of this chapter or equivalent regulations of an agreement State. The exemption in this section does not authorize the manufacture or initial transfer for sale or distribution of any resins containing scandium 46.

(Sec. 161, as amended, Pub. L. 83-703, 68 Stat. 948 (42 U.S.C. 2201); sec. 201, as amended, Pub. L. 93-438, 88 Stat. 1243 (42 U.S.C. 5841))

[32 FR 4241, Mar. 18, 1967, as amended at 43 FR 6921, Feb. 21, 1978]

§ 30.18 Exempt quantities.

(a) Except as provided in paragraphs (c) and (d) of this section, any person is exempt from the requirements for a license set forth in section 81 of the Act and from the regulations in Parts 30-34 of this chapter to the extent that such person receives, possesses, uses, transfers, owns, or acquires byproduct material in individual quantities each of which does not exceed the applicable quantity set forth in § 30.71, Schedule B.

(b) Any person who possesses byproduct material received or acquired prior to September 25, 1971 under the general license then provided in § 31.4 of this chapter is exempt from the requirements for a license set forth in section 81 of the Act and from the regulations in Parts 30-34 of this chapter to the extent that such person possesses, uses, transfers, or owns such byproduct material.

(c) This section does not authorize for purposes of commercial distribution the production, packaging, repackaging, or transfer of byproduct material or the incorporation of byproduct material into products intended for commercial distribution.

(d) No person may, for purposes of commercial distribution, transfer byproduct material in the individual quantities set forth in § 30.71 Schedule B, knowing or having reason to believe that such quantities of byproduct material will be transferred to persons exempt under this section or equivalent regulations of an Agreement State, except in accordance with a license issued under § 32.18 of this chapter, which license states that the byproduct material may be transferred by the licensee to persons exempt under this section or the equivalent regulations of an Agreement State.

(Sec. 161, as amended, Pub. L. 83-703, 68 Stat. 948 (42 U.S.C. 2201); sec. 201, as amended, Pub. L. 93-438, 88 Stat. 1243 (42 U.S.C. 5841))

[35 FR 6427, Apr. 22, 1970, as amended at 36 FR 16898, Aug. 26, 1971; 43 FR 6921, Feb. 17, 1978]

§ 30.19 Self-luminous products containing tritium, krypton-85, or promethium-147.

(a) Except for persons who manufacture, process, produce or initially transfer for sale or distribution self-luminous products containing tritium, krypton-85, or promethium-147, and except as provided in paragraph (c) of this section, any person is exempt from the requirements for a license set forth in section 81 of the Act and from the regulations in Parts 20 and 30-35 of this chapter to the extent that such

Combined"; and (3) set forth appropriate additional radiation safety precautions and instructions relating to the handling, use, storage, and disposal of the radioactive material.

[35 FR 6428, Apr. 22, 1970]

§ 32.20 Same: records and material transfer reports.

Each person licensed under § 32.18 shall maintain records of transfer of material for a period of two years after such transfer, identifying, by name and address, each person to whom byproduct material is transferred for use under § 30.18 of this chapter or the equivalent regulations of an Agreement State, and stating the kinds and quantities of byproduct material transferred. An annual summary report stating the total quantity of each isotope transferred under the specific license shall be filed with the Director of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the appropriate NRC Regional Office listed in Appendix D of Part 20 of this chapter. Each report shall cover the year ending June 30, and shall be filed within thirty (30) days thereafter. If no transfers of byproduct material have been made pursuant to § 32.18 during the reporting period, the report shall so indicate.

[35 FR 6429, Apr. 22, 1970, as amended at 38 FR 1271, Jan. 11, 1973; 41 FR 16446, Apr. 19, 1976; 41 FR 18302, May 3, 1976]

§ 32.22 Self-luminous products containing tritium, krypton-85 or promethium-147: requirements for license to manufacture, process, produce, or initially transfer.

(a) An application for a specific license to manufacture, process, or produce self-luminous products containing tritium, krypton-85, or promethium-147, or to initially transfer such products for use pursuant to § 30.19 of this chapter or equivalent regulations of an Agreement State, will be approved if:

(1) The applicant satisfies the general requirements specified in § 30.33 of this chapter: *Provided, however,* That the requirements of § 30.33(a) (2) and (3) do not apply to an application for a license to transfer tritium, krypton-85, or promethium-147 in self-luminous products manufactured, processed, or produced pursuant to a license issued by an Agreement State.

(2) The applicant submits sufficient information relating to the design, manufacture, prototype testing, quality control procedures, labeling or marking, and conditions of handling, storage, use, and disposal of the self-luminous product to demonstrate that the product will meet the safety criteria set forth in § 32.23. The information should include:

(i) A description of the product and its intended use or uses.

(ii) The type and quantity of byproduct material in each unit.

(iii) Chemical and physical form of the byproduct material in the product and changes in chemical and physical form that may occur during the useful life of the product.

(iv) Solubility in water and body fluids of the forms of the byproduct material identified in paragraph (a)(2) (iii) and (xii) of this section.

(v) Details of construction and design of the product as related to containment and shielding of the byproduct material and other safety features under normal and severe conditions of handling, storage, use, and disposal of the product.

(vi) Maximum external radiation levels at 5 and 25 centimeters from any external surface of the product, averaged over an area not to exceed 10 square centimeters, and the method of measurement.

(vii) Degree of access of human beings to the product during normal handling and use.

(viii) Total quantity of byproduct material expected to be distributed in the product annually.

(ix) The expected useful life of the product.

(x) The proposed method of labeling or marking each unit with identification of the manufacturer or initial transferor of the product and the byproduct material in the product.

(xi) Procedures for prototype testing of the product to demonstrate the effectiveness of the containment, shielding, and other safety features under both normal and severe conditions of handling, storage, use, and disposal of the product.

(xii) Results of the prototype testing of the product, including any change in the form of the byproduct material contained in the product, the extent to which the byproduct material may be released to the environment, any increase in external radiation levels, and any other changes in safety features.

(xiii) The estimated external radiation doses and dose commitments relevant to the safety criteria in § 32.23 and the basis for such estimates.

(xiv) A determination that the probabilities with respect to the doses referred to in § 32.23(d) meet the criteria of that paragraph.

(xv) Quality control procedures to be followed in the fabrication of production lots of the product and the quality control standards the product will be required to meet.

(xvi) Any additional information, including experimental studies and tests, required by the Commission.

(b) Notwithstanding the provisions of paragraph (a) of this section, the Commission may deny an application for a specific license under this section if the end uses of the product cannot be reasonably foreseen.

(Sec. 161, as amended, Pub. L. 83-703, 68 Stat. 948 (42 U.S.C. 2201); sec. 201, as amended, Pub. L. 93-438, 88 Stat. 1243 (42 U.S.C. 5841))

[34 FR 9026, June 6, 1969, as amended at 43 FR 6923, Feb. 17, 1978]

§ 32.23 Same: safety criteria.

An applicant for a license under § 32.22 shall demonstrate that the product is designed and will be manufactured so that:

(a) In normal use and disposal of a single exempt unit, it is unlikely that the external radiation dose in any one year, or the dose commitment resulting from the intake of radioactive material in any one year, to a suitable sample of the group of individuals expected to be most highly exposed to radiation or radioactive material from the product will exceed the dose to the appropriate organ as specified in Column I of the Table in § 32.24.

(b) In normal handling and storage of the quantities of exempt units likely to accumulate in one location during marketing, distribution, installation, and servicing of the product, it is unlikely that the external radiation dose in any one year, or the dose commitment resulting from the intake of radioactive material in any one year, to a suitable sample of the group of individuals expected to be most highly exposed to radiation or radioactive material from the product will exceed the dose to the appropriate organ as specified in Column II of the Table in § 32.24.

(c) It is unlikely that there will be a significant reduction in the effectiveness of the containment, shielding, or other safety features of the product from wear and abuse likely to occur in normal handling and use of the product during its useful life.

(d)[1] In use and disposal of a single exempt unit, or in handling and storage of the quantities of exempt units 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 a person would receive an external radiation dose or dose commitment in excess of the dose to the appropriate organ as specified in Column III of the Table in § 32.24, and the probability is negligible that a person would receive an external radiation dose or dose commitment in excess of the dose to the apprpriate organ as specified in Column IV of the Table in § 32.24.

[34 FR 9027, June 6, 1969]

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[1] It is the intent of this paragraph 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 have been expressed in general terms to emphasize the approximate nature of the estimates which are to be made. The following values may be used as guides in estimating compliance with the criteria:

Low—not more than one such failure per year for each 10,000 exempt units distributed.

Negligible—not more than one such failure per year for each 1 million exempt units distributed.

§ 32.24 Same: table of organ doses.

Part of body	Column I (rem)	Column II (rem)	Column III (rem)	Column IV (rem)
Whole body; head and trunk; active blood-forming organs; gonads; or lens of eye..	0.001	0.01	0.5	15
Hands and forearms; feet and ankles; localized areas of skin averaged over areas no larger than 1 square centimeter..................	0.015	0.15	7.5	200
Other organs..................	0.003	0.03	1.5	50

[34 FR 9329, June 13, 1969]

§ 32.25 Conditions of licenses issued under § 32.22: quality control, labeling, and reports of transfer.

Each person licensed under § 32.22 shall:

(a) Carry out adequate control procedures in the manufacture of the product to assure that each production lot meets the quality control standards approved by the Commission;

(b) Label or mark each unit so that the manufacturer, processor, producer, or initial transferor of the product and the byproduct material in the product can be identified; and

(c) File an annual report with the Director of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the appropriate NRC Regional Office listed in Appendix D of Part 20 of this chapter, which shall include the following information on products transferred to other persons for use under § 30.19 of this chapter or equivalent regulations of an Agreement State: (1) A description or identification of the type of each product; (2) for each radionuclide in each type of product, the total quantity of the radionuclide; and (3) the number of units of each type of product during the reporting period. If no transfers of byproduct material have been made pursuant to § 32.22 during the reporting period, the report shall so indicate. The report shall cover the year ending June 30, and shall be filed within 30 days thereafter.

(Sec. 161, as amended, Pub. L. 83-703, 68 Stat. 948 (42 U.S.C. 2201); sec. 201, as amended, Pub. L. 93-438, 88 Stat. 1243 (42 U.S.C. 5841))

[34 FR 9027, June 6, 1969, as amended at 35 FR 6703, Apr. 28, 1970, as amended at 38 FR 1271, Jan. 11, 1973; 41 FR 16446, Apr. 19, 1976; 43 FR 6923, Feb. 17, 1978]

§ 32.26 Gas and aerosol detectors containing byproduct material: requirements for license to manufacture, process, produce, or initially transfer.

An application for a specific license to manufacture, process, or produce gas and aerosol detectors containing byproduct material and designed to protect life or property from fires and airborne hazards, or to initial transfer such products for use pursuant to § 30.20 of this chapter or equivalent regulations of an Agreement State, will be approved if:

(a) The applicant satisfies the general requirements specified in § 30.33 of this chapter: *Provided, however,* That the requirements of § 30.33(a) (2) and (3) do not apply to an application for a license to transfer byproduct material in gas and aerosol detectors manufactured, processed or produced pursuant to a license issued by an Agreement State.

(b) The applicant submits sufficient information relating to the design, manufacture, prototype testing, quality control procedures, labeling or marking, and conditions of handling, storage, use, and disposal of the gas and aerosol detector to demonstrate that the product will meet the safety criteria set forth in § 32.27. The information should include:

(1) A description of the product and its intended use or uses;

(2) The type and quantity of byproduct material in each unit;

(3) Chemical and physical form of the byproduct material in the product and changes in chemical and physical form that may occur during the useful life of the product;

(4) Solubility in water and body fluids of the forms of the byproduct material identified in paragraphs (b) (3) and (12) of this section;

(5) Details of construction and design of the product as related to containment and shielding of the byproduct material and other safety features under normal and severe conditions of handling, storage, use, and disposal of the product;

(6) Maximum external radiation levels at 5 and 25 centimeters from any external surface of the product, averaged over an area not to exceed 10 square centimeters, and the method of measurement;

(7) Degree of access of human beings to the product during normal handling and use;

(8) Total quantity of byproduct material expected to be distributed in the product annually;

(9) The expected useful life of the product;

(10) The proposed method of labeling or marking each unit with identification of the manufacturer or initial transferor of the product and the byproduct material in the product;

(11) Procedures for prototype testing of the product to demonstrate the effectiveness of the containment, shielding, and other safety features under both normal and severe conditions of handling, storage, use, and disposal of the product;

(12) Results of the prototype testing of the product, including any change in the form of the byproduct material contained in the product, the extent to which the byproduct material may be released to the environment, any increase in external radiation levels, and any other changes in safety features;

(13) The estimated external radiation doses and dose commitments relevant to the safety criteria in § 32.27 and the basis for such estimates;

(14) A determination that the probabilities with respect to the doses referred to in § 32.27(c) meet the criteria of that paragraph;

(15) Quality control procedures to be followed in the fabrication of production lots of the product and the quality control standards the product will be required to meet; and

(16) Any additional information, including experimental studies and tests, required by the Commission.

(Sec. 161, as amended, Pub. L. 83-703, 68 Stat. 948 (42 U.S.C. 2201); sec. 201, as amended, Pub. L. 93-438, 88 Stat. 1243 (42 U.S.C. 5841))

[34 FR 6653, Apr. 18, 1969, as amended at 43 FR 6923, Feb. 17, 1978]

§ 32.27 Same: safety criteria.

An applicant for a license under § 32.26 shall demonstrate that the product is designed and will be manufactured so that:

(a) In normal use and disposal of a single exempt unit, and in normal handling and storage of the quantities of exempt units likely to accumulate in one location during marketing, distribution, installation, and servicing of the product, it is unlikely that the external radiation dose in any one year, or the dose commitment resulting from the intake of radioactive material in any one year, to a suitable sample of the group of individuals expected to be most highly exposed to radiation or radioactive material from the product will exceed the dose to the appropriate organ as specified in Column I of the table in § 32.28.

(b) It is unlikely that there will be a significant reduction in the effectiveness of the containment, shielding, or other safety features of the product from wear and abuse likely to occur in normal handling and use of the product during its useful life.

(c) In use and disposal of a single exempt unit and in handling and storage of the quantities of exempt units 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 a person would receive an external radiation dose or dose commitment in excess of the dose to the appropriate organ as specified in Column II of the table in § 32.28, and the probability is negligible that a person would receive an external radiation dose or dose commitment in excess of the dose to the appropriate organ as specified in Column III of the table in § 32.28.[1]

[34 FR 6654, Apr. 18, 1969]

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[1] It is the intent of this paragraph that as the magnitude of the potential dose in-

Footnotes continued on next page

§ 32.28 Same: table of organ doses.

Part of body	Column I (rem)	Column II (rem)	Column III (rem)
Whole body; head and trunk; active blood-forming organs; gonads; or lens of eye	0.005	0.5	15
Hands and forearms; feet and ankles; localized areas of skin averaged over areas no larger than 1 square centimeter	0.075	7.5	200
Other organs	0.015	1.5	50

[34 FR 6654, Apr. 18, 1969]

(1) § 32.29 Conditions of licenses issued under § 32.26: quality control, labeling, and reports of transfer.

Each person licensed under § 32.26 shall:

(a) Carry out adequate control procedures in the manufacture of the product to assure that each production lot meets the quality control standards approved by the Commission;

(b) Label or mark each unit so that the manufacturer or initial transferor of the product and the byproduct material in the product can be identified; and provide such other information with each unit as may be required by the Commission, including disposal instructions when appropriate; and

(c) File an annual report with the Director of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the appropriate NRC Regional Office listed in Appendix D of Part 20 of this chapter, which shall include the following information on products transferred to other persons for use under § 30.20 of this chapter or equivalent regulations of an Agreement State: (1) A description or identification of the type of each product; (2) for each radionuclide in each type of product, the total quantity of the radionuclide; and (3) the number of units of each type of product during the reporting period. If no transfers of byproduct material have been made pursuant to § 32.26 during the reporting period, the report shall so indicate. The report shall cover the year ending June 30, and shall be filed within 30 days thereafter.

(Sec. 161, as amended, Pub. L. 83-703, 68 Stat. 948 (42 U.S.C. 2201); sec. 201, as amended, Pub. L. 93-438, 88 Stat. 1243 (42 U.S.C. 5841))

[34 FR 6654, Apr. 18, 1969, as amended at 35 FR 6703, Apr. 28, 1970; 38 FR 1271, Jan. 11, 1973; 41 FR 16446, Apr. 19, 1976; 43 FR 6923, Feb. 17, 1978]

§ 32.40 Schedule A—Prototype tests for automobile lock illuminators.

An applicant for a license pursuant to § 32.14 to install lock illuminators into automobile locks, or to initially transfer lock illuminators in automobile locks for use pursuant to § 30.15 of this chapter shall conduct the following prototype tests on each of five prototype devices, consisting of the automobile lock with the installed illuminator in the following order:

(a) The device shall be subjected to 100 hours of accelerated weathering in a suitable weathering machine which simulates the most severe conditions of normal use;

(b) The device shall be dropped upon a concrete or iron surface in a 3-foot free gravitational fall, or shall be subjected to an equivalent treatment in a test device simulating such a fall. The drop test shall be repeated 100 times from random orientations;

(c) The device shall be attached to a vibratory fixture and vibrated at a rate of not less than 26 cycles per second and a vibration acceleration of not less than 2 G for a period of not less than 1 hour;

(d) On completion of the foregoing tests, the device shall be immersed in

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Footnotes continued from previous page creases above that permitted under normal conditions, the probability that any individual will receive such a dose must decrease. The probabilities have been expressed in general terms to emphasize the approximate nature of the estimates which are to be made. The following values may be used as guides in estimating compliance with the criteria:

Low—not more than one such failure per year for each 10,000 exempt units distributed.

Negligible—not more than one such failure per year for each one million exempt units distributed.

(1) Section 32.29 update of May 27, 1980 not included.

prime contractor of the Administration is exempt from the requirements for a license set forth in sections 62, 63, and 64 of the Act and from the regulations in this part to the extent that such contractor, under his prime contract with the Administration, receives, possesses, uses, transfers or delivers source material for: (a) The performance of work for the Administration at a United States Government-owned or controlled site, including the transportation of source material to or from such site and the performance of contract services during temporary interruptions of such transportation; (b) research in, or development, manufacture, storage, testing or transportation of, atomic weapons or components thereof; or (c) the use or operation of nuclear reactors or other nuclear devices in a United States Government-owned vehicle or vessel. In addition to the foregoing exemptions, and subject to the requirement for licensing of Administration facilities and activities pursuant to section 202 of the Energy Reorganization Act of 1974, any prime contractor or subcontractor of the Administration or the Commission is exempt from the requirements for a license set forth in sections 62, 63, and 64 of the Act and from the regulations in this part to the extent that such prime contractor or subcontractor receives, possesses, uses, transfers or delivers source material under his prime contract or subcontract when the Commission determines that the exemption of the prime contractor or subcontractor is authorized by law; and that, under the terms of the contract or subcontract, there is adequate assurance that the work thereunder can be accomplished without undue risk to the public health and safety.

(Sec. 161, as amended, Pub. L. 83-703, 68 Stat. 948 (42 U.S.C. 2201); sec. 201, as amended, Pub. L. 93-438, 88 Stat. 1243 (42 U.S.C. 5841))

[40 FR 8787, Mar. 3, 1975, as amended at 43 FR 6923, Feb. 17, 1978]

§ 40.12 Carriers.

Common and contract carriers, freight forwarders, warehousemen, and the U.S. Postal Service are exempt from the regulations in this part and the requirements for a license set forth in section 62 of the Act to the extent that they transport or store source material in the regular course of carriage for another or storage incident thereto.

[37 FR 3985, Feb. 25, 1972]

§ 40.13 Unimportant quantities of source material.

(a) Any person is exempt from the regulations in this part and from the requirements for a license set forth in section 62 of the Act to the extent that 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 one-twentieth of 1 percent (0.05 percent) of the mixture, compound, solution or alloy.

(b) Any person is exempt from the regulations in this part and from the requirements for a license set forth in section 62 of the act to the extent that such person receives, possesses, uses, or transfers unrefined and unprocessed ore containing source material; provided, that, except as authorized in a specific license, such person shall not refine or process such ore.

(c) Any person is exempt from the regulation in this part and from the requirements for a license set forth in section 62 of the Act to the extent that such person receives, possesses, uses, or transfers:

(1) Any quantities of thorium contained in (i) incandescent gas mantles, (ii) vacuum tubes, (iii) welding rods, (iv) electric lamps for illuminating purposes: *Provided*, That each lamp does not contain more than 50 milligrams of thorium, (v) germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting: *Provided*, That each lamp does not contain more than 2 grams of thorium, (vi) rare earth metals and compounds, mixtures, and products containing not more than 0.25 percent by weight thorium, uranium, or any combination of these, or (vii) personnel neutron dosimeters: *Provided*, That each dosimeter does not contain more than 50 milligrams of thorium.

(2) Source material contained in the following products: (i) Glazed ceramic tableware, provided that the glaze con-

tains not more than 20 percent by weight source material; (ii) piezoelectric ceramic containing not more than 2 percent by weight source material; (iii) glassware, glass enamel, and glass enamel frit containing not more than 10 percent by weight source material; but not including commercially manufactured glass brick, pane glass, ceramic tile, or other glass, glass enamel or ceramic used in construction;

(3) Photographic film, negatives, and prints containing uranium or thorium;

(4) Any finished product or part fabricated of, or containing tungsten or magnesium-thorium alloys, provided that the thorium content of the alloy does not exceed 4 percent by weight and that the exemption contained in this subparagraph shall not be deemed to authorize the chemical, physical or metallurgical treatment or processing of any such product or part; and

(5) Uranium contained in counterweights installed in aircraft, rockets, projectiles, and missiles, or stored or handled in connection with installation or removal of such counterweights: *Provided*, That:

(i) The counterweights are manufactured in accordance with a specific license issued by the Commission or the Atomic Energy Commission authorizing distribution by the licensee pursuant to this paragraph;

(ii) Each counterweight has been impressed with the following legend clearly legible through any plating or other covering: "Depleted Uranium";[1]

(iii) Each counterweight is durably and legibly labeled or marked with the identification of the manufacturer, and the statement: "Unauthorized Alterations Prohibited";[1] and

(iv) The exemption contained in this paragraph shall not be deemed to authorize the chemical, physical, or metallurgical treatment or processing of any such counterweights other than repair or restoration of any plating or other covering.

(6) Uranium used as shielding constituting part of any shipping container which is conspicuously and legibly impressed with the legend "CAUTION—RADIOACTIVE SHIELDING—URANIUM" and which meets the specifications for containers for radioactive materials prescribed by § 178.250, Specification 55, Part 178 of the regulations of the Department of Transportation (49 CFR 178.250).

(7) Thorium contained in finished optical lenses, provided that each lens does not contain more than 30 percent by weight of thorium; and that the exemption contained in this subparagraph shall not be deemed to authorize either:

(i) The shaping, grinding or polishing of such lens or manufacturing processes other than the assembly of such lens into optical systems and devices without any alteration of the lens; or

(ii) The receipt, possession, use, transfer, or of thorium contained in contact lenses, or in spectacles, or in eyepieces in binoculars or other optical instruments.

(8) Thorium contained in any finished aircraft engine part containing nickel-thoria alloy, *Provided*, That:

(i) The thorium is dispersed in the nickel-thoria alloy in the form of finely divided thoria (thorium dioxide); and

(ii) The thorium content in the nickel-thoria alloy does not exceed 4 percent by weight.

(9) The exemptions in this paragraph (c) do not authorize the manufacture of any of the products described.

(d) Any person is exempt from the regulations in this part and from the requirements for a license set forth in section 62 of the Act to the extent that such person receives, possesses, uses, or transfers uranium contained in detector heads for use in fire detection units, provided that each detector head contains not more than 0.005 microcurie of uranium. The exemption in this paragraph does not authorize the manufacture of any detector head containing uranium.

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[1] The requirements specified in paragraphs (c)(5) (ii) and (iii) of this section need not be met by counterweights manufactured prior to Dec. 31, 1969: *Provided*, That such counterweights were manufactured under a specific license issued by the Atomic Energy Commission and were impressed with the legend required by § 40.13(c)(5)(ii) in effect on June 30, 1969.

(Sec. 201, Pub. L. 93-438, 88 Stat. 1242 (42 U.S.C. 5841))

[26 FR 284, Jan. 14, 1961, as amended at 26 FR 10929, Nov. 22, 1961; 28 FR 8021, Aug. 7, 1963; 28 FR 14309, Dec. 27, 1963; 30 FR 15802, Dec. 22, 1965; 32 FR 15873, Nov. 18, 1967; 34 FR 14067, Sept. 5, 1969; 34 FR 19546, Dec. 11, 1969; 35 FR 6313, Apr. 18, 1970; 42 FR 6580, Feb. 3, 1977; 43 FR 6923, Feb. 17, 1978]

§ 40.14 Specific exemptions.

(a) The Commission may, upon application of any interested person or upon its own initiative, grant such exemptions from the requirements of the regulation in this part as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest.

(b) Any person subject to the provisions of §§ 40.31(f) and 40.32(e) may request an exemption from the requirements of those provisions. The Commission may grant an exemption from the provisions of §§ 40.31(f) and 40.32(e) upon considering and balancing the following factors:

(1) Whether conduct of the proposed activities will give rise to a significant adverse impact on the environment and the nature and extent of such impact, if any;

(2) Whether redress of any adverse environmental impact from conduct of the proposed activities can reasonably be effected should such redress be necessary;

(3) Whether conduct of the proposed activities would foreclose subsequent adoption of alternatives; and

(4) The effect of delay in conducting such activities on the public interest. During the period of any exemption granted pursuant to this paragraph (b), any activities conducted shall be carried out in such a manner as will minimize or reduce their environmental impact.

[37 FR 5747, Mar. 21, 1972, as amended at 39 FR 26279, July 18, 1974; 40 FR 8787, Mar. 3, 1975]

GENERAL LICENSES

§ 40.20 Types of licenses.

Licenses for source material are of two types: general and specific. The general licenses provided in this part are effective without the filing of applications with the Commission or the issuance of licensing documents to particular persons. Specific licenses are issued to named persons upon applications filed pursuant to the regulations in this part.

§ 40.21 General license to receive title to source material.

A general license is hereby issued authorizing the receipt of title to source material without regard to quantity. This general license does not authorize any person to receive, possess, deliver, use, or transfer source material.

(Sec. 161, as amended, Pub. L. 83-703, 68 Stat. 948 (42 U.S.C. 2201); sec. 201, as amended, Pub. L. 93-438, 88 Stat. 1243 (42 U.S.C. 5841))

[26 FR 284, Jan. 14, 1961, as amended at 43 FR 6923, Feb. 17, 1978]

§ 40.22 Small quantities of source material.

(a) A general license is hereby issued authorizing use and transfer of not more than fifteen (15) pounds of source material at any one time by persons in the following categories:

(1) Pharmacists using the source material solely for the compounding of medicinals;

(2) Physicians using the source material for medicinal purposes;

(3) Persons receiving possession of source material from pharmacists and physicians in the form of medicinals or drugs;

(4) Commercial and industrial firms and research, educational and medical institutions and Federal, State and local governmental agencies for research, development, educational, commercial or operational purposes;

And provided, That no such person shall pursuant to this general license receive more than a total of 150 pounds of source material in any one calendar year.

(b) Persons who receive, possess, use, or transfer source material pursuant to the general license issued in paragraph (a) of this section are exempt from the provisions of Parts 19, 20, and 21, of this chapter to the extent that such receipt, possession, use or transfer are within the terms of such

APPENDIX D - DISPOSAL OF PRODUCTS WITH REFUSE

D.1 INTRODUCTION

This appendix describes aspects of solid waste disposal as they relate to the fate of radioactive materials contained in products disposed of as refuse. Particular attention is paid to landfill disposal and incineration practices since they substantially influence the magnitude of the exposure of populations to released radioactivity. Unless specified otherwise, the models, assumptions, and data presented in this appendix are those used to calculate the radiological impact of the end of life or disposal phase of the product life span.

D.2 MUNICIPAL SOLID WASTE PRACTICES

Municipal solid waste from residential, commercial, and institutional sources amounted to 130 million tonnes in 1976 (1.8 kg per capita per day), and the yearly total is expected to increase to 180 million tonnes by 1985 (USEPA 78). The composition of municipal refuse varies widely, but a typical composite has been summarized in Table D.1. Most of these wastes are collected and transported to a processing or disposal site. Approximately nine percent of municipal refuse is incinerated, seven percent is recycled, six percent is disposed of in sanitary landfills, and less than one percent is used for producing compost. The largest fraction of municipal solid waste, approximately 78 percent of the total amount collected, is deposited in open dumps which do not meet the minimum qualifications of a sanitary landfill (USEPA 78, Mantell 75). A significant amount of the solid waste generated is not collected but rather is disposed of in apartment house, institutional or backyard incinerators, in sewer systems, or in unauthorized dumping areas. This is reflected by the fact that in certain urban areas the amount of waste collected per capita is significantly less than the estimated amount of per capita generation. Municipal waste accounts for about seven percent of the total volume of solid waste generated nationally. Agricultural wastes, such as animal carcasses, manure, and crop harvesting residues account for about 59 percent of the annual generation, and

Table D.1. Composition of Municipal Solid Waste[(1)]
Collected From Four Different Cities

Item	Percent Composition of Municipal Waste	
	Range	Average
Paper	40-54	45
Wood	1.5-3.5	3
Plastic	2-3	2.5
Glass	3.5-10	6
Metal	7-13	9
Stone, sand	7-8.5	7.5
Organics (food and yard wastes)	20-27	22.5
Rags	1.5-8	4.5

(1) USDHEW, 1968

mineral wastes, such as mill tailings and slag, represent 31 percent of the waste. The remaining three percent is due to waste generation in the manufacturing industry (Hickman 77). Although 70 to 80 percent of these industrial wastes are disposed of on the generator's property, a significant fraction will ultimately be deposited in landfills with municipal wastes. The individual steps in the practice of waste collection, processing, and disposal are discussed separately below with regards to their importance as pathways of exposure.

D.2.1 Collection

Approximately 175,000 people and 138,000 vehicles are directly involved in the practice of municipal solid waste collection in this country. Of these vehicles, approximately 78,000 are of the large, compactor variety while the remaining 60,000 are classified simply as vehicles other than compactor trucks. An additional 165,000 operators and 115,000 vehicles are involved in the collection of industrial wastes (USEPA 78, Mantell 75).

Since the gamma radiation levels from most products are low, the dose to personnel involved in the collection of refuse containing these products will also be low. In addition, considerable shielding would be provided by the steel walls of the truck bin as well as the contained refuse. Compaction of the trash could increase the margin of protection by increasing the degree of shielding and the distance between source and operator. Unless specified otherwise, the dose to the municipal waste collection population is assessed on the basis of the following assumptions:

1. All products containing radioactive material are discarded with municipal refuse, and are collected for disposal.
2. The average time that each collected product spends in a collection truck bin is two hours.
3. The average product-to-operator distance is two meters.
4. An average shielding factor of 0.5 is used to account for gamma-ray attenuation by the truck bin, other refuse, etc.
5. The waste collection population consists of 175,000 people and 138,000 vehicles, or an average of 1.3 persons per vehicle.

Disposal in Landfills

According to a 1976 EPA Report to Congress on the effects of waste disposal practices on ground water, municipal wastes in the U.S. are currently disposed of in about 18,500 land disposal sites, only about 20 percent of which are considered "authorized" (USEPA 77). About 20 sites are lined and only 60 or so have provisions for leachate control. Most of the 18,500 sites are open dumps which are unacceptable because of burning, water pollution, lack of daily cover, or some combination of these factors. Only about six percent meet the minimum requirements for being considered a sanitary landfill. Table D.2 summarizes the results of a national survey of land disposal site problems. Of the total sites surveyed, 93 percent had insufficient or no daily cover, 84 percent burned some of their waste, and 45 percent had an existing or potential water pollution problem. Most of these sites also received some industrial wastes.

The disposal of consumer products containing radioactive material in solid waste landfills can result in the exposure of population groups through direct radiation exposure, ingestion of contaminated ground water or food crops, and inhalation of resuspended radioactivity (Figure D.1). Due to the dilution of consumer product sources in large amounts of other waste and/or cover materials, the low levels of penetrating radiation emitted by product sources, and the absence of persons in close proximity to a landfill site for long periods of time, direct radiation should not be considered an important mode of exposure. The ingestion and inhalation pathways are assessed separately below.

Groundwater Contamination

Municipal waste disposal in landfills can result in groundwater contamination through the generation of leachate formed when water percolates through cover material and refuse. Precipitation which falls on the site can either run off, return to the atmosphere by evaporation or transpiration, or infiltrate the landfill. Surface runoff from surrounding areas, moisture contained in or produced by the decomposition of the refuse, and water entering the landfill from the bottom or sides also contribute to the generation of leachate. The problem of ground water contamination by landfill disposal practices is compounded by the frequent location of landfills or dumps at sites which are particularly susceptible to this type of contamination. Examples of such contamination-prone sites are marshlands, abandoned sand or gravel pits, old

Table D.2. A Summary of Problems Reported for 11,781 Unacceptable Land Disposal Sites⁽¹⁾.

Problem	Number of Sites	Percent
No cover	1,046	8.9
Burning	304	2.6
Water pollution	320	2.7
Burning and no cover	5,393	45.8
Water pollution and no cover	764	6.5
Burning and water pollution	199	1.6
Burning water pollution and no cover	3,755	31.8
Total	11,781	

(1) Mantell, 1975

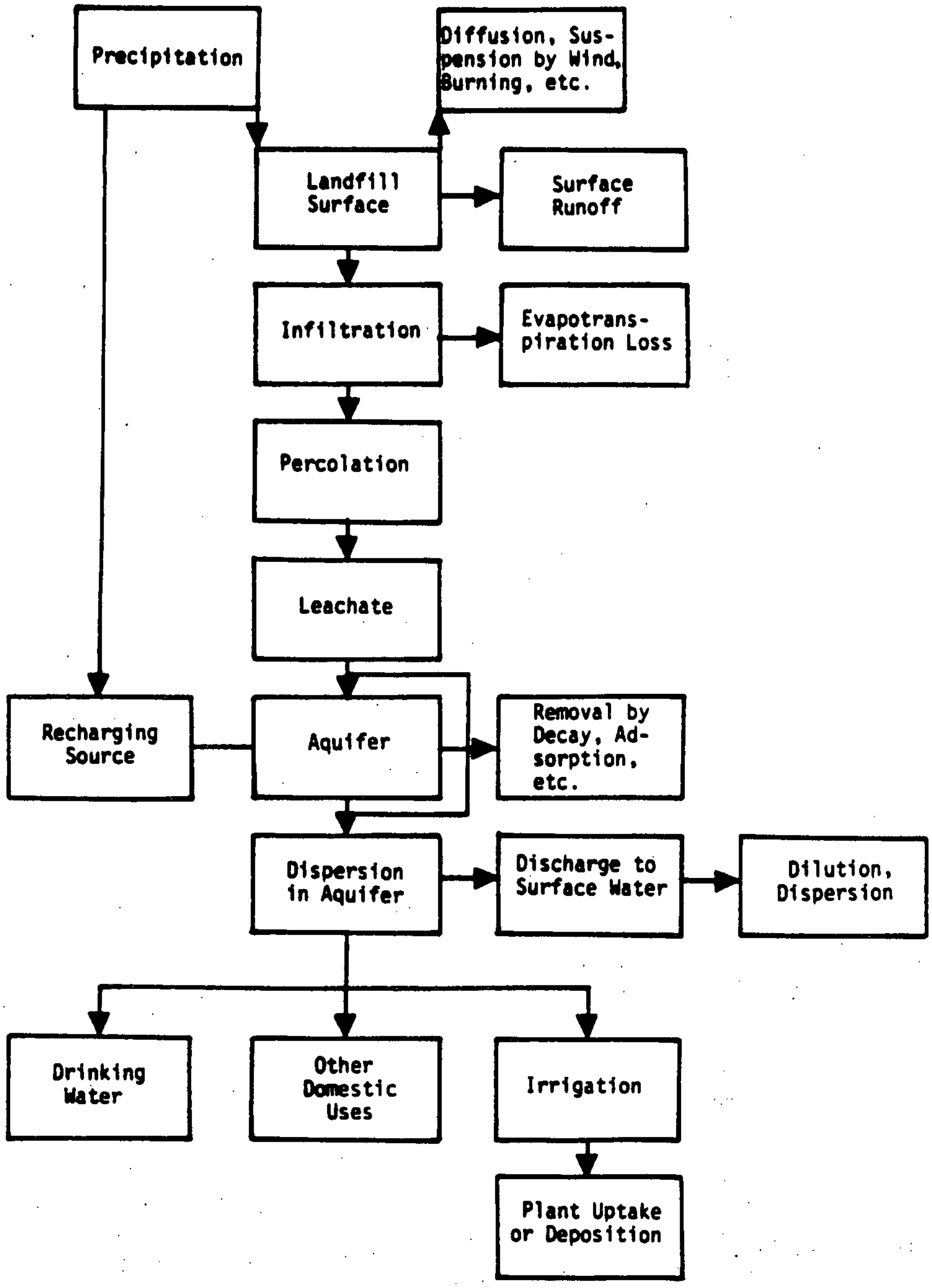


Figure D.1. Major Factors Influencing the Contamination of Air and Ground Water from Landfill Disposal of Radioactive Material in Consumer Products.

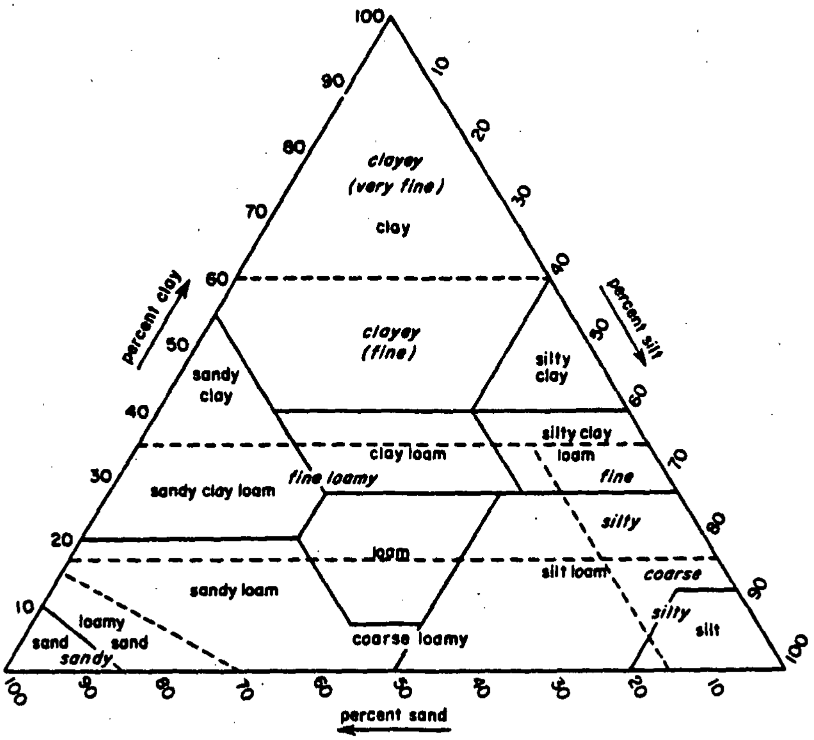
strip mines and limestone sinkholes. The EPA estimates that 70 percent of the municipal solid waste landfill sites are in ground water surplus areas, and that the average infiltration of precipitation in these areas is about 250 mm per year. Thus, assuming that the average site comprises an area of 25 acres, the total estimated leachate generation is 90 billion gallons per year, most of which enters the groundwater system (USEPA 77).

Contaminants that have entered the ground water can move horizontally or vertically, depending on the comparative density and natural flow pattern of the water in the aquifer. The contaminants do not mix readily with native water but tend rather to move as a well-defined slug or plume. The concentration of contaminants in ground water is usually reduced with time and distance by such mechanisms as adsorption, ion exchange, dispersion, and decay. These factors are discussed below.

In order for ground water to become contaminated, the following conditions must exist: (1) the waste material must be in hydraulic communication with an aquifer, (2) the soil and waste material must be totally saturated and possess a positive hydraulic potential, (3) leached fluids must be produced and this leachate must be capable of entering an aquifer, and (4) the contaminant of interest must be capable of being transported by the leachate (Pohland 75). The three major components of the ground water pathway are the leaching of the contaminant, the vertical movement of the contaminant through the waste material and unsaturated zone soil columns, and the horizontal movement of the contaminant in the aquifer.

The soil should be thought of as a complex, dynamic, biological, chemical and physical system which transforms all material, including waste materials, with which it remains in contact. Some waste material components will be permanently incorporated into the system, while others pass through and eventually enter ground water or are expelled to the atmosphere as gases. The rate at which contaminants migrate through soil is dependent on a number of factors, the most important of which include particle size distribution, pore size distribution (void fraction), hydrogen ion concentration (pH), chemical composition and ion exchange capacity of the soil, and climate.

Since many contaminant attenuation mechanisms involve physical and chemical reactions on soil particle surfaces, the potential for attenuation is greatest in soils containing smaller particles. Finer soil materials, such as silts, clays, and colloids (see Figure D.2), have greater surface area per unit



(1) Fuller, 1977

Figure D.2. Textural Classification of Soil Based on the Percentage of Different Sizes of Particles It Contains.(1)

weight and, in general, can be considered as having greater attenuating characteristics than coarser materials such as sands or gravels. It should be kept in mind, however, that soil contains a variety of organic and inorganic colloidal species which vary widely with respect to ion exchange capacities and ability to hold the trace contaminants against removal by solutions of salts, acids, alkalies, etc. For example, certain colloidal hydroxy oxides react strongly with most of the trace contaminants in soil and retain them against ion exchange much more tenaciously than the clay minerals. It would be very advantageous, therefore, to locate disposal sites on fine textured soils containing significant amounts of hydroxy oxides. Alternatively, disposal sites should be lined with such soils or with clay minerals where practical (Fuller 77).

The pore size distribution of soils is the volume of the various sizes of pores or voids in a soil expressed as a percentage of the bulk volume (i.e., soil particles plus voids). Although fine textured soils generally have a greater total volume of pore space than coarse-textured soils, the pores in fine-textured soils are usually much smaller. Some clay soils can impede water from vertical flow almost completely because of the very small pore sizes, whereas sandy soils tend to transmit water rapidly. Soils with smaller diameter pores restrict the migration of trace contaminants by impeding the rate of water flow through the soil which, in turn, allows more time for the contaminants to physically or chemically react with the soil particles (see Table D.3). The total pore size in any soil can be reduced by disturbing and compacting the soil as deeply as practical. The "effective porosity" of a soil medium is an empirically derived value frequently used in estimating contaminant flow in saturated zones.

The term "sorption" has often been used to describe, in a nonmechanistic way, a reaction in which a species is removed from a liquid phase and held in some manner by a solid. The term "ion exchange" is generally reserved for reactions which are reversible and, as such, serve to retard but not to stop the migration of the contaminants. When ion exchange occurs under equilibrium or near equilibrium conditions, a useful parameter termed the distribution coefficient, K_d, may be used. The distribution coefficient of a species A may be defined as (Kaufman 63):

Table D.3. Relative Immobilizing Capabilities of Soils for Selected Elements in Municipal Landfill Leachates Under Anaerobic Conditions(1)

Mobility Class	Texture	Soil Series	Order
Strongly immobilizing	Clay	Molokai	Oxisol
	Silty clay	Nicholson	Alfisol
Usually strongly immobilizing	Silty clay Loam	Chalmers	Molisol
	Clay	Davidson	Ultisol
	Clay	Fanno	Alfisol
	Sandy clay Loam	Mojave (limy)	Aridisol
Moderately immobilizing	Silty clay Loam	Ava	Alfisol
	Sandy loam	Mojave (lime) free)	Aridisol
Weakly immobilizing	Sandy loam	Anthony	Entisol
	Sand	Kalkaska	Spotosol
	Loamy sand	Wagram	Ultisol

(1)Fuller, 1977

$$K_d = \frac{\left(\text{weight of A} \atop \text{in solid phase}\right) \Big/ \left(\text{weight} \atop \text{of solid}\right)}{\left(\text{weight of A} \atop \text{in liquid phase}\right) \Big/ \left(\text{volume} \atop \text{of liquid}\right)}$$

It can be shown mathematically (Levy 72) that the the retardation factor, K, which is the ratio of the average velocity of the ground water, V_w, to the average velocity of the contaminant, V_c, is described by:

$$K = 1 + \rho K_d/\theta$$

where

ρ = bulk density of the medium (g/cm^3)

K_d = distribution coefficient of species A in the medium (ml/g)

θ = effective porosity of the medium (i.e., the fraction of the total volume of a given mass of medium consisting of interconnecting interstices)

Values for K_d can be measured relatively easily in the laboratory. These values, however, will vary widely with the composition of the liquid phase, the chemical and physical nature of the solid phase, the temperature, etc. For example, Table D.4 illustrates a very wide range of K_d values measured for cesium under different soil and solution conditions. Thus, the value of K_d used in a given assessment should be one determined under conditions closely approximating those of the situation being assessed. Borg and co-authors (1976) have discussed the difficulties involved in trying to duplicate field conditions in the laboratory. In general, although it is much easier to measure K_d values in the laboratory than in the field, it is difficult to set conditions for the laboratory experiment such that its results will mirror those that would be obtained in the field. Values for K_d used in this assessment are presented in Table D.5.

Another important parameter concerning the movement of contaminants in ground water is the degree to which the contamination is dispersed both longitudinally (in the direction of ground water flow) and transversely. The dispersion coefficient of an aquifer is a measure of the actual dispersion as determined by aquifer permeabilities and hydraulic gradients. The

Table D.4. Distribution Coefficients for Cesium in Various Media(1).

Medium	Particle Size	K_d (ml/g)	Reference
Alluvium, Central Nevada	500-4000 µm	121-3165	Nork, 1971
Tuff, NTS(2)	<400 µm	1020	Nork, 1970
Tuff, NTS	100-200 mesh(3)	12,100-17,800	Goldbert, 1962
Granodiorite, NTS	100-200 mesh 0.5-1mm	8-9(4) 1030-1810	Beetem, 1962
Basalt, NTS	32-80 mesh	792-9520	Angelo, 1962
Pillow basalt, Amchitka	500-4000 µm	39	Essington, 1969
Basalt, Amchitka	500-4000 µm 500-400 µm	280(5) 6.5(6)	Essington, 1969
Sediments under Hanford waste beds	ns(7)	300	Brown, 1967
Sands and sandy Shales	ns	250->1000	Inoue, 1967
Quartz sand, Mol, Belgium	ns	22-314	Baetslé, 1962
Tuff	ns	800-1000	Dloughy, 1967
Silty clay, Silty clay loam	ns	617-1053	Godse, 1967
Soil, Hanford N- Area	ns	189-420(8)	Hajek, 1967

(1) Summarized from Borg, 1976

(2) NTS = Nevada Test Site

(3) 100 mesh = 0.01 inches or 0.25 mm

(4) The wide range of values is related to slight differences in prepared solutions, which were meant to simulate groundwaters.

(5) 310 ppm dissolved solids in prepared water solution

(6) 310 ppm dissolved solids in seawater solution

(7) ns - not specified

(8) Range due to differences in prepared solution

Table D.5. Distribution Coefficients and Retardation Factors.

Radionuclide	K_d (ml/g)			Retardation Factor (K_i) for Reference Soil (2)	Reference
	Sand	Silt and Clay	Reference Soil (1)		
H-3	0	0	0	1	Staley, 1977
C-14	0	0	0	1	-
Co-60	100	1000	640	2270	Staley, 1977
Ni-63	100	1000	640	2270	Staley, 1977
Kr-85	0	0	0	1	Scrudato, 1973
Cs-137	(3)	(3)	300	1035	-
Pm-147	-	-	400	1420	Essington, 1969
Po-210	-	-	200	710	Hansen, 1971
Ra-226	-	-	140	500	Lester, 1975
Th-232	-	-	14,125	50,000	Lester, 1975
U-238	-	-	3,955	14,000	Lester, 1975
Am-241	-	-	2,825	10,000	Lester, 1975

(1) Reference soil consists of 40 percent sand, 60 percent silt and clay, and has a bulk density of 1.7 g/cm^3, a void fraction of 0.48 and a pH of 6.3 (summarized from Table 3.1 of Fuller, 1977).

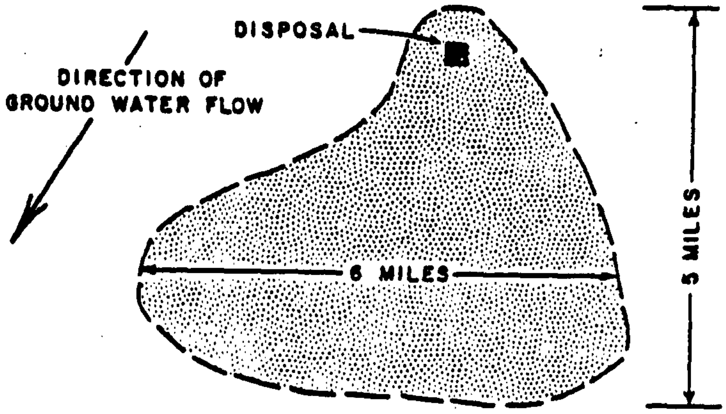
(2) Retardation factor for reference soil = 1 + 3.54K_d.

(3) See Table D.4.

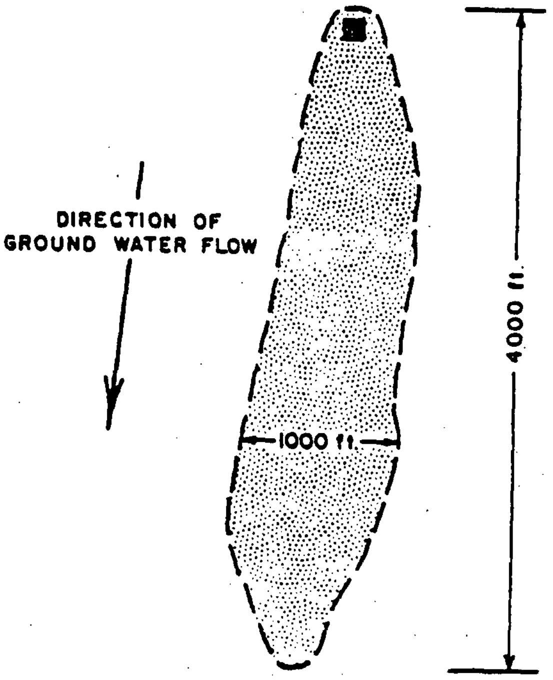
U. S. Geological Survey has determined that, under a gradient of 2 m/km, the rate of ground water flow can vary from 18 m/day in gravel to 0.3 m in 30,000 years for clay (USEPA 77). A typical aquifer will have a rate of water flow between 0.16 to 1.6 km per year (Fried 76). The transverse dispersivity, which may be described as the inherent capability of the aquifer to cause dispersion in a transverse direction, is dependent primarily on the complexity (on a microscopic level) of the paths taken by the fluid, and (on a macroscopic level) the inhomogeneities within the aquifer. Figure D.3 depicts two examples of contamination plumes in aquifers with different transverse dipersivities. Since contamination plumes are observed to travel in a rather well-defined manner, it is evident that dilution by the surrounding body of ground water is not a major factor. Although dispersivity tends to significantly reduce the contaminant concentration at the point of withdrawal, this assessment will neglect this effect. Instead, it will be assumed that one-half of the ground water withdrawal points will be in the general direction of the ground water flow (i.e., downstream from the landfill), and that the contamination plume is confined and migrates directly toward the well. This will more than compensate for the tendancy of the well to attract surrounding ground water as a result of withdrawal.

The model landfill used in this assessment is illustrated in Figure D.4. The total depth of the fill is assumed to be 15 meters, a value which is representative of fills in areas where rainfall and infiltration are sufficient to produce an abundance of leachate (U. S. Army 76, Fenn 75). There are, of course, landfill sites which are less than 15 meters deep. The depth of landfills in most of the eastern coastal United States, for example, are limited by the proximity of the water table to the ground surface. In such cases, the surface area required for a typical site would be proportionately greater. The volume of leachate produced in such shallow fills would be greater, and this leachate would reach the saturated zone sooner, than the reference site. The concentration of the contaminant in the leachate, however, would be considerably lower than that of the reference site. Table D.6 lists some examples of landfill site characteristics.

Fenn, et al. (1975) have discussed the use of the water balance method for predicting leachate generation from solid waste disposal sites. The infiltration of precipitation into the soil cover and subsequent percolation down to and through the solid waste layer is the principal mechanism responsible for



a) CHLORIDE PLUME, INEL, IDAHO
Transverse dispersivity: 450 feet
Time: 16 years



b) CHROMIUM PLUME, LONG ISLAND
Transverse dispersivity: 14 feet
Time: 13 years

Figure D.3. Examples of Transverse Dispersion Patterns of Contamination Plumes (From USEPA, 1977).

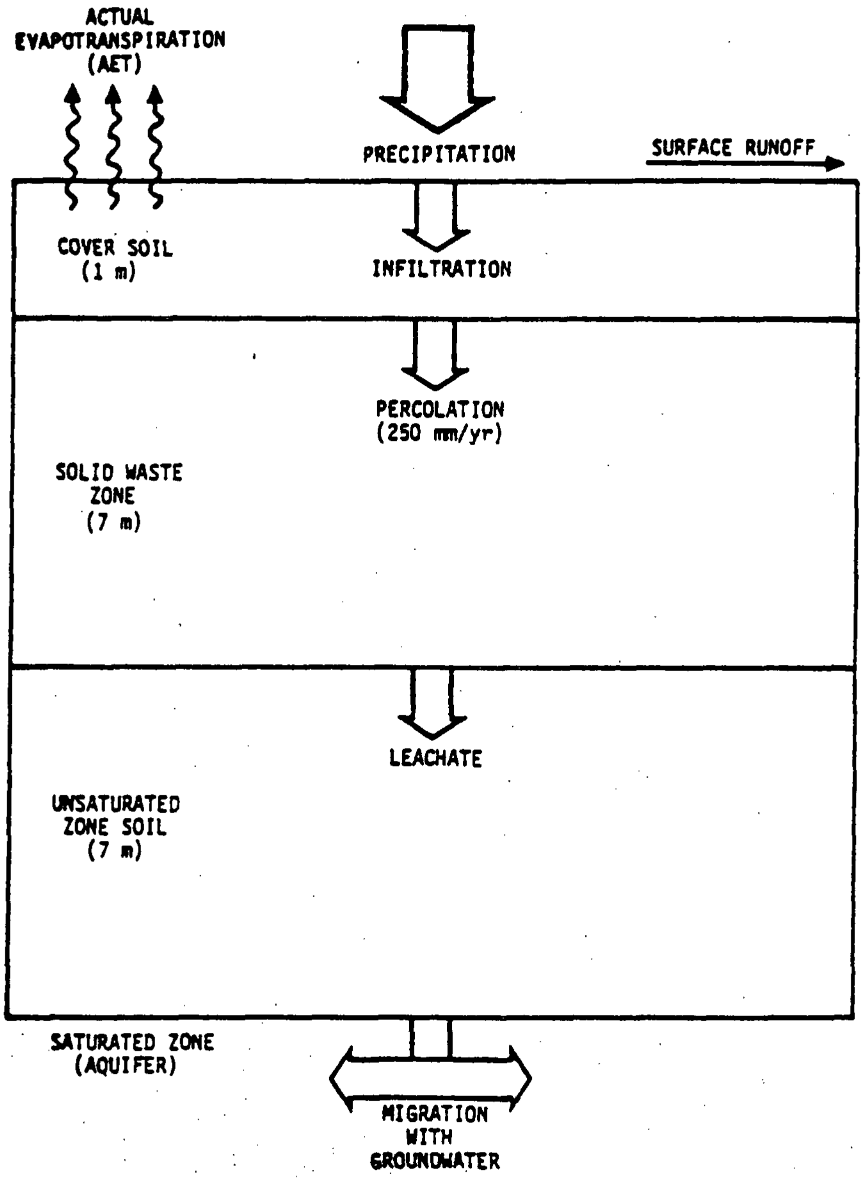


Figure D.4. Reference Landfill Disposal Site.

Table D.6. Summary of Characteristics of Representative Landfills$^{(1)}$.

Location	Mean Annual Precipitation (mm)	Mean Annual Percolation (mm)	Average$^{(2)}$ Depth (m)	Type of Fill
Ohio	1025	210	15	Trench and Fill
Florida	1340	70	8	Surface Fill
Southern California	380	0	40	Canyon Fill
North Central U. S.	740	ns$^{(3)}$	26	Trench and Fill
South Central U. S.	1240	ns	6	Surface Fill
East Central U. S.	1040	ns	25	Surface Fill

(1) Fenn, 1975, U. S. Army, 1978

(2) Average depth of cover material, waste, and unsaturated zone

(3) ns = not specified

leachate generation. Other factors, such as water of decomposition, the initial moisture content of the waste, and infiltration of ground water, can also influence the amount of leachate produced. These factors, however, will be assumed to be insignificant contributors for the reference case.

The water balance method is based on the relationship among precipitation, surface runoff, actual evapotranspiration (AET), and soil moisture storage. Using representative data for these parameters, Fenn, et al. were able to estimate both the time of first appearance of leachate for landfills of various depths (Figure D.5) and the annual leachate production rate for landfills of various areas (Figure D.6). A summary of values assumed as representative for all of the important reference case parameters is contained in Table D.7.

The model used to assess the ground water ingestion pathway considers two distinct components: vertical migration of leached radionuclides through the unsaturated zone and horizontal migration in the saturated zone or aquifer. The time, t_m, required for the activity to migrate to a ground water withdrawal point is given by:

$$t_m = t_u + t_s$$

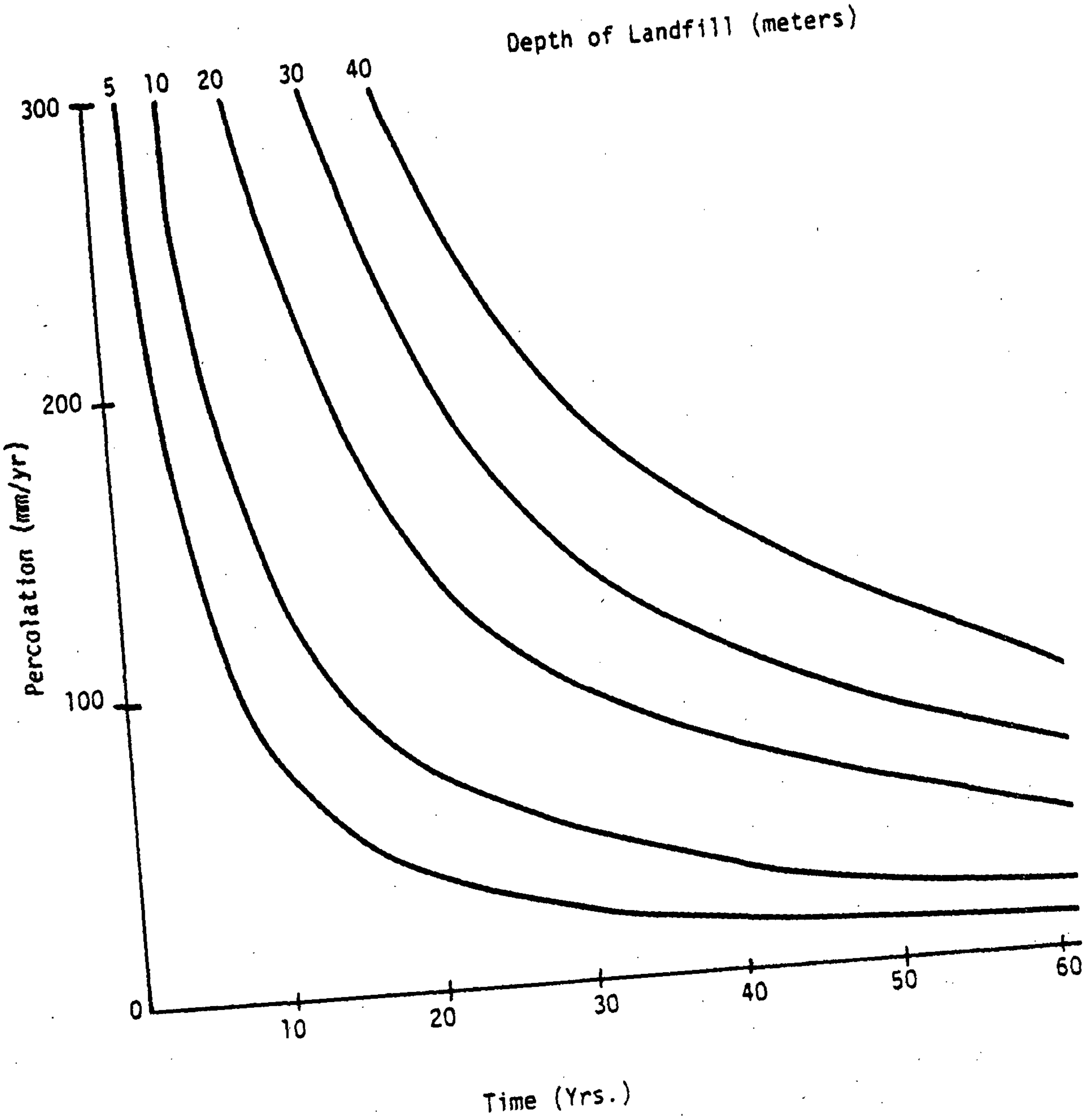
where

t_u = time of first appearance of radionuclide at the upper boundary of the aquifer (years)

t_s = time required for radionuclide to migrate horizontally in the aquifer to a ground water withdrawal point (years)

The time of first appearance of leachate at the aquifer boundary, t_ℓ is found from Figure D.5. The time of first appearance of radionuclide i at this point, t_{ui}, is related to t_ℓ by the relative rate of radionuclide movement to ground water movement (i.e., the equilibrium constant, K_i):

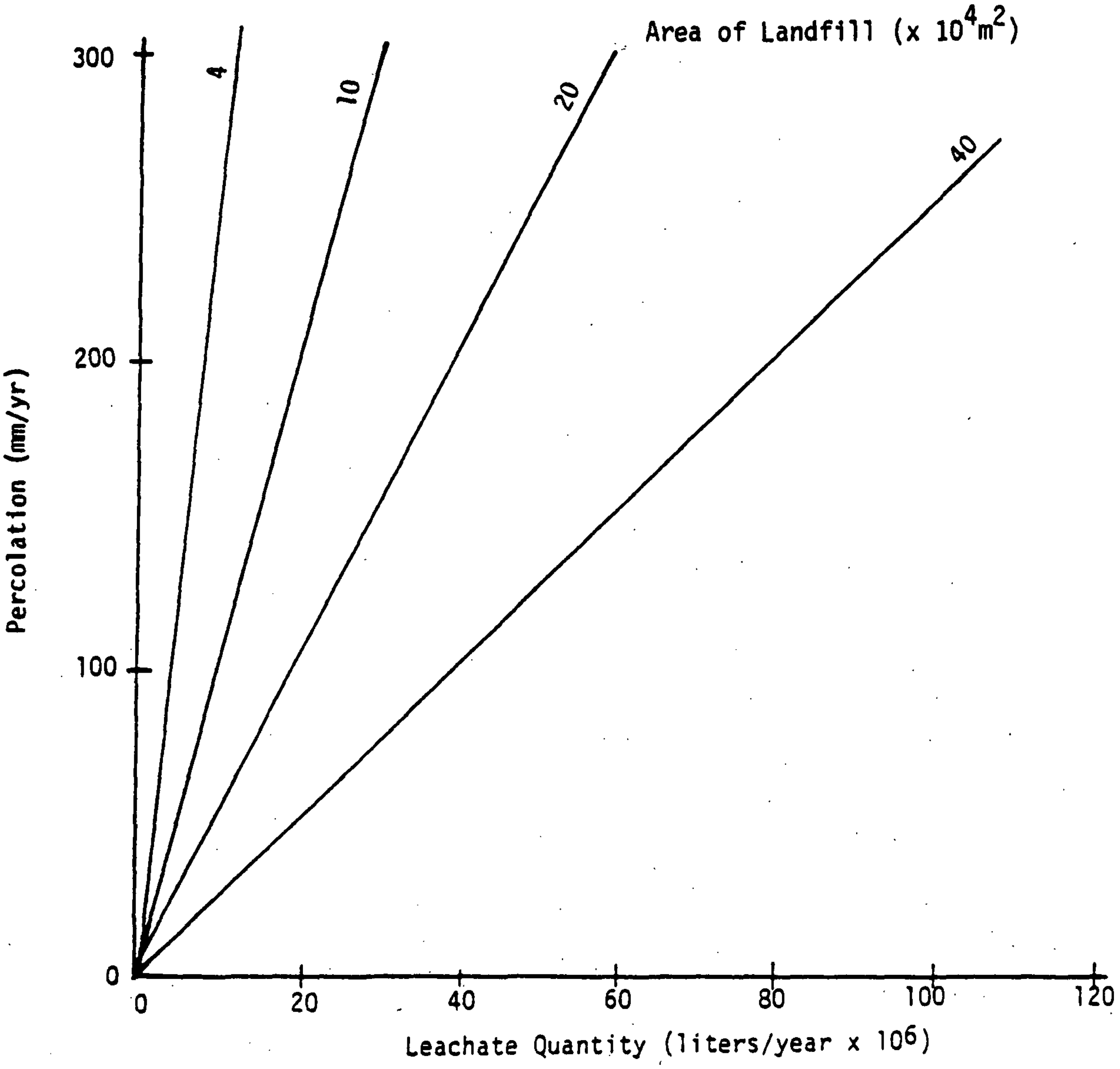
$$t_{ui} = t_\ell K_i$$



(1) Based on a solid waste moisture absorption capacity of 150 mm/m.

(2) From Fenn, 1975.

Figure D.5. Time of First Appearance of Leachate for Landfills of Different Depths(1,2).



(1) From Fenn, 1975

Figure D.6. Annual Leachate Quantities After Time of First Appearance(1).

Table D.7. Summary of Assumed Conditions at Reference Landfill Site.

Area	25 acres ($\sim 10^5 m^2$)
Depth	
Soil cover	1m
Waste layer	7m
Unsaturated zone	7m
Soil	
Composition	40 percent sand, 30 percent silt, 30 percent clay
Bulk density	1.7 g/cm^3
Void fraction	0.48
Annual infiltration(1)	250mm
Groundwater velocity	900 m/yr
Fraction of wells in general direction of groundwater flow	0.5
Average distance to well	1000m
Total number of sites in water excess areas	13,000

(1) Infiltration is assumed to equal percolation

The time required for radionuclide i to migrate horizontally in the aquifer to the point of ground water withdrawal, t_{si}, is given by:

$$t_{si} = dK_i/V_g$$

where

d = distance to point of ground water withdrawal (m)

V_g = ground water velocity (m/yr)

It will be assumed that all activity leached in one year will be withdrawn over a similar interval, t, following the time of first appearance at the well. The amount of activity that is ingested, A_{ing}, is given by:

$$A_{ing}(t_m + t) = f_m f_{ing} A_\ell(t) e^{-\lambda t_m}$$

where

f_m = fraction of activity that migrates toward a ground water withdrawal point (0.5)

f_{ing} = fraction of withdrawn ground water which is ingested as drinking water (0.05)

$A_\ell(t)$ = amount of activity leached from product during year t (Ci/yr).

λ = decay constant of radionuclide (yr^{-1})

The rate at which activity is leached from the products is assumed to remain constant. However, since approximately ten percent of municipal solid waste is incinerated with subsequent landfill disposal of the residue, there will be two separate components of the leach rate, one for intact product sources and one for previously incinerated sources:

$$A_\ell(t) = N_o A_o \left[f_t R_t e^{-(\lambda + R_t)t} + f_c R_c e^{-(\lambda + R_c)t} \right]$$

where

N_o = total number of products disposed of at time 0

A_o = average activity per product (Ci)

f_t = fraction of products disposed of intact

f_c = fraction of products previously incinerated

R_t = leach rate of activity from intact products (fraction/yr)

R_c = leach rate of activity from previously incinerated products (fraction/yr)

If it is assumed that there is no delay between the time of arrival at the well and the time of consumption, the total activity that is consumed in drinking water over all time, A_{tot}, is given by:

$$A_{tot} = \int_0^\infty A_{ing}(t_m + t)dt$$

$$= \int_0^\infty f_m f_{ing} A_\ell(t) e^{-\lambda t_m} dt$$

$$= f_m f_{ing} N_o A_o e^{-\lambda t_m} \int_0^\infty \left(f_t R_t e^{-(\lambda + R_t)t} + f_c R_c e^{-(\lambda + R_c)t} \right) dt$$

$$= f_m f_{ing} N_o A_o e^{-\lambda t_m} \left[\frac{f_t R_t}{\lambda + R_t} + \frac{f_c R_c}{\lambda + R_c} \right]$$

For the reference case, this reduces to:

$$A_{tot} = 0.025 N_o A_o e^{-\lambda t_m} \left[\frac{0.9 R_t}{\lambda + R_t} + \frac{0.1 R_c}{\lambda + R_c} \right]$$

In order to illustrate the method used to assess the activity ingested by way of ground water contamination and subsequent consumption, a sample calculation is presented below.

Example: One million units of a particular consumer product containing one mCi of tritium each are distributed annually. (Total tritium distribution is 1000 Ci). What would be the total ingested activity resulting from the disposal of these units?

Assumed parameters:

$$R_t = 0.01/\mathrm{yr}$$
$$R_c = 1.0/\mathrm{yr}$$
$$\lambda = 0.0564\ \mathrm{yr}^{-1}$$

From Table D.5:

$$K_{H-3} = 1$$

From Table D.7:

$$V_g = 900\ \mathrm{m/yr}$$
$$d = 1000\ \mathrm{m}$$
$$D = 15\ \mathrm{m}$$

From Figure D.5:

$$t_\ell = 8\ \mathrm{yr}$$

Therefore,

$$t_m = t_u + t_s$$

$$= t_\ell K_{H-3} + dK_{H-3}/V_g$$

$$= (8\ \mathrm{yr})(1) + (1000\ \mathrm{m})(1)/(900\ \mathrm{m/yr})$$

$$= 8\ \mathrm{yr} + 1.1\ \mathrm{yr}$$

$$= 9.1\ \mathrm{yr}$$

$$A_{tot} = 0.025\ N_o A_o e^{-\lambda t_m}\ \frac{0.9R_t}{\lambda + R_t} + \frac{0.1R_c}{\lambda + R_c}$$

$$= (0.025)(1000\ \text{Ci})(0.6)(0.136 + 0.095)$$

$$= 3.5\ \text{Ci}$$

Since there are over 18,500 landfill sites in the U. S. and since these sites can vary widely with respect to potential for ground water contamination, a sensitivity analysis of the various associated parameters is warranted. Of primary concern is the degree of impact resulting from variations in the values used for radionuclide leach and migration rates (R_t, R_c, and K_i), ground water velocity (V_g), distance to nearest well (d), and landfill site characteristics such as depth (D) and annual percolation (P). In order to estimate the incremental increase in radiological impact which would result from the use of values typical of contamination prone sites, a "worst case" will be considered for each product using the following values:

R_t and R_c = twice or ten times the reference case values

K_i = one tenth the reference case value

V_g = 1600 m/yr

d = 100 m

D = 0

P = 250 mm (same as reference case)

D.2.3 Incineration

Approximately 13 million tonnes of municipal solid waste are incinerated annually in this country. Most of this incineration occurs in the 300 or so large municipal incinerators currently operating. A significant amount also occurs in thousands of intermediate or apartment complex incinerators which are prevalent in many large urban areas, such as New York and Philadelphia. The relative importance of conventional incineration as a means of waste processing in the near future is questionable at this time. Stricter air quality standards and increased capital and operating costs are combining to drive up the overall cost to prohibitive levels. One review recently reported that, despite the increased sophistication of present-day incinerators in the U. S., none is apparently operating in a wholly satisfactory manner (Wilson 77).

The primary function of an incinerator is to reduce the municipal waste volume to a compact, sterile residue. Some incinerators also have secondary functions, such as the generation of steam or some other utilization of waste heat. Non-conventional incineration methods, such as pyrolysis, result in the production of a storable, transportable fuel from municipal solid waste. These methods, however, are still in early developmental stages in this country.

In conventional incinerators, refuse is either pretreated for size reduction (e.g., shredding or grinding) or fed directly into the combustion chamber via some continuous-feed or batch-feed process. The refuse is oxidized in the combustion chamber in which the temperature is usually regulated by using excess air (i.e., air that is in excess of the amount theoretically required to completely burn the combustible portion of the refuse). Typical furnace gas temperatures are controlled to around 1000-1100°C, although the actual temperature of the flame is approximately 1300°C. These conditions will usually result in reducing the bulk by oxidation to 20-25 percent of the original weight and 10-12 percent of the original volume. The residue which remains consists mostly of metals, glass, ceramics, stones, ash, and unburned combustibles. If temperatures in the furnace are maintained at about 1650°C or more, the residue will be reduced to a molten state called slag. Incinerator residue is essentially sterile and compacts well. Thus, if used for landfill, the land can be immediately reclaimed. Alternatively, the metallics in the residue can be separated out for salvage and the residual ash can be used as a valuable substitute for sand and aggregate in building or road construction (Mantell 75).

Most large incinerators require water in various parts of the incineration process (e.g., gas scrubbing and residue quenching). The water usually becomes contaminated with dissolved or suspended matter and in most cases requires treatment before discharge to prevent or control pollution of surface or ground water systems. The most serious environmental problem associated with incinerators, however, is pollution of the air.

When refuse is burned, large quantities of fine ash particles, water vapor, carbon dioxide, and other gases are released. Recently constructed incinerators control the amount of particulate emissions by means of high efficiency combustion and particulate collection systems, such as electrostatic precipitators and gas scrubbers. Table D.8 shows the particle size distribution obtained from measurements of stack gases in a large municipal incinerator. Particle size and other physical properties of the particulate emissions, such as density, depend on many factors, such as refuse composition, incinerator design and operation, and air pollution control systems. The particle size distribution in Table D.8, for example, is typical of stack gases that have passed through a cleanup system consisting of a combustion settling chamber and a wet baffle system. The density of these particles is about 1.85 g/cm^3.

The concentration of particulates in stack gases and the total amount of particulate releases for several different types of large incinerators are given in Table D.9. The particle concentration or "dust loading" is expressed in limits of grains (7000 grains equals one pound) per standard cubic foot of gas normalized to 12 percent CO_2.

Various processing systems have been used to remove particulates and other pollutants from incinerator effluents. Older incinerators used such methods as settling chambers, wet baffle systems, or cyclone collectors. Collection efficiencies for these systems vary from around 35 to 75 percent, an insufficient amount of removal for compliance with current emission standards. Efficiencies of 90 to 99 percent are available in currently used systems, such as gas scrubbers and electrostatic precipitators. Some average removal efficiencies for various types of air pollution control systems are given in Table D.10. It is noteworthy that about 70 to 75 percent of the incinerators in the U. S. were built before 1960, with many dating back to the 1940's and beyond (Rubel 74). These incinerators require air pollution control retrofitting in order to comply with current regulations.

Three reference cases are assessed here concerning the environmental health impact of incineration of discarded consumer products:

1. The calculated dose to an individual at the point of maximum exposure to the emission plume from a large municipal incinerator.
2. The calculated dose to the population feeding one incinerator disposal route.
3. The calculated dose to the total U. S. population residing in the environs of large municipal incinerators.

Table D.8. Size Distribution of Particles in Incinerator Stack Gas.(1)

Size(μm)	Percent by Weight
>30	31.3
>20	52.8
>10	79.5
>5	94.0
<5	6.0

(1) From Mantell, 1975

Table D.9. Particulate Emission for Incinerators of Various Designs(1).

Incinerator	gr/SCF at 12 percent CO_2	lb/ton of Waste Charged	Gas-Flow Rate (ft^3/min)
A	0.55	10.4	69,800
B	1.12	14.5	131,000
C-1	0.56	4.1	3,890
C-2	0.41	3.4	3,990
C-3	0.30	2.9	4,460
D	0.46	8.8	120,000
E	0.73	8.6	186,000
F	0.72	12.5	165,000
G	1.35	20.4	130,000

---

(1) From Achinger, 1970

Table D.10. Average Air Pollution Control System Removal Efficiencies (Weight Percent)(1).

Air Pollution Control System	Mineral Particulates	Combustible Particulate	Polynuclear Hydrocarbons	Volatile Metals
None (flue settling only)	20	2	10	2
Dry Expansion Chamber	20	2	10	0
Wet Bottom Expansion Chamber	33	4	22	4
Spray Chamber	40	5	40	5
Wetted Wall Chamber	35	7	40	7
Wetted, Closed-Spaced Baffles	50	10	85	10
Mechanical Cyclone (dry)	70	30	35	0
Medium-Energy Wet Scrubber	90	80	95	80
Electrostatic Precipitator	99.9	99	67	99
Fabric Filter				

---

(1) From Niessen, 1970

Unless specifed otherwise, the assumptions used to estimate the amount of radioactivity in incinerator emissions and resultant exposures are as follows:

1. There are a total of 300 municipal incinerators, each processing 300 tonnes of refuse per day at 50 percent excess air.
2. The number of products incinerated per year is ten percent of the total number disposed of in peak years.
3. The efficiency of installed air pollution control systems for radioactive particulates is 90 percent.
4. The aerodynamic mean activity diameter (AMAD) of the released particulates is one micron.
5. The number of persons feeding one incinerator disposal route (assumed to be the exposed population) is (2.2×10^8) (0.1)/(300) = 73,000 persons. (U. S. population is 2.2×10^8).
6. The average daily breathing rate is 20 m^3/day.

The total activity, Q, released in a year would be:

$$Q = Q_i f_s f_r$$

where

Q_i = initial product activity (Ci)

f_s = fraction of activity released by the incinerator process

f_r = fraction of released activity which escapes with stack emissions

The concentration of activity, χ_s, in the stack gas would be:

$$\chi_s = Q/V_a W_r$$

where

V_a = 50 percent excess of the theoretical volume of air required for complete combustion of one pound of refuse $(2.0\times10^6\ cm^3/lb)$

W_r = weight of refuse incinerated annually $(2.4\times10^8$ lb)

The average ground level concentration, χ, resulting after dispersion of radioactive effluents in the atmosphere is calculated by:

$$\chi = Q' \, (\chi/Q)$$

where

Q' = average release rate (Ci/s)

χ/Q = atmospheric dispersion coefficient (s/m^3)

Values for χ/Q are obtained from the data of Martin (1968) and Hilsmeier and Gifford (1962). Assuming an effective stack height of 30 meters, a Pasquill stability category D (neutral conditions), and an average wind speed of two meters per second, these values are:

(1) $6\text{x}10^{-5}$ s/m^3 for annual average ground-level concentration at point of maximum concentration.

(2) $3\text{x}10^{-5}$ s/m^3 for annual average ground-level concentration at point of average concentration.

The former value is used for estimating the annual dose to the maximally exposed individual while the latter is used for estimating the dose to the population in the vicinity of the incinerator. An example calculation follows.

Example: What is the total inhaled radioactivity resulting from the incineration of products containing a total of 100 Ci of Pm-147 at the time of disposal?

Assume:

$$f_s = 1.0$$
$$f_r = 0.1$$

The total activity released annually would be:

$$Q = Q_i f_s f_r$$

$$= (100\ \text{Ci/yr})(1.0)(0.1)$$

$$= 10\ \text{Ci}$$

Assuming equal distribution of the products at 300 municipal incinerators, the release rate per incinerator is:

$$\begin{aligned} Q' &= Q/300 \\ &= (10\ \text{Ci/yr})/300 \\ &= 0.033\ \text{Ci/yr} \end{aligned}$$

The average ground level concentration of Pm-147 at point of maximum concentration would be:

$$\begin{aligned} \chi &= Q'\ (\chi/Q) \\ &= (0.033\ \text{Ci/yr})(3.17\text{x}10^{-8}\ \text{yr/s})(6\text{x}10^{-5}\ \text{s/m}^3) \\ &= 6.3\text{x}10^{-14}\ \text{Ci/m}^3 \end{aligned}$$

The amount of activity inhaled would be:

(a) Maximum individual

$$\begin{aligned} A &= (20\ \text{m}^3\text{/d})(365\ \text{d/yr})(6.3\text{x}10^{-14}\ \text{Ci/m}^3) \\ &= 4.6\text{x}10^{-10}\ \text{Ci} \end{aligned}$$

(b) Average individual (ratio of χ/Q's)

$$\begin{aligned} A &= (4.6\text{x}10^{-10}\ \text{Ci})(3\text{x}10^{-5}\ \text{s/m}^3)/(6\text{x}10^{-5}\ \text{s/m}^3) \\ &= 2.3\text{x}10^{-10}\ \text{Ci} \end{aligned}$$

(c) Population feeding one incinerator

$$\begin{aligned} A &= (2.3\text{x}10^{-10}\ \text{Ci/person})(73{,}000\ \text{persons}) \\ &= 1.7\text{x}10^{-5}\ \text{Ci} \end{aligned}$$

(d) Total U. S. population

$$A = (1.7\times10^{-5}\ \text{Ci})(300)$$
$$= 5.1\times10^{-3}\ \text{Ci}$$

For reasons of economy, municipal incinerators are usually located in or near densely populated areas. The contamination and subsequent ingestion of food crops will, therefore, be of less significance than the inhalation exposure pathway. It has been previously estimated that the incineration of ionization chamber smoke detectors containing Am-241 results in an ingestion population dose which is about two orders of magnitude less than the inhalation dose (Belanger 79). For this and other reasons (most importantly, the unavailability of required data), no attempt to calculate the radiological impact resulting from the incineration-ingestion pathway has been made in this report.

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The NRC is reevaluating the adequacy of existing policy dealing with the distribution of consumer products containing radioactive material. As part of this reexamination a generic environmental impact statement (GEIS) will be prepared.

This report provides assessments of the impact of several categories of consumer products on people and the environment including the benefits and risks of possible alternatives. In general, the report concludes that the radiological impact of all the categories is very low; however, in the case of glassware and ceramic table-ware, the report concludes that radioactivity in this product is unnecessary and unwarranted.

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