

THE CONTINUING PROBLEM OF RADIOACTIVE METAL SCRAP

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PA Department of Environment Resources
and
US Nuclear Regulatory Commission

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ABSTRACT

Metal scrap found to contain radioactive materials continues to challenge regulatory agencies as discoveries of this unwanted constituent increase. And while efforts are made to prevent the exposure of personnel at metal manufacturing mills and scrap yard when radioactivity is discovered in a shipment of metal scrap, this has not stemmed the number of discoveries. Sources and devices continue to be found, leading to difficulties in the disposal of the radioactive materials, especially with the closure of licensed LLRW facilities to non-compact state members. Naturally-occurring radioactive materials continue to be found, principally as surface contaminants of metals for recycling. And although NORM contamination does not generally pose a threat to the health and safety of personnel at metal mills and scrap yards, there is no consensus about the disposition of NORM-contaminated metal. The changing of trade barriers (such as the North American Free Trade Agreement) also factors into the problem, as materials cross international boundaries and enter the recycling stream. The efforts of entities such as Conference committees, federal regulatory agencies (e.g., NRC, EPA, DOT), state radiation control agencies and the affected industries will be presented and discussed.

INTRODUCTION

A scrap dealer in Illinois, after borrowing the survey meters and expertise of a nearby national laboratory which occasionally sold metal scrap to him, decided to purchase a radiation survey meter. He initially found a radiation level above what he was led to believe was a normal ambient background. He verified that the batteries were installed properly in the instrument, and proceeded to conduct a more complete survey of his property. He found radiation levels up to 1 millirem per hour. He called for help. Eventually, a source of cesium-137 containing 14 CBq (370 mCi) was found to have been buried on the property. No one knows where the source came from, and it was on his property for at least three years before it was discovered in December, 1994. Obviously, there was the potential for radiation exposure of scrap yard workers.

In January, 1994, a scrap dealer in Estonia detected radiation in a load of scrap delivered by a truck. A cesium-137 source was found and was transferred to a hazardous waste disposal facility, but it was then stolen in October, 1994, when three brothers removed the metal box containing the source. When the source fell out of the box, one of the men picked up the source and put it in his pocket. The source was found by Estonian authorities after the man and the family dog died, and a stepson was hospitalized because of radiation injuries. Radiation levels in the man's kitchen were above one gray (100 rad) per hour. Later, a second source was found along a roadside by authorities who happened to have their detector turned on as they drove by.

These recent episodes illustrate dramatically the continuing problem of radioactive materials in metal scrap. It is also a problem whose true scope is yet to be measured. In the April, 1995, issue of *Health Physics*, we reported on 315 events involving radioactive materials in metal scrap for the period ending in 1993. We also reported that we were working with the Steel Manufacturers Association (SMA) to add similar data collected by its members. This effort has been successful and we can now report on the combined data updated through December, 1994. Canadian data is also included.

RECENT EXPERIENCE

Through the end of December, 1994, the authors are aware of 1101 cases where radioactivity has been discovered in shipments of metal scrap. Of this total, there have been 24 smeltings in the United States, principally affecting the steel industry [figure 1]. Both sets of data have a significant, common feature: there has been a clear, upward trend in the discoveries of radioactive material in metal scrap.

As can be seen from the accompanying slide [figure 2], what are generally referred to as portal monitors have by far detected the greatest number of abnormal radiation finds, although the use of hand-held, portable survey meters also has found about 2% of the total, and even visual sightings have resulted in sources being found. One operator became suspicious when he saw an object stencilled "property of the U. S. government" on it and tritium gauges were found. As you can see, only a few of the discoveries have been by sight, although an interesting, more recent sighting will be discussed later. The portal monitors come in many configurations, some of which have detectors to cover four sides of a vehicle (two sides, top and even bottom); some of which are coupled to computer microprocessors, which update background radiation levels continuously and can be set to alarm if variations as little as a few percent above background are detected. Others may be either "simple" sodium iodide or plastic scintillators coupled to alarming rate meters.

This chart [figure 3] shows the distribution of radioactive materials found. About 10% of the finds have involved Atomic Energy Act materials, but the vast majority, about 70%, have involved either radium (as discrete finds or sources) or naturally-occurring radioactive materials -- NORM. However, sources or devices have accounted for about one eighth of the finds. As can be seen, when sources or devices are found, Atomic Energy Act materials comprise the majority of finds.

As noted earlier, the number of discoveries has generally been increasing every year [figure 4]. One possible explanation for this is that there are more sources finding their way to the scrap yard, but the data we have on discovered sources shows a less gradual increase [figure 5]. Another explanation is the installation and use of portal monitors by the metal manufacturing mills and their (feeder) scrap yards. This reminds one author of something he learned in an epidemiology class a few decades ago: the monitors may be seeing prevalence, not incidence. It may be that the problem -- especially of NORM-contaminated items -- has always been there, but we're only recently seeing how widespread this is, especially as the instrumentation grows more sophisticated.

World-wide, we are aware of 36 smeltings of radioactive materials in metals [figure 6]. Most of these (25) have occurred in steel mills and foundries, although there have also been five smeltings in aluminum mills, two each in copper and gold, and one each in lead and zinc mills. Although most of the events have occurred in the United States, this aspect may be due to better domestic information flow. This may also be due to our reliance upon technology, too, in that these portal monitors do an excellent (although by no means fool-proof) job at detection. An example of this was the detection early this year of contaminated steel plate, which originated in Bulgaria, and which was detected first in Mississippi. Other foreign smeltings which resulted in the exportation of contaminated product(s) were detected similarly. Counting this Bulgaria smelting, there have been seven cases where contaminated product was exported to the United States. The radioactive materials involved in these smeltings are also shown.

The Steel Manufacturers Association is an organization representing 55 steel manufacturing companies with 107 mills in North America. Their members report their discoveries of abnormal radiation found in (incoming) shipments of metal scrap feed. This is done on a quarterly basis, and the information is added to the incident database. Information they provided proved to be the largest volume of detections in 1994. The Radiation in Resource Recovery (E-23) Committee has been active in this field since it was chartered in 1991. Besides the SMA, we are seeking to have other industry trade organizations adapt a similar reporting scheme. Although the steel industry has been affected the most by smeltings, no metal manufacturing process is immune from the problem. What our efforts have shown, however, is how diverse the trade organizations are. As an example, the most recent smelting

in the U.S. occurred in a foundry which is not a member of organizations such as the SMA, the Institute of Scrap Recycling Industries, Inc. (ISRI), the American Iron and Steel Institute (AISI), the Specialty Steel Institute of the United States (SSIUS) or other organizations with whom E-23 committee members have (previously) interacted.

A good portion of the incidents are reported by the use of Department of Transportation exemption E-10656, for shipments of metal and scrap discovered in transit as containing or being contaminated with radioactive materials. To those of you who are involved with the investigations on these discoveries, we thank you for your efforts and assistance.

The enactment of the North American Free Trade Agreement (NAFTA) is expected to result in an increase in the trade on scrap metal between the Agreement member companies. In this context, it is interesting to note that none of the three member companies in Mexico reported any discoveries, but this may also be due to those companies not having portal monitors. The database includes 21 discoveries by industries in Canada, although the data is not as specific as desired. This may be due to a lack of uniform reporting requirements, but then, no uniform reporting requirements exist in the U.S., either.

As many of you are aware, the Environmental Protection Agency (EPA) has been named as the lead federal agency on incidents where abnormal radioactivity has been discovered in the public sector. This would include not only scrap metal radioactivity but also unusual or abnormal radioactivity in such places as landfills. The lead federal agency is responsible for providing technical assistance, when called upon by the state, in handling these situations. An early test of this system is underway. Early in March, 1995, a scrap dealer in Pennsylvania notified the state radiation control agency that he had found a radioactive device [slide] in a load of scrap from South Carolina. The label on the device said that it contained 37 GBq (1 Ci) of hydrogen-3. Further, the state determined that the device was a generally-licensed device distributed by a company in California. California representatives informed Pennsylvania that the company had gone out of business in 1988, and that if any records of the distribution of generally-licensed devices existed, they were archived, and thus not readily accessible -- to man or machine. South Carolina was also contacted, but their records showed no match for between the device and any of its licensees. NRC has been requested to provide assistance, to help identify the original licensee and to arrange for disposal of this orphan source. EPA, as lead federal agency, has been similarly requested to provide assistance. We would like to report there is a happy ending to this, but one does not (yet) exist.

Last September, the E-23 Committee, along with representatives of both other industry trade organizations and several government agencies, met with the EPA about the problem of accidental

smeltings of radioactive materials in the steel industry. The EPA is looking at the problem of smeltings and may sponsor workshops or other venues in their efforts at preventing pollution. One major problem from the smelting of radiation sources, principally cesium-137, is the creation of a mixed waste, since the furnace dust now contains a radioactive constituent beyond the usual heavy metals. Details of this meeting were given in the November, 1994, *Newsbrief* of the CRCPD.

IMPACTS OF SMELTINGS AND DISCOVERIES

In 1993, a scrap dealer in Florida who performs radiation monitoring discovered what turned out to be a radiation therapy treatment head. Fortunately, no source was inside, but the device could have contained a few terabecquerel (kilocuries) of cobalt-60. Had this therapy head gone through a shredder at a scrap yard, the head would have been flailed into small pieces, and, had a source been inside, the source could have been breached, resulting in the contamination of the facility beyond potentially lethal radiation levels from the bare source. In 1994, a gauge containing 12 GBq (330 mCi) of cesium-137 was inadvertently disposed of with metal scrap and transferred to a scrap processor. The metal scrap was then shredded and this processing separated the source from the source housing. Fortunately, in this case, the shredding process did not result in breaching the source. [As a note on the shredded source: markings present on the source capsule enabled the regulatory agency to determine the owner of the device. The owner was fined \$250 for improper disposal.] Recall that in the Mexican incident the source capsule was breached, and its thousands of millimeter-sized pellets contaminated the junk yard and were also tracked around the neighborhood. We have already noted the unfortunate incident in Estonia which resulted in injury and death.

Recall further the smelting of cobalt-60 in Juarez, Mexico of early 1984. A tractor trailer carrying reinforcing rods was discovered as being "radioactive" when it exited a gate at the Los Alamos National Laboratory. Some of the reinforcing rods were used in residential construction and had to be removed. What has come to light recently were stories from the Far East of the construction of high-rise apartments built with contaminated reinforcing rods. This may have been a serendipitous discovery: in Taiwan, a dentist was establishing an office in a building, and had called upon the local authorities to check the shielding. The authorities discovered high levels of radiation without the X-ray machine being activated! The authorities traced this to the reinforcing rods in the walls of the buildings. A Japanese television program showed there may have been several dozen -- I apologize for not using an SI unit for this! -- of these buildings constructed. Estimates were that individuals living in these could have received doses on the order of one sievert (100 rem) or more before the exposures were discovered. It is possible that the extent of the usage of this contaminated product and consequently the radiation doses received by individuals residing in or using

those buildings may never be known with any high degree of certainty.

Can this happen here? Yes. These incidents show that radioactive contamination and radiation injuries can result when radioactive materials become mixed with metal scrap. Thus, there is a strong health and safety incentive for developing ways to prevent radioactive sources from entering the metal scrap recycling stream, to detect those sources which do enter it, and to enable the safe storage and eventual disposal of sources which are found.

CONCLUSIONS AND RECOMMENDATIONS

The problem of radioactive scrap is going to be with us for the foreseeable future. To counter the threat, there have been a number of initiatives taken, both to enhance awareness and to solve the problem of what to do now that the alarm has gone off.

Many of you have seen the warning poster "Radioactive Scrap - Beware" issued by the NRC (NUREG/BR-0108). With the EPA now being the lead federal agency, self-adhesive labels to update the notice, background, and "where to get help" sections of the posters are available from the NRC. Note that, although the telephone numbers for (all) the state radiation control agencies were not printed on the poster, it recommends calling the State radiation control agency first. We strongly suggest that when you visit the metal mills and scrap yards, either to gather information or as part of an investigation, leave your calling cards, look at their procedures and recommend that they contact you (or your agency) first.

Beyond this, the Institute of Scrap Recycling Industries, Inc. (ISRI) has published its second issue of their Recommended Practice and Procedure "Radioactivity in the Scrap Recycling Process," which was also distributed to the Conference. ISRI also has produced a videotape about this, and copies have been sent to ISRI members and to the state radiation control agencies. A Spanish version of this videotape is also being prepared. The Radioactivity in Resource Recovery (E-23) Committee participated in providing the technical expertise for the ISRI document and videotape.

As stated before, the use of radiation monitors has been the most effective measure in preventing accidental smeltings. These, however, cannot guarantee 100.00% protection. This can be demonstrated by a few mills having had the unpleasant experiences of a second smelting. But monitoring of incoming scrap at a steel mill may be represent the least effective time and place for such monitoring. We recommend that you monitor early and often -- when it is received by the "mom and pop" scrap dealers, when it leaves them to go to a larger facility, at this facility, before it is bundled or shredded -- in fact, all along the line. We also recommend that users and suppliers of scrap consider specifying in their contracts that the scrap has been monitored and found to be

free of abnormal radioactivity. This, however, leads to another issue, that of the lack of consensus standards for monitoring equipment or for acceptable levels of NORM or even other radioactive materials. We need to work further with the affected industries and other federal agencies, most notably the EPA, in the development of such standards or levels. As an example of the concern about equipment, during April, 1995, the SMA sponsored a gauntlet test, in which manufacturers of scrap monitoring systems were invited to install their radiation monitors for tests of system sensitivity in detecting a 7.4 GBq (0.2 Ci) cesium-137 source buried in different configurations of shredded scrap. Eight companies participated in this test, which was held April 17-28.

The DOT exemption for scrap metal radioactivity, which this Conference sought, has been in existence since 1991, and has undergone one extension (another will be sought, since these exemptions are for a duration of two years only). This has been an effective means of providing assistance to the facilities where the scrap metal has been discovered as radioactive. The assistance this brings can prevent workers from unnecessary radiation exposure or even from breaching a discovered source of radiation at a facility.

The NRC has been requested to improve its control over, and licensees' accountability for, licensed devices. This would apply not only to specific licensees but would also involve looking at the concepts behind the general license philosophy as well. The presence of gages in scrap metal loads indicates a breakdown in the control of the devices. If a device makes its way into a steel mill and is smelted, it may cost that mill upwards of \$10,000,000 (US\$) to decontaminate the mill and dispose of the radioactive contamination, assuming that there is a place to send the contaminated material and that no mixed waste was generated.

Although NRC is looking at regulatory alternatives to improve the control and accountability of sources and devices by its licensees, the truth is that the number of radioactive sources that inadvertently enter the scrap stream can never be reduced to zero (assuming that radiation sources will continue to be used and produced). The industries involved with metal recycling must maintain and enhance their vigilance. An estimate is that only two thirds of the steel mills and only half of the scrap metal dealers have any sort of radiation detection instruments, either fixed or portable, to screen incoming metal for radioactivity.

To help address this part of the problem, the E-23 Committee developed a position paper advocating universal radiation monitoring by all persons who handle metal scrap. The CRCPD Executive Board endorsed this position paper, and it is being circulated for comment, and, we hope, endorsement by industry trade organizations. Eventually, we seek to gain appropriate governmental agency endorsement as well. Initial reactions have been favorable; such a statement can provide needed leverage for mills to include provisions in scrap metal purchase contracts to

require suppliers to perform radiation monitoring of the metal scrap at the processing or recycling facilities where the volumes and densities of the scrap metal are smaller, and thus less likely to shield a radiation source from detection.

If this tactic succeeds, regardless of government action to improve control over licensed sources, we should see a trend of fewer sources uncovered by the metal manufacturing facilities, which should ultimately result in fewer smeltings. We will still need to address the problem of providing for safe storage and ultimate disposal of found sources, but we will be moving in the right direction.

Smelting of radioactive materials summary of U.S. occurrences

(FIGURE 1)

- Since 1983, twenty-four (24) domestic instances of the accidental smelting of radioactive sources
- 15 Cs-137; 1 Co-60; 3 Ra-226; 1 Th; 3 Acc.; 1 Am
- 16 occurred in steel mills, 3 Al; 2 Au; 1 @ Cu,Pb,Zn
- Events discovered by monitoring of slag/dross (4) or of flue dust shipments (15). One discovery occurred at a highway weigh station.
- There were 12 foreign events, in addition to these: Mexico, Taiwan, Brazil, Italy (3), Ireland, India, Russia, Estonia, Kazakhstan, Bulgaria. Others??

(revised January, 1995)

Discoveries of radioactivity in metal scrap (selected cases)

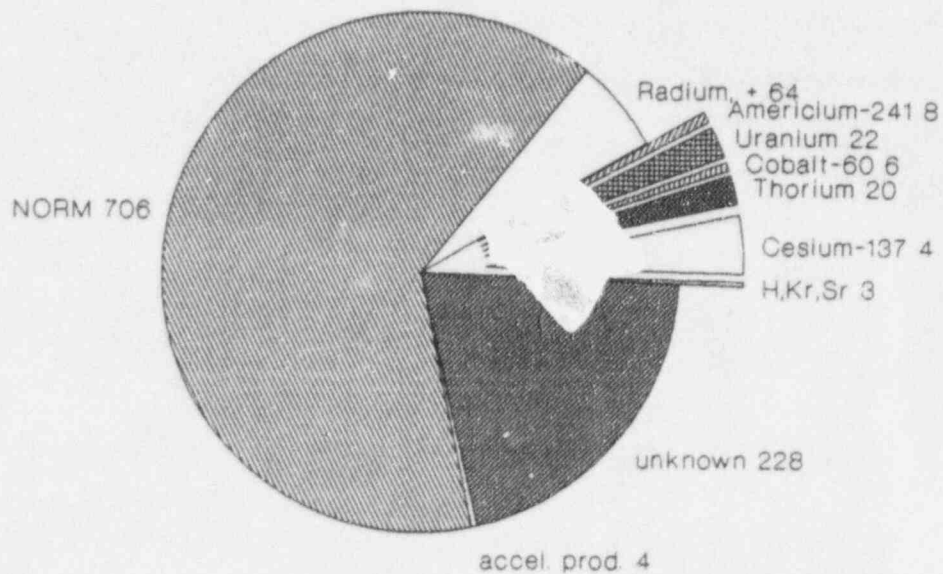
(FIGURE 2)

- reported/documented U.S., Canadian cases only
- Time span: January, 1985 - December, 1994
- 1101 Cases (1077 discoveries, 24 smeltings)
- How discovered:
 - stationary monitors -1053 (95.7%)
 - hand surveys - 19 (1.7%)
 - warning labels - 9 (~1%)
 - unknown - 19 (~1.7%)
 - other - 1 (~0.1%)

(revised April 21, 1995) JGY

Discoveries of radioactivity in metal scrap (continued) thru 12.31.94

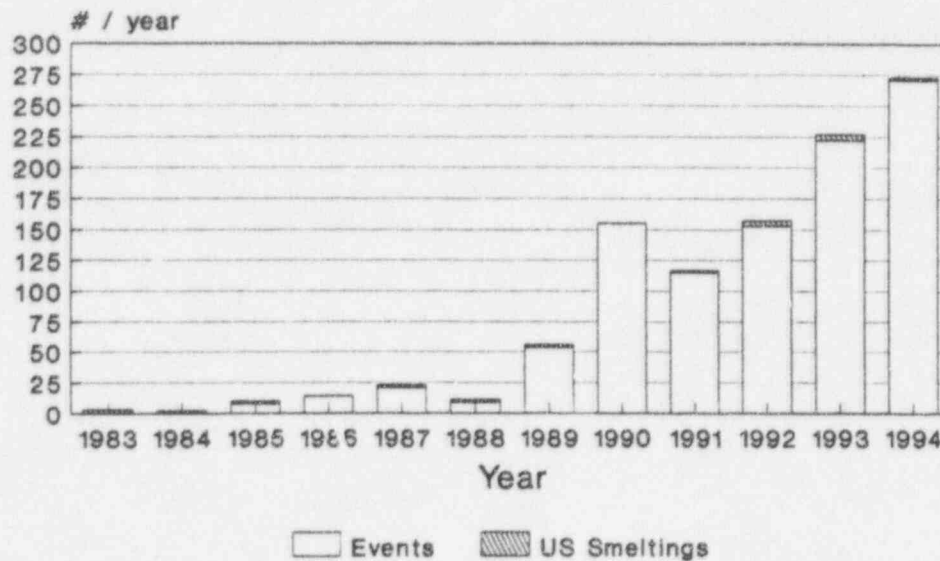
(FIGURE 3)



(N=1101; revised April 21, 1995 JGY)

Events, Smeltings of RAM "selected" (=reported) cases

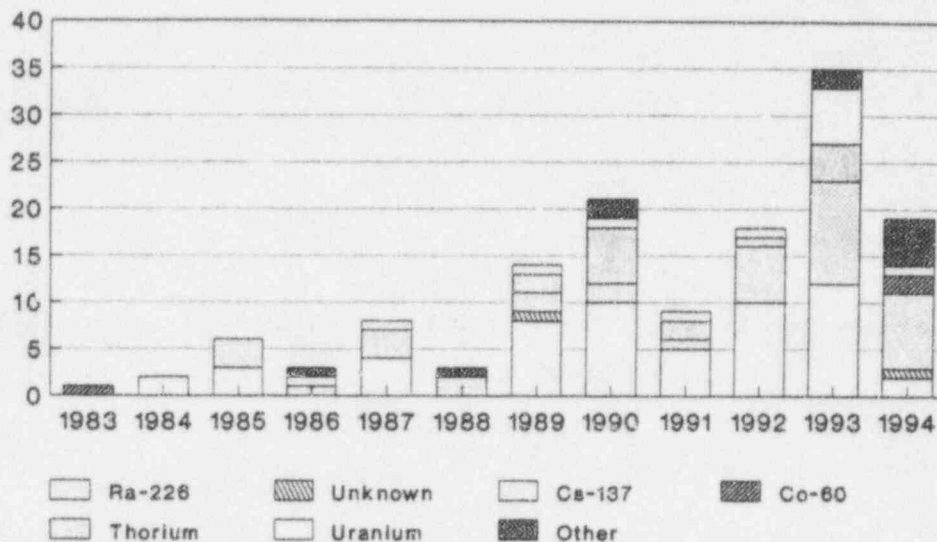
(FIGURE 4)



as of December 31, 1994

Sources, devices found in metal scrap, 1983 - 1994

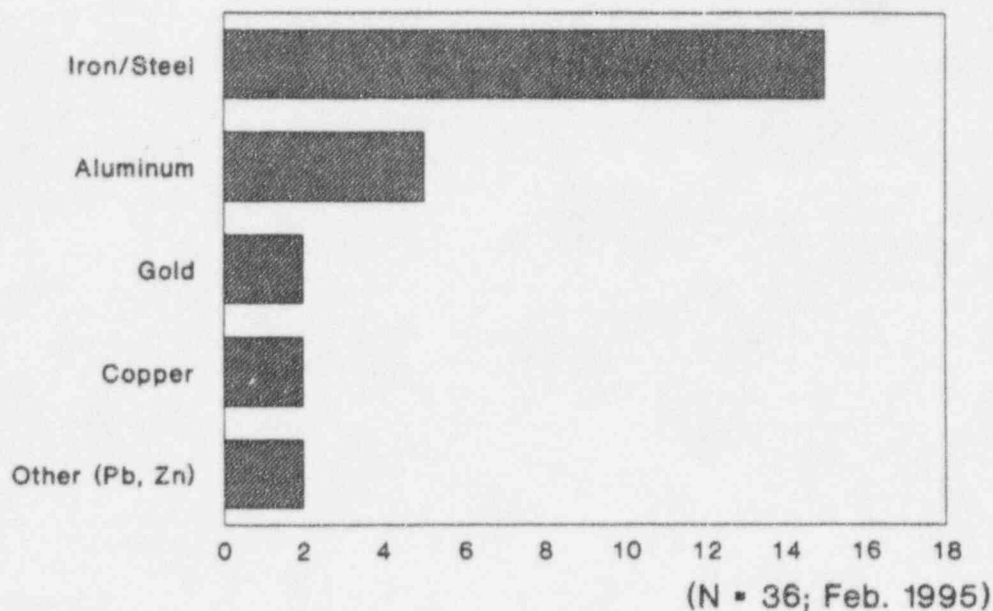
(FIGURE 5)



N = 141; 4.24.95 JGY

World-wide Smeltings of Radioactive Sources

(FIGURE 6)



IMPROPER TRANSFER/DISPOSAL SCENARIOS
FOR GENERALLY LICENSED DEVICES

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Abstract

A recent study by the Nuclear Regulatory Commission revealed areas of safety concern related to the improper transfer or disposal of certain generally licensed devices containing radioactive material. This report provides an evaluation of the potential for inadvertent exposure of the public to the radioactive material in these devices and gives some estimates of the radiation doses which could result from such exposures. Devices listed in an NRC registry were organized into sixteen categories, and scenarios were developed which predicted the probabilities of a device in a category passing through a specific process, being contacted by a particular type of individual, and ending up in a given situation. Radiation dose estimates for external and internal irradiation were derived for two broad types of situations: one in which the activity of a device is essentially intact and one in which the activity has been dispersed over a wide area.

Executive Summary

Many products containing radioactive material may be used by members of the general public and industry without extensive radiation safety programs because the nature of the products allows them to be regulated under general licenses. A recent study by the Nuclear Regulatory Commission revealed several areas of safety concern related to the improper transfer or disposal of these products by persons unfamiliar with license requirements. This report analyzes potential consequences of such improper handling of sixteen classes of such devices containing sealed sources. Probabilities of certain events resulting from the improper transfer or disposal of these devices were derived. Radiation dose estimates were calculated for exposure of individuals to radioactivity from these devices. Although the results generated through this analysis were based on a small amount of data, some conclusions about public health concerns are evident.

The two major dosimetry concerns are external irradiation and internal irradiation. The former concern relates to exposure to the radiation field from the radioactive source in a device or contact to the source. Internal irradiation may occur from improper handling of the device, in which activity may be released from the source, or intake of radioactivity which has been dispersed over a wide area because of the device inadvertently being incinerated, passed through metals recycling, or buried in a landfill, with leaching of the radioactive material into public drinking water supplies.

Exposure to the radiation field was not a significant concern for most of the devices, even under a worst case assumption of exposure for 20 weeks at 100 cm. Devices for which exposures under this assumption were greater than 500 mrem (0.005 Sv) include all types of gamma gauges (class B), X-ray fluorescence analyzers (class E-1 and E-2), and self-luminous devices containing Kr-85 (class G-1). In almost all cases, contact to the radioactive source for three hours will produce localized radiation doses which are inappropriate for members of the general public. For many devices, the source is extremely inaccessible, and these dose equivalents would not be expected (e.g., gas chromatographs, analytical instruments with calibration or reference sources). In other device types, however, the source can be accessed through a built-in mechanism, and significant radiation doses can occur if the device is manipulated by persons unaware of the radiation hazards.

Internal radiation doses were derived for several situations. A worst case situation was defined as intake of 30% of the radioactive material by inhalation or ingestion. More realistic numbers were generated by assuming a much smaller fraction, usually between 10^{-6} and 10^{-4} . Estimates were also generated for inhalation of radioactive material if a device were incinerated, and for ingestion of radioactive material which may have leached from a landfill site. For these situations, the results varied greatly, depending on the radiotoxicity and activity level of the radionuclide concerned. Significant estimates were derived for inhalation of Am-241, Ra-226, Cm-244, Pm-147, Pu-239,

and, to a lesser extent, Po-210, Pb-210, Fe-55, Ni-63, Sr-90, Co-60, Cs-137, and Cd-109. High dose equivalents were estimated for ingestion of Am-241, Cm-244, Pm-147, and, to a lesser extent, Po-210, Pb-210, Fe-55, Sr-90, Co-60, Cs-137, and Cd-109. Dose equivalents from intake of material from the plume of an incinerator were highest for Am-241, Ra-226, and Cm-244. Ingestion of radioactive material from landfills were significant for Ra-226 and Sr-90. The model used for transport from the landfill predicted very long holdup times in the landfill soil matrix for some nuclides, which resulted in the prediction that no Am-241, Cm-244, or Pm-147 would be ingested. These nuclides were all significant hazards if ingested, and would be of concern if the migration rates were significantly faster than predicted by this model.

Although more literature was available for consequences of metal recycling of radionuclides than for any other category, only quantitative information was listed for Co-60. Therefore, this analysis was performed only for Co-60 in gamma gauges. Inferences based on available literature showed that dose equivalents received by members of the general public who purchase contaminated products would most likely not exceed 500 mrem/yr (0.005 Sv/yr) in most cases.

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1) Introduction

The Nuclear Regulatory Commission (NRC) and the Agreement States regulate the distribution and use of all products within the United States that contain byproduct material. The NRC classifies the regulatory control of byproduct material into one of three categories: specific license, general license, or exempt from regulations. The classification depends on the type, quantity, and use of the material.

Many products containing radioactive material can be used by the general public and industry without extensive radiation safety programs. These products contain relatively small amounts of radioactive materials that are sealed within the device (sealed source) so that they can be used by persons without formal training in radiation safety. The use and distribution of such products, as well as small amounts of source material, are controlled through general licenses. General licenses are in effect for persons using certain radioactive materials without the filing of an application with the NRC. Estimates indicate that over 240,000 such devices are in use in the United States under six categories of such licenses.

A study conducted by the NRC in 1984, 1985, and 1986 revealed several areas of safety concern about the use of some sealed source devices under general license. Investigators observed that accountability for some devices was inadequate and that users were frequently unaware of regulations. Furthermore, some devices could not be located and final disposition of some devices could not be determined by the user or the NRC.

This report summarizes work that was done to develop potential scenarios for improper transfer or disposal of these devices and provides an assessment of both realistic and maximum dose equivalent to persons potentially involved in each scenario. Additionally, dose equivalents for the general public are given for those scenarios that indicate a pathway to the general public.

2) Methods

2.1) Devices considered

The NRC maintains a nationwide registry of sealed sources and device designs that are deemed acceptable for licensing purposes. A data base was developed from this registry for over 600 examples of licensed devices containing radioactive materials. The devices were arranged into categories based on such characteristics as radionuclide, activity, relative radiotoxicity, principal use, manufacture, date of registry, etc. This resulted in the selection of sixteen classes of devices for scenario development; these classes are listed in table 1.

An analysis of hazard to the public was performed by the NRC for those gamma and beta gauges having activities greater than 20 millicuries (740 MBq). Therefore, for this assessment, the activity for the gamma and beta gauges (classes B, C-1, and C-2) was limited to 20 mCi (740 MBq).

May 12, 1987

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April 10, 1987

A brief description of each type of device is given in the Appendix. Radiotoxicity of the nuclides in these devices is based on the classification of radionuclides according to relative radiotoxicity per unit activity given in Safe Handling of Radionuclides (IAEA 1973). A radionuclide may be classified in one of four groups: a) very high radiotoxicity, b) high radiotoxicity, c) moderate radiotoxicity, and d) low radiotoxicity.

2.2) Scenario Development

The task of developing scenarios for improper transfer or disposal for each class of device began with listing a wide variety of possibilities. The number of possible events was reduced to a manageable number through literature searches, interviews with people familiar with the devices and disposal technologies, and committee reviews. The result is a set of six different initial events (scenario beginnings) that would result in improper transfer or disposal, with "event trees" leading to eight potential final conditions (scenario endings). A "generic scenario" is illustrated by Figure 1. In this figure, rounded rectangles are assigned to initial events, squared rectangles represent intermediate processes or individuals, and ellipses represent final conditions. The different initial events are listed in table 2 and the potential final conditions are listed in table 3.

Probabilities were assigned to the initial event pathways and intermediate process pathways on the basis of available literature, inferences about device characteristics, and personal interviews with knowledgeable sources such as manufacturers and users. An extensive literature search was performed; however, few data were found which could be used to assign probabilities. The estimated probabilities are educated guesses and may be subject to many sources of variability. Data may vary from region to region depending on local practices and ordinances.

Assigned probabilities for all pathways leading from an initial event or intermediate process add up to 1.0. Pathway probabilities leading into an intermediate process or final conditions add to give the total probability of a device reaching that point. For intermediate events, these probabilities are then multiplied by the appropriate pathway probabilities to give the contribution to the next intermediate process or final condition. The resulting probabilities for all final conditions also add up to 1.0. More than one significant number was used to calculate the results, but final results are only estimates and should be considered accurate to only one significant figure.

Probabilities assigned to the various pathways are based on the assumption that a device has been mishandled. That is, the total probability of 1.0 for all initial events applies to the devices mishandled, not the total number of devices in use. Table 4 shows, for each device type, the known number of cases of mishandled devices. This report presents possible scenarios, probabilities assigned to

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those scenarios, and dose estimates to the public, assuming that a device has been mishandled in some way.

2.3) Dose Assessment

Radiation dose estimates were derived for several different situations. The largest source activity recorded in the NRC registry for a particular class of device was used to estimate the radiation doses for that device. The radiation hazard to members of the general public was estimated for (1) an 'intact' device, in which the source itself is not damaged or only slightly damaged and the activity is not dispersed, and (2) a device in which the source integrity has been breached and the activity has been dispersed over a wide area. According to the scenario format used, members of the general public might contact radioactive material at several points. Because of the two general types of radiation hazard envisioned ('intact' and dispersed sources), these various members of the public were associated with one or the other type of hazard. Dose equivalents were estimated from both external and internal exposure pathways for the maximally exposed and average individual and for the exposed population. Unless otherwise noted, the activity intake value for the average individual was estimated to be half of that for the maximum individual (NUREG 1980).

The probabilities resulting from the scenario analysis should not be used to modify the associated dose estimates; the scenarios show the estimated probabilities that a device will follow a particular pathway if it is mishandled. If a member of the public comes into contact with a device, the dose that the individual receives will not be affected by a theoretical probability.

2.3.1) 'Intact' source

Because the source is assumed to be mostly intact, only a few individuals might be expected to contact the source at any time. In the scenarios defined in this report, individuals who might contact an intact source include trash handlers, persons who find or receive the device, salvage dealers, and workers at sanitary landfills, incinerators, or metal recycling plants. The dosimetry concerns for such situations include external doses, either to the whole body from exposure to the radiation field or to the skin from contact, and internal doses, from inhalation or ingestion of some of the activity.

Maximum and realistic dose equivalents were estimated for external and internal exposures to activity in the source. Because only one or, at most, a few individuals would be expected to contact 'intact' sources, population dose equivalents were not calculated.

Whole body dose equivalents were estimated based on a point source geometry. Gamma ray dose constants were taken from a report by Unger and Trubey (Unger 1981), the Radiological Health Handbook (USDHEW 1970), or AECL 7617 (Cross 1982). Some estimate of the time and distance relationships involved in a worst case scenario may be derived from the incident in Mexico in 1962 (Andrews 1963) in which a young boy

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found a Co-60 source in a field, carried it around in his pocket for several days, and then left it in his home for 15-25 weeks. Five people, including an unborn child, died in that incident; however the source intensity was much higher in that incident than would be involved with the sources considered in this report. The average distances were not known with any reasonable accuracy, but 3.3 feet (100 cm) may be used as the average distance from a source in the center of a small room.

Surface dose rates to the skin were estimated from the values for sealed sources in NCRP Report 40 (NCRP 1972) or from a study by Kocher and Eckerman (Kocher 1986). The values in NCRP 40 are for encapsulated sources of Co-60, Cs-137, and Ra-226, and therefore only include photon emissions. These values are given with the qualification that the listed values must be increased by 25-45 percent to account for electron production in the stainless steel walls assumed to encapsulate the source. For the calculations in this report, the values were all increased by 45 percent. Values in the report by Kocher and Eckerman include only beta emissions. Dose rates at a depth of 7 mg/cm² in the report by Kocher and Eckerman were used to estimate the maximum dose from an unencapsulated source to healthy skin. No reference was found that has information needed to estimate the photon dose rates from contact with other sources. For Co-60, Cs-137, and Ra-226, dose rates for the encapsulated and unencapsulated source are given, based on the information in these two reports.

If an individual carries a source for several hours in contact with the skin, some transitory or permanent effects may occur. In the Mexican incident, the source was carried for several days, but was not in direct contact with the skin. Radiation symptoms were evident, but were not identified as such, even after the child's death (Andrews 1963). Because of the size of most of these devices, they cannot be carried in a pocket, so a maximum contact time of three hours was hypothesized.

Internal dose estimates were based on the fifty-year dose equivalents per unit ingestion or inhalation of the individual radionuclides listed in ICRP Publication 30 (ICRP 1978). Because the chemical form of the material was not known, estimates were derived for all listed inhalation classes (D, W, or Y) and values of gastrointestinal absorption (f_1).

Intakes for the maximum individual were hypothesized based on device characteristics. To derive intake values for the more realistic case, available literature, which presented intake fractions for the general public in certain accident scenarios, was studied. In an analysis of possible accident scenarios involving spent fuel transportation (Wilmot 1981), the maximum fraction of Cs-137 released to the environment in the form of respirable material was postulated to be 10^{-3} , that for Co-60 was 10^{-1} , and that for actinides was 3×10^{-6} . These scenarios involved impacts, immersion in water, and punctures. A report (Ricks 1981) prepared for the Department of Transportation (DOT) cites a DOT rationale that in an accident of moderate severity, 1/1000 of the contents of a package containing radioactive materials might be

released. The authors further hypothesize that an individual in the vicinity of the accident might take in 1/1000 of the released material, implying that the fraction of original activity which might be taken up is 10^{-6} . The fraction released (10^{-3}) is of the same order of magnitude as the value specified by Wilmot for Cs-137, but is lower than his assumed values for Co-60 and higher than his assumed value for actinides. Although the type of enclosures and devices in these two studies were much different than those described in this document, these numbers give some guidance for postulation of the amounts of materials which could be released from these devices. If a radionuclide was not specifically treated by these studies, an attempt was made to use the maximum value listed for a similar nuclide.

2.3.2) Dispersed source

If the source in a device becomes extensively damaged through a destructive process, the activity may be dispersed over a large area with the possibility that a large number of people may come in contact with smaller amounts of activity than would be encountered with an 'intact' source. This might occur at an incinerator (with material subsequently released to the environment or buried in a landfill), in metals recycling (with material subsequently incorporated into consumer products or construction materials), or at a landfill (where material may leach from the landfill into nearby surface water or groundwater supplies and be ingested).

2.3.2.1) Incineration

Incineration is used by some cities and, in some urban areas, in apartment buildings and other smaller applications to reduce the volume of municipal wastes. Refuse may be pretreated (size sorted, shredded, or ground) and is oxidized at temperatures between 1000 and 1500 degrees Fahrenheit (540 and 820 degrees Centigrade). The residue, which consists primarily of metals, glass, and other unburned materials, may then be compacted and sent to a landfill or to salvage if the metals can be identified and isolated. Some residual materials may be used in building or road construction materials (NUREG 1980).

Effluents from incinerators are carefully controlled by combustion and particle removal technologies, with particulate removal efficiencies of over 0.9 in most cases. Buckley et al. (NUREG 1980) estimated the amounts of activity expected to be inhaled by the maximally exposed individual, the average individual, and the population near an incinerator which burns consumer products containing radioactivity as part of its normal refuse. Assumptions needed to derive the estimates include the number of devices incinerated per year, the fraction of activity which would be released by the incineration process, and the fraction of released activity which would escape with stack emissions. The other dose calculations in this report were done on the basis of activity from one device. Because this model requires input of a number of devices per year, a value of 5 devices per year was arbitrarily chosen for model input. With the exception of portable static eliminators (which were treated differently), the devices were assumed to be incinerated at one site, with a nearby

population of 73000. Other built in model assumptions allow the estimation of the activity inhaled by the average and maximum individual and the surrounding population. Once the activity inhaled is known, the standard models of ICRP 30 can be employed to estimate the committed dose equivalents.

Several authors have studied the release of radionuclides from medical or other institutional incinerators. Materials burned include medical and research wastes (planchets, filter paper, syringes, etc.), animal carcasses, and mixed general wastes. Table 5 summarizes these findings, which were used to estimate the fraction of material which may be released during incineration. The results were fairly consistent, with the majority of the Co-57, Co-60, Rb-86, Ru-103, and Cs-137 retained in the incinerator ash, while most of the H-3, C-14, I-125, and Tl-204 were released. A significant difference exists between the results of Classic et al. and those of Brekke et al. and Bush and Hundal for cobalt, with Classic et al. predicting a much higher fraction (0.273) than the other authors (0.006-0.038). The largest value for a particular element was used to estimate the fraction of burned material which would be released during incineration. A value of 0.5 was typically assigned to nuclides not specifically listed in the table, although higher values were assigned to special radionuclides (e.g. H-3, Kr-85). The assumption of Buckley et al. that 0.1 of the material released during incineration is released from the stack, as well as other assumptions about the number of incinerators, dispersion characteristics, etc., were used directly.

2.3.2.2) Metals Recycling

Lubenau and Nussbaumer (Lubenau 1986) have reviewed incidents involving licensed devices containing Co-60 and Cs-137 that have been inadvertently incorporated into consumer products and construction materials. They quoted surface exposure rates for some products surveyed. Exposure rates near contaminated products were in the range of 80 $\mu\text{R/hr}$ (21 nC/kg-hr). This was the reported exposure rate near steel products from the Brazilian contamination incident; the measured activity concentration in the products was 26 pCi/g (0.96 Bq/g) (NRC 1985). Products involved in the Taiwan contamination incident had an activity concentrations of up to 150 pCi/g (5.6 Bq/g) and exposure rates on contact of 100 $\mu\text{R/hr}$ (26 nC/kg-hr). The activity concentration expected in steel contaminated with activity from products such as those considered in this report may be implied from a report on the Auburn Steel company contamination incident (NUREG 1986) and a report by Lubenau (Lubenau 1985). The former report suggests a value of 50 tons for one steel melt; the report of Lubenau analyzes consequences of contamination of melts of up to 700 tons. Using the 50 ton value, the activity concentration from loss of a 10 mCi (370 MBq) Co-60 source would be 220 pCi/g (8.1 Bq/g). Based on the average value for exposure rate (1.9 $\mu\text{R/hr}$ per pCi/g), this would result in a product with an exposure rate of 410 $\mu\text{R/hr}$ (106 nC/kg-hr). Continuous exposure to this product would result in an annual dose equivalent of 3.6 rem (36 mSv); however, the dose rate will fall off quickly from the surface value. Using a reduction factor of 0.1 for distance, the annual dose equivalent to an individual would not exceed 500 mrem (5 mSv). If the

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activity is in a construction material, shielding of the radiation from the product and surrounding structures will further reduce the dose equivalent.

The steel billets from the Auburn plant weighed about 55 tons (5×10^4 kg) each (NUREG 1986). The Taiwanese steel fittings weighed 0.77 lb (350 g) each (NRC 1985). The maximum number of persons exposed would result from choosing the smaller of these two numbers. If the product activity concentration were 220 pCi/g (8.1 Bq/g) and the weight was 350 g, this would imply 77 nCi (2.8 kBq) per product, and the possibility that up to 130,000 products might become contaminated. If the product weight were 5×10^4 g, the number of products would be 910. Therefore, between 1,000 and 100,000 products may be contaminated in such an incident. These values imply annual population dose equivalents of between 2.7×10^2 and 2.7×10^4 person-rem for the product exposure rate assumed above, using the factor of 0.1 for distance and assuming one person per product and an average occupancy factor of 0.75.

This result only applies to Co-60 gauges of 10 mCi (370 MBq) under the stated assumptions. Because of the paucity of data and models on this subject, more refined estimates for Co-60 or estimates for other radionuclides and devices cannot be calculated.

2.3.2.3) Burial in Landfill

Buckley et al. (NUREG 1980) also discussed hazards to the public from disposal of radioactive consumer products in landfills. They concluded that of three major pathways (direct irradiation, inhalation of suspended particles, and ingestion of contaminated food or water) only ingestion of contamination would be of concern because of the low activities and energies of the nuclides involved. Although the activities and energies of the nuclides considered in this report may be higher, suspension of radioactive particles with subsequent inhalation is probably not a significant pathway to consider. Risks from direct irradiation are highest for exposure to the intact source, so ingestion of contaminated groundwater is the only pathway considered in this report.

Through a study of landfill characteristics, soil characteristics, rainfall rates, groundwater migration, and radionuclide decay, Buckley et al. constructed a model which allows estimation of total amounts of activity ingested per year from a given type of device. The model includes contributions from sources which have been directly deposited in the landfill as well as those which have passed through an incinerator before reaching the landfill. The latter type of device typically has a higher fraction of activity release.

Parameters for leaching from landfills were derived from the assumptions of Buckley et al. for their defined 'Reference Landfill Site' and for the various radionuclides. Retardation factors for their reference soil were given for all of the radionuclides considered in this section. Leach rates of 0.01 yr^{-1} and 1.0 yr^{-1} from intact products and previously incinerated products, respectively, were

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assigned. The time required for activity to migrate to a ground water withdrawal point was calculated for the radionuclides considered. This parameter allows calculation of the total activity ingested from landfill leaching. Standard models of ICRP 30 (ICRP 1978) were then applied to estimate the organ committed dose equivalents.

Buckley et al. did not explicitly define an average and maximum individual dose for this model or attempt to estimate the population that might be affected by activity leaching from a single landfill. If the same criterion as for the incineration model is used, the average individual might be expected to receive a dose equivalent that is one half as large as the maximum individual. They do state that there are 18,500 landfill sites in the United States, so a first approximation of population served by one landfill would be 12,000 ($220,000,000/18,500$). This is a crude estimate, because many of these landfill sites are small and because landfills are not evenly distributed with respect to population.

TABLE 1 CLASSES OF DEVICES FOR SCENARIO DEVELOPMENT*

APPLICABLE REGULATORY SECTION**	CLASS	DEVICE	RADIONUCLIDES AND MAXIMUM ACTIVITIES
31.3	A-1	Static Eliminators: Hand-Held/Portable/ Small Brushes	Po-210 - 0.50 mCi (18.5 MBq)
31.5	A-2	Static Eliminators or Detectors: In Equipment or Process Line (Very High Toxicity)	Po-210 - 100 mCi (3700 MBq) Am-241 - 0.0005 mCi (0.0185 MBq) Ra-226 - 0.0005 mCi (0.0185 MBq)
31.5	A-3	Static Eliminators or Detectors: In Equipment or Process Line (Low Toxicity)	H-3 - 250 mCi (9250 MBq) Kr-85 - 2 mCi (74 MBq)
31.5	B	Gamma Gauges	Co-60 - 10 mCi (370 MBq) Cs-137 - 20 mCi (740 MBq) Am-241 - 20 mCi (740 MBq) Ra-226 - 10 mCi (370 MBq)
31.5	C-1	Beta Gauges: Backscatter Type	Sr-90 - 0.025 mCi (0.925 MBq) Tl-204 - 0.10 mCi (3.7 MBq) Ru-106 - 0.025 mCi (0.925 MBq) Pm-147 - 0.050 mCi (1.85 MBq) C-14 - 0.050 mCi (1.85 MBq) Pb-210 - 0.010 mCi (0.37 MBq)
31.5	C-2	Beta Gauges: Transmission Type	Sr-90 - 20 mCi (740 MBq)
31.5	D	Gas Chromatographs	Ni-63 - 20 mCi (740 MBq) H-3 - 1000 mCi (37 GBq)
31.5	E-1	X-Ray Fluorescence Analyzers (Very High Toxicity)	Am-241 - 30 mCi (1100 MBq) Cm-244 - 100 mCi (3700 MBq)

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TABLE 1 CLASSES OF DEVICES FOR SCENARIO DEVELOPMENT* - CONTINUED

APPLICABLE REGULATORY SECTION	CLASS	DEVICE	RADIONUCLIDES AND MAXIMUM ACTIVITIES
31.5	E-2	X-Ray Fluorescence Analyzers (Moderate Toxicity)	Cd-109 - 20 mCi (740 MBq) Fe-55 - 100 mCi (3700 MBq)
31.5	F	Calibration or Reference Sources	Cs-137 - 0.10 mCi (3.7 MBq) Co-60 - 0.01 mCi (0.37 MBq) Ra-226 - 0.004 mCi (0.15 MBq) Sr-90 - 0.001 mCi (0.037 MBq)
31.5	G-1	Self-Luminous Devices	H-3 - 5000 mCi (185 GBq) Kr-85 - 1700 mCi (62.9 GBq) C-14 - 0.10 mCi (3.7 MBq)
31.7	G-2	Self-Luminous Devices in Aircraft	H-3 - 5000 mCi (185 GBq) Pm-147 - 300 mCi (11 GBq)
31.8	H	Analytical Instruments Containing Small Calibration or Reference Sources	Cs-137 - 0.040 mCi (1.5 MBq) Ni-63 - 15 mCi (555 MBq)
31.8	I	Calibration or Reference Sources	Am-241 - 0.005 mCi (0.185 MBq)
40.22	J	Small Quantities of Source Material	U-238 and Th-232 - 15 pounds at any one time, no more than 150 pounds per calendar year
70.19	K	Calibration or Reference Sources	Pu-239 - 0.005 mCi (0.185 MBq)

* See Appendix for device descriptions

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TABLE 2 INITIAL EVENTS (SCENARIO BEGINNINGS)

- I. Owner takes device out of use. Device remains in place. Loss of control results.
- II. Owner takes device out of use. Device is transferred to a storage location. Loss of control results.
- III. Owner takes device out of service and improperly transfers device to an unauthorized user (or some other individual). Loss of control results.
- IV. Device is discarded in the trash.
- V. Device is discarded in the environment.
- VI. Device is sold to a salvage dealer.

TABLE 3. FINAL STATUS OF DEVICE (SCENARIO ENDINGS)

- I. Device remains in place but not in use. No control of device.
- II. Device remains in uncontrolled storage on-site.
- III. Possession by unauthorized individual. Potential for improper use. No control of device.
- IV. Device remains in uncontrolled storage at salvage yard.
- V. Device reprocessed; incorporated into consumer products.
- VI. Device reprocessed; incorporated into construction materials.
- VII. Device buried in sanitary landfill.
- VIII. Device remains uncovered in the environment.

TABLE 4. KNOWN NUMBER OF CASES OF MISHANDLED DEVICES

Class*	Estimated Number Distributed	Number of Known Incidents**	Percentage
A-1	20,000	1	5.0×10^{-3}
A-2,A-3	170,000	21	1.2×10^{-2}
B	4,200	11	2.6×10^{-1}
C-1,C-2	8,000	1	1.2×10^{-2}
D	8,000	1	1.2×10^{-2}
E-1,E-2	720	0	0
F	---	2	---
G-1	180,000	4	2.2×10^{-3}
G-2	90,000	1	1.1×10^{-3}
H	7,000	1	1.4×10^{-2}
I	2,000	0	0
J	---	---	---
K	---	---	---

* See table 1

- ** Sources:
- 1) Incident Summaries, Texas Dept. of Health, 1985-86.
 - 2) The U.S. Nuclear Regulatory Commission and the Agreement States, Licensing Statistics and other data, 1983.
 - 3) Agreement State Incidents Involving Generally Licensed Gauges, Licensing Statistics and other data, 1984, 1985.

TABLE 5. RADIONUCLIDE RELEASE DURING INCINERATION
IN VARIOUS STUDIES

Nuclide	Form	Fraction Released	Reference
H-3	Animal carcasses	1.0	Classic 1985
C-14	Animal carcasses	0.992	Classic 1985
C-14	Planchets	0.96,0.98	Bush and Hundal 1973
Co-57	Animal carcasses - microspheres	0.038	Brekke 1985
Co-57	Animal carcasses	0.273	Classic 1985
Co-60	Filter paper	0.009,0.006	Bush and Hundal 1973
Rb-86	Filter paper	0.449,0	Bush and Hundal 1973
Ru-103	Animal carcasses - microspheres	0.165	Brekke 1985
I-125	Animal carcasses	1.0	Classic 1985
I-125	Filter paper and animal carcasses	0.056-0.9998	Bush and Hundal 1973
Cs-137	Syringes	0.525,0.18	Bush and Hundal 1973
Tl-204	Planchet	0.948,0.993,0.953	Bush and Hundal 1973

3) Results

3.1) CLASS A-1 - PORTABLE STATIC ELIMINATORS

3.1.1) Device Description

This class of static eliminators includes small, hand-held antistatic brushes used for reducing static from records, photographic film, or artists' canvases. They consist of a small plastic handle joined between two aluminum braces with a soft bristle attachment on the front. The source is usually contained in microspheres on a thin metal strip located inside of the handle and behind the brush bristles. Po-210 (an alpha emitter) is the most commonly used radionuclide, with activities up to 0.50 mCi (18.5 MBq) when sold. Because of the short half life (138 days), the activity at the end of their useful life (about one year) is about 0.080 mCi (3.0 MBq). Instructions included with the devices indicate they should be returned to the distributor, but distributors indicate that this is rarely done (NUREG 1980).

Static eliminators are themselves consumer products and the public may be exposed internally or externally before the eliminator is discarded in a landfill or a salvage yard. External exposure is often assumed to be negligible with internal exposure via ingestion, inhalation, or absorption being a greater hazard (NUREG 1980). Because they are a portable consumer product, it is difficult to account for the approximately 7,000 distributed per year. However, distributors interviewed claim that users frequently discard the used devices in the trash.

3.1.2) Scenario Development

If most of these devices are thrown away, the probability that they will be buried in a sanitary landfill is the highest probability in the final conditions of the scenario (0.63), as seen in Figure 2. High probabilities are found for the device remaining in place without control (0.1) and the device remaining in uncontrolled storage on site (0.1). Distributors and users of the devices may continue to keep the devices after their usefulness has expired instead of discarding them. These portable static eliminators may also fall into the possession of unauthorized individuals (0.078) because they can be transferred easily from individual to individual. The device may also remain unrecovered in the environment (0.08). Other outcomes seem less likely. Portable static eliminators contain little salvagable material, and would not generally be sent through metals recycling. This accounts for the lower probabilities in storage at the salvage yard (0.001) since salvage dealers would not save them. It is also unlikely that the devices will be used in construction materials (0.007) or in consumer products (0.003).

CLASS A-1 PORTABLE STATIC ELIMINATORS

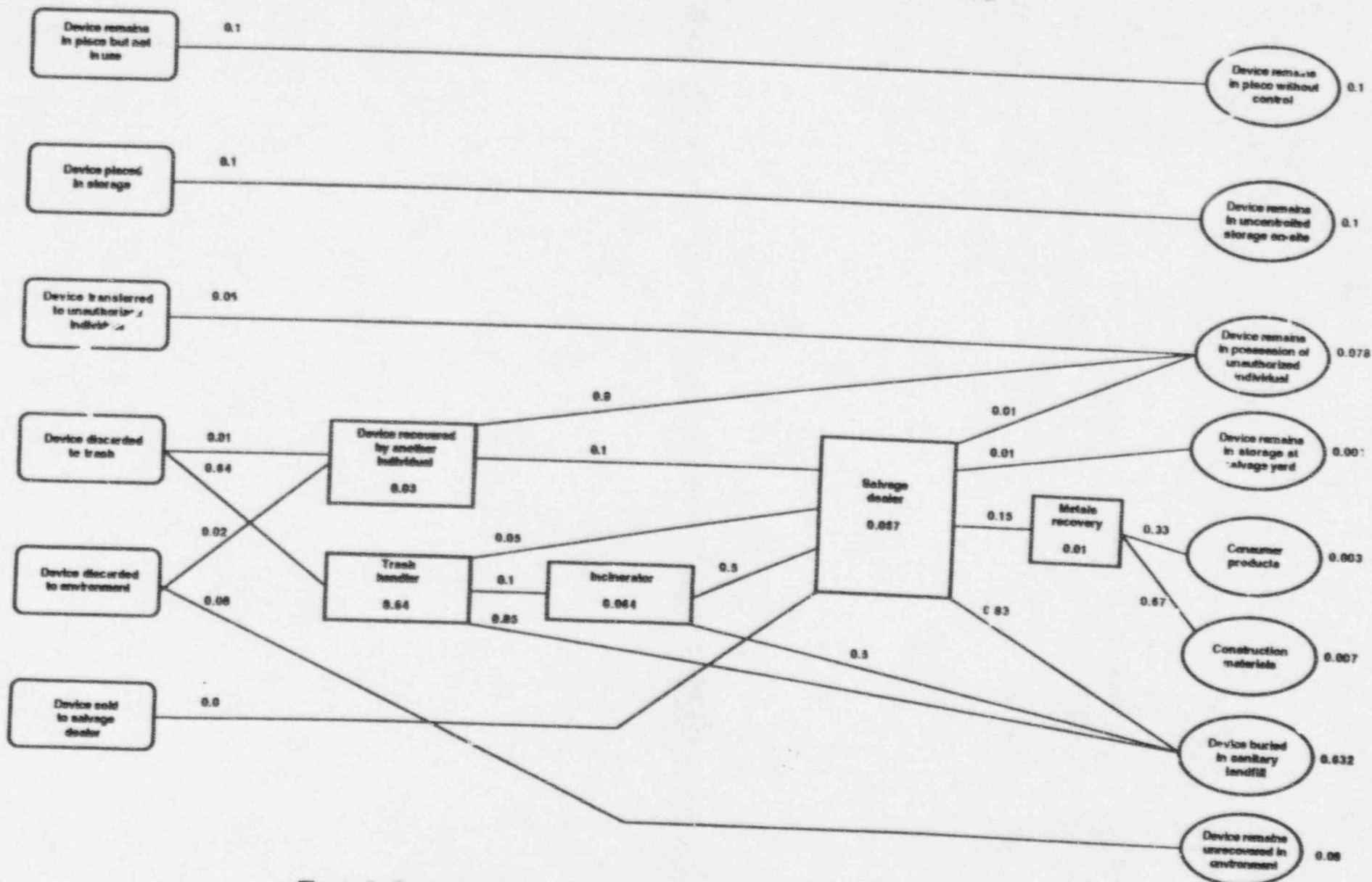


Figure 2 Scenario development for device class A-1, portable static eliminators

3.1.3) Dose Assessment

3.1.3.1) 'Intact' Source

External dose - maximum individual

Table 6 lists the estimated dose equivalents for exposure to point sources containing the maximum amounts of activity from the NRC Registry for each device class and radionuclide. The estimate for exposure to a 0.5 mCi (18.5 MBq) Po-210 source at 100 cm for 20 weeks is given for the gamma emissions. The value is of the same order of magnitude as natural background. Po-210 emits no beta particles with energy sufficient to penetrate the skin dead layer.

External dose - realistic case

Because the maximum external dose equivalent was on the order of 10^{-6} rem, the dose equivalent for a more realistic case will be lower than this value, and will not represent a significant hazard.

Internal dose - maximum individual

ICRP 30 assigns oxides, hydroxides, and nitrates of polonium to class W and all others to class D, with all compounds assumed to have an f_1 of 0.1. Table 7 lists dose equivalents predicted by the ICRP models for Po-210 if all of the device activity were taken in. The assumption that 100 percent of the activity might be taken in was used because the devices are routinely used by members of the general public, and children or others who tamper with the devices could cause the release of all of the activity. Even the small amounts of activity used in these sources can result in very high dose equivalents to the spleen and lungs if inhaled or ingested (table 7). However, these sources, much like smoke detectors, only constitute a significant risk if broken open in such a way that the radioactive material is available for intake.

In a study of the hazard from these devices, Webb et al. (Webb et al. 1975) found that under normal handling, very small amounts of the activity would be released from the microspheres, and that the dose equivalents to major organs would be much lower than the values in table 7. The microspheres were not susceptible to leaching and were assumed to pass directly through the gastrointestinal system. However, up to 50 percent of the activity could be removed from the microspheres if they had been subjected to impact or heating. Therefore, there is probably more risk to trash handlers and others who might contact the sources after they are thrown away. However, because of the short half-life of Po-210, less activity will remain associated with the devices at the time these people could contact them.

Internal dose - realistic case

Wilmot does not assign a release fraction for Po-210, so a value of 10^{-6} was assigned. Table 8 lists dose equivalents for ingestion or inhalation of 5×10^{-7} mCi (18.5 Bq).

3.1.3.2) Dispersed source

3.1.3.2.1) Incineration

Because these devices are routinely thrown into the trash, the number of devices incinerated per year will be higher than for other device types. From an estimate of the number of devices distributed per year (7000) and from the estimate in Figure 2 that 6.4 percent will end up at the incinerator, the estimated number of devices incinerated per year is 450. Using this value in the model of Buckley, the activity intake by the maximum individual is 5.2×10^{-10} mCi/year (0.019 Bq/year). Table 9 lists the dose equivalents for the maximum individual, average individual, and population residing near incinerators (22×10^6 persons) (NUREG 1980) based on this conclusion.

3.1.3.2.2) Metals Recycling

As stated in section 2.3.2.2, a general result for Co-60 gauges is the only result afforded by the available data. See that section for a discussion of this result.

3.1.3.2.3) Burial in Landfill

Based on the model of Buckley, the time for Po-210 to reach the withdrawal point is so long that all of the activity will decay before it reaches that point.

TABLE 6. ESTIMATED MAXIMUM DOSE EQUIVALENTS
FROM EXTERNAL EXPOSURE TO INTACT SOURCES

Device Class	Nuclide	Estimated Dose Equivalent (rem)		
		Whole Body*	Contact Encapsulated**	Contact Non-Encapsulated**
A-1	Po-210	8.8×10^{-6}	-	-
A-2	Po-210	1.8×10^{-3}	-	-
	Am-241	5.3×10^{-4}	-	-
	Ra-226	1.4×10^{-3}	1.7×10^{-1}	1.4×10^{-2} 3.9×10^1
A-3	Kr-85	1.05×10^{-2}	-	-
	H-3	-	-	-
B	Co-60	4.6×10^1	5.4×10^3	1.25×10^5
	Cs-137	2.6×10^1	2.7×10^3	4.05×10^5
	Ra-226	2.7×10^1	3.4×10^3	7.8×10^5
	Am-241	2.1×10^1	-	5.6×10^2
C-1	Sr-90	-	-	1.2×10^3
	Tl-204	3.7×10^{-4}	-	2.0×10^3
	Ru-106	-	-	7.0×10^2
	Pm-147	4.5×10^{-7}	-	3.4×10^2
	C-14	-	-	1.8×10^2
	Pb-210	-	-	2.4×10^2
C-2	Sr-90	-	-	9.4×10^5
D	Ni-63	-	-	-
	H-3	-	-	-
E-1	Am-241	3.2×10^1	-	8.4×10^2
	Cm-244	2.2×10^1	-	-
E-2	Cd-109	1.2×10^1	-	-
	Fe-55	2.6×10^1	-	-
F	Ra-226	1.1×10^{-2}	1.4×10^0	3.2×10^2
	Cs-137	1.3×10^{-1}	1.4×10^1	2.0×10^3
	Co-60	4.6×10^{-2}	5.4×10^0	1.25×10^2
	Sr-90	-	-	4.7×10^1
G-1	H-3	-	-	-
	Kr-85	9.0×10^0	-	-
	C-14	-	-	3.7×10^2

TABLE 6. ESTIMATED MAXIMUM DOSE EQUIVALENTS
FROM EXTERNAL EXPOSURE TO INTACT SOURCES - CONTINUED

Device Class	Nuclide	Estimated Dose Equivalent (rem)		
		Whole Body*	Contact Encapsulated**	Contact Non-Encapsulated**
G-2	H-3	-	-	-
	Pm-147	2.7×10^{-3}	-	2.05×10^6
H	Cs-137	5.1×10^{-2}	5.4×10^1	8.2×10^2
	Ni-63	-	-	-
I	Am-241	1.06×10^{-2}	-	1.4×10^{-1}
J	U-238	-	-	-
	Th-232	-	-	-
K	Pu-239	-	-	-

* 20 week exposure at 100 cm.
** 3 hour contact time.

TABLE 7. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION

OF 5×10^{-1} mCi (18.5 MBq) Po-210

Organ	Dose Equivalent (rem)		
	Ingestion	Inhalation	
		Class D	Class W
Kidneys	4.6×10^3	2.2×10^4	7.2×10^3
Liver	8.1×10^2	4.1×10^3	-
Spleen	8.1×10^3	4.1×10^4	1.2×10^4
Lungs	-	-	2.4×10^4

TABLE 8. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION

OF 5×10^{-7} mCi (18.5 Bq) Po-210

Organ	Dose Equivalent (rem)		
	Ingestion	Inhalation	
		Class D	Class W
Kidneys	4.6×10^{-3}	2.2×10^{-2}	7.2×10^{-3}
Liver	8.1×10^{-4}	4.1×10^{-3}	-
Spleen	8.1×10^{-3}	4.1×10^{-2}	1.2×10^{-2}
Lungs	-	-	2.4×10^{-2}

TABLE 9. COMMITTED DOSE EQUIVALENTS FOR INHALATION

OF 5.2×10^{-10} mCi (0.019 Bq) Po-210 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)		Average Individual (rem)		Population (Person-rem)	
	Class D	Class W	Class D	Class W	Class D	Class W
Kidneys	2.3×10^{-5}	7.5×10^{-6}	1.2×10^{-5}	3.8×10^{-6}	2.5×10^2	8.2×10^1
Liver	4.2×10^{-6}	-	2.1×10^{-6}	-	4.6×10^1	-
Spleen	4.2×10^{-5}	1.3×10^{-5}	2.1×10^{-5}	6.4×10^{-6}	4.6×10^2	1.4×10^2
Lungs	-	2.5×10^{-5}	-	1.2×10^{-5}	-	2.7×10^2

3.2) CLASS A-2 - STATIC ELIMINATORS OR DETECTORS: IN EQUIPMENT OR PROCESS LINE (VERY HIGH TOXICITY SOURCES)

3.2.1) Device Description

The ionization sources in these devices are used to remove static charge build-up in equipment, on production lines (conveyor belts, roller systems) or in air ducts. Detectors in this class are used to sense and measure static charge. They may be attached to the process line or be portable. Commonly used nuclides include Po-210 with activities up to 100 mCi (3700 MBq), Am-241 with activity of 0.0005 mCi (0.0185 MBq), and Ra-226 with activity of 0.0005 mCi (0.0185 MBq).

This class of static eliminator is covered under section 31.5 of the CFR as opposed to the portable static eliminator covered under section 31.3. Industrial static eliminator users are therefore required to perform leak tests, and observe transfer, disposal, and maintenance restrictions. Most of these devices differ from other products covered by section 31.5 because the major manufacturer leases the devices rather than selling them. The lease period is usually six months to a year. The major distributor of the devices revealed that approximately 95 percent of the devices are returned without a reminder to the client. If the product is not returned, a reminder is sent with the warning that if the device is not returned, another year's fee will be charged. They report good results using this method.

In a 1984 survey, the NRC also contacted licensees of equipment static eliminators to determine their awareness of license regulations. Eight of the twelve (67 percent) licensees surveyed were aware of the regulations and 80 percent were keeping proper receipt and transfer records (NRC 1987). Even with a seemingly greater awareness of the general license requirements for this class of devices, reports are filed concerning missing static eliminators or incidences of damage. Four of the licensees contacted in the NRC survey reported misplacing a total of ten static eliminators (NRC 87). Fixed static eliminators also have a problem similar to the gauges in that they are on a process line and may encounter a corrosive atmosphere (in which labels may disappear), or the devices may be lost because knowledgeable personnel have left the plant. Plant closings may also contribute to the devices being lost.

3.2.2) Scenario Development

Due to plant closings, the devices may remain in place without control (0.2) or be placed in storage (0.1) (Figure 3). Probabilities of 0.3 were assigned to the devices being discarded to the trash or the environment, with 0.15 each going to recovery of the device by another individual and to either the trash handler or remaining unrecovered in the environment. Probabilities are lower for the device being transferred to an unauthorized individual (0.05) or sold to a salvage dealer (0.05).

Final probabilities are highest for the device remaining in possession of an unauthorized individual (0.23). This is a result of

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CLASS A-2 EQUIPMENT STATIC ELIMINATORS

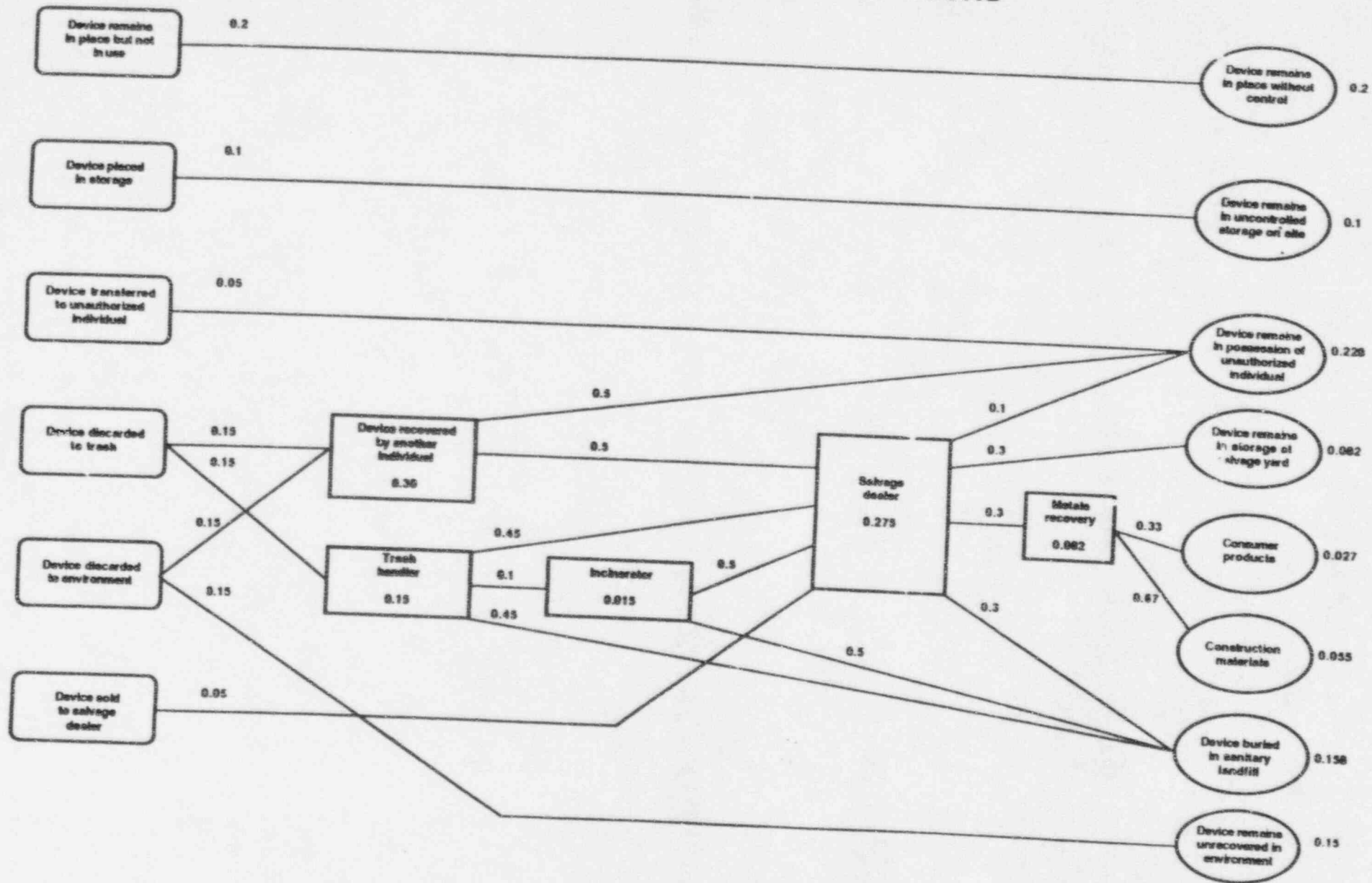


Figure 3 Scenario development for device class A-2, equipment static eliminators

the various pathways leading to this condition. The probabilities of remaining in place without control (0.20), being buried in a sanitary landfill (0.16), and remaining unrecovered in the environment (0.15) are also high.

3.2.3) Dose Assessment

3.2.3.1) 'Intact' Source

External dose - maximum individual

Table 6 shows dose equivalents for exposure to point sources at the activity levels stated in section 3.2.1 for Po-210, Am-241, and Ra-226 for 20 weeks at 100 cm. The values are all on the order of 1 millirem (0.01 mSv). As stated in section 3.1.3.1, Po-210 has no betas with energy sufficient to penetrate the dead layer of the skin. Dose equivalents to the skin at a depth of 0.007 cm are listed in table 6 for unencapsulated sources of Am-241 and for encapsulated and unencapsulated sources of Ra-226, assuming a three hour exposure in all cases. None of the values listed are sufficient to produce even transitory erythema.

External dose - realistic case

Because the maximum external dose equivalents for photons were around 1 millirem, the realistic estimates will be much lower and will not be significant. The maximum beta dose equivalents for Am-241 and Ra-226 were 14 mrem (0.014 mSv) and 39 rem (0.39 Sv), respectively. Realistic estimates for these nuclides will be lower; those for Am-241 will not be significant. Those for Ra-226 will be lower than the limits for radiation workers (18.75 rem per calendar quarter), or will be negligible if the source activity is not exposed.

Internal dose - maximum individual

Although it would take a significant effort, the source might be damaged in such a way as to make the material available for uptake into the body by inhalation or ingestion. The maximum dose equivalent possible would result from intake of all of the material by either pathway. Either scenario is quite unlikely; if the source were damaged so that the activity was scattered, it is not likely that even half of the radioactivity would be inadvertently swallowed or inhaled. Material suspended in air may be directly inhaled, resulting in activity deposition in the lungs and gastrointestinal tract. Scattered material could be spread over the skin surfaces and retained long enough to produce a dose to the skin which might result in some damage. The most likely pathway for ingestion would be from activity on the skin transferred to the mouth. Tables 10-12 list the committed dose equivalents which would result from intake of 0.3 of the source activity. The fraction of 0.3 was arbitrarily chosen as a maximum amount which might be inadvertently taken in.

Internal dose - realistic case

Based on the conclusions of Wilmot, uptake fractions of 10^{-6} are assigned to all three radionuclides under this section. Tables 13-16 list the associated dose equivalents for intake of 10^{-6} of the maximum source activities of Po-210, Am-241, or Ra-226, respectively.

3.2.3.2) Dispersed Source

3.2.3.2.1) Incineration

Based on the model of Buckley, the amount of activity inhaled by the maximum individual near an incinerator would be 3.47×10^{-7} mCi (1.3×10^{-5} MBq) of Po-210, and 1.7×10^{-12} mCi (6.4×10^{-11} MBq) of either Am-241 or Ra-226. Tables 16-18 show the associated dose equivalents for intake of these levels for the maximum individual, average individual, and population.

3.2.3.2.2) Metals Recycling

As stated in section 2.3.2.2, a general result for Co-60 gauges is the only result afforded by the available data. See that section for a discussion of this result.

3.2.3.2.3) Burial in Landfill

The model of Buckley predicts that the transit time from the landfill to a water withdrawal point for Po-210 and Am-241 is so long that all of the activity decays before it reaches the withdrawal point. For Ra-226, the model predicts an uptake of 1.7×10^{-6} mCi (6.2×10^{-5} MBq); dose equivalents corresponding to this uptake are shown in table 19.

TABLE 10. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION
OF 30 mCi (1.11×10^9 Bq) OF Po-210

Organ	Dose Equivalent (rem)		
	Ingestion	Inhalation	
		Class D	Class W
Kidneys	2.8×10^5	1.3×10^6	4.3×10^5
Liver	4.9×10^4	2.4×10^5	-
Spleen	4.9×10^5	2.4×10^6	7.4×10^5
Lungs	-	-	1.4×10^6

TABLE 11. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION
OF 1.5×10^{-4} mCi (5.6×10^3 Bq) OF Am-241

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Gonads	7.7×10^{-2}	1.8×10^1
Red Marrow	4.7×10^{-1}	1.1×10^2
Bone Surfaces	6.1×10^0	1.4×10^3
Liver	1.3×10^0	3.1×10^2

TABLE 12. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION
OF 1.5×10^{-4} mCi (5.6×10^3 Bq) OF Ra-226

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Gonads	5.1×10^{-2}	-
Red Marrow	3.3×10^{-1}	-
Bone Surfaces	3.8×10^0	4.2×10^0
Lungs	-	8.8×10^0

TABLE 13. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION
OF 10^{-4} mCi (3700 Bq) Po-210

Organ	Dose Equivalent (rem)		
	Ingestion	Inhalation Class D	Class W
Kidneys	0.92	4.4	1.4
Liver	0.16	0.81	-
Spleen	1.6	8.1	2.5
Lungs	-	-	4.8

TABLE 14. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION
OF 5×10^{-10} mCi (0.0185 Bq) Am-241

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Gonads	2.6×10^{-7}	5.9×10^{-5}
Red Marrow	1.6×10^{-6}	3.7×10^{-4}
Bone Surfaces	2.0×10^{-5}	4.6×10^{-3}
Liver	4.3×10^{-6}	1.0×10^{-3}

TABLE 15. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION
OF 5×10^{-10} mCi (0.0185 Bq) Ra-226

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Gonads	1.7×10^{-7}	-
Red Marrow	1.1×10^{-6}	-
Bone Surfaces	1.3×10^{-5}	1.4×10^{-5}
Lungs	-	3.0×10^{-5}

TABLE 16. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Po-210 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)		Average Individual (rem)		Population (Person-rem)	
	Class D	Class W	Class D	Class W	Class D	Class W
Kidneys	1.5×10^{-2}	5.0×10^{-3}	7.7×10^{-3}	2.5×10^{-3}	5.6×10^2	1.8×10^2
Liver	2.8×10^{-3}	-	1.4×10^{-3}	-	1.0×10^2	-
Spleen	2.8×10^{-2}	8.6×10^{-3}	1.4×10^{-2}	4.3×10^{-3}	1.0×10^3	3.1×10^2
Lungs	-	1.7×10^{-2}	-	8.4×10^{-3}	-	6.1×10^2

TABLE 17. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Am-241 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)		Average Individual (rem)		Population (Person-rem)	
	Class D	Class W	Class D	Class W	Class D	Class W
Gonads	2.1×10^{-7}	-	1.0×10^{-7}	-	7.5×10^{-3}	-
Red Marrow	1.3×10^{-6}	-	6.4×10^{-7}	-	4.7×10^{-2}	-
Bone Surfaces	1.6×10^{-5}	-	8.0×10^{-6}	-	5.9×10^{-1}	-
Liver	3.5×10^{-6}	-	1.8×10^{-6}	-	1.3×10^{-1}	-

TABLE 18. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Ra-226 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)		Average Individual (rem)		Population (Person-rem)	
	Class D	Class W	Class D	Class W	Class D	Class W
Lungs	1.0×10^{-7}	-	5.1×10^{-8}	-	3.8×10^{-3}	-
Bone Surfaces	4.9×10^{-8}	-	2.4×10^{-8}	-	1.8×10^{-3}	-

TABLE 19. COMMITTED DOSE EQUIVALENTS FOR INGESTION OF

1.7×10^{-6} mCi (6.2×10^{-5} MBq) of Ra-226

Organ	Dose Equivalent (rem)
Gonads	5.7×10^{-4}
Red Marrow	3.8×10^{-3}
Bone Surfaces	4.3×10^{-2}

3.3) CLASS A-3 - STATIC ELIMINATORS OR DETECTORS: IN EQUIPMENT OR PROCESS LINE (LOW TOXICITY SOURCES)

3.3.1) Device Description

This class of static eliminators is also used by industry in a variety of ways. They remove static build-up in areas such as spray paint and ink jet nozzles, plastic laminates, printing presses, and other areas. Detectors are used to sense and measure static charge. They are similar in every way to the A-2 class of static eliminators except typical nuclides are H-3 with an activity of 250 mCi (9250 MBq) and Kr-85 with an activity of 2 mCi (74 MBq). Estimates for distribution of these devices was not readily available. As with the A-2 class, these static eliminators may be attached or portable and are covered by the same section of the CFR. They also are covered under the same leasing arrangements as the previous class of static eliminator.

3.3.2) Scenario Development

The same probabilities were assigned to this class of static eliminators as to the A-2 class of static eliminators (Figure 4).

3.3.3) Dose Assessment

3.3.3.1) 'Intact' Source

External dose - maximum individual

Table 6 shows dose equivalents for exposure to point sources at the activity levels stated in 3.3.1 for Kr-85 and H-3 for 20 weeks at 100 cm. H-3 presents no external hazard; the estimate for Kr-85 under these assumptions is around 10 mrem (0.1 mSv) from the photons. If the Kr-85 source were exposed, the activity would not pose a risk to the skin as a point source beta emitter, because the gas would disperse into the nearby environment.

External dose - realistic case

Because the maximum external dose equivalent was around 10 mrem, the realistic estimates will be much lower and will not be significant.

Internal dose - maximum individual

Tritiated water taken into the body quickly equilibrates with the body water, and is assumed to be distributed uniformly throughout the body and retained with a biological half time of 10 days (ICRP 1978). As for the high toxicity static eliminators, a fraction of 0.3 was arbitrarily chosen as the maximum amount which might inadvertently be taken in. The dose equivalent for intake of 75 mCi (2800 MBq) of H-3 as tritiated water is 4.7 rem (0.047 Sv) to the soft tissue of the whole body. Because H-3 in the form of gaseous hydrogen is usually converted to tritiated water in the atmosphere (Jacobs 1968), dosimetry for H-3 as gaseous hydrogen is not considered in this report.

CLASS A-3 STATIC ELIMINATORS/DETECTORS

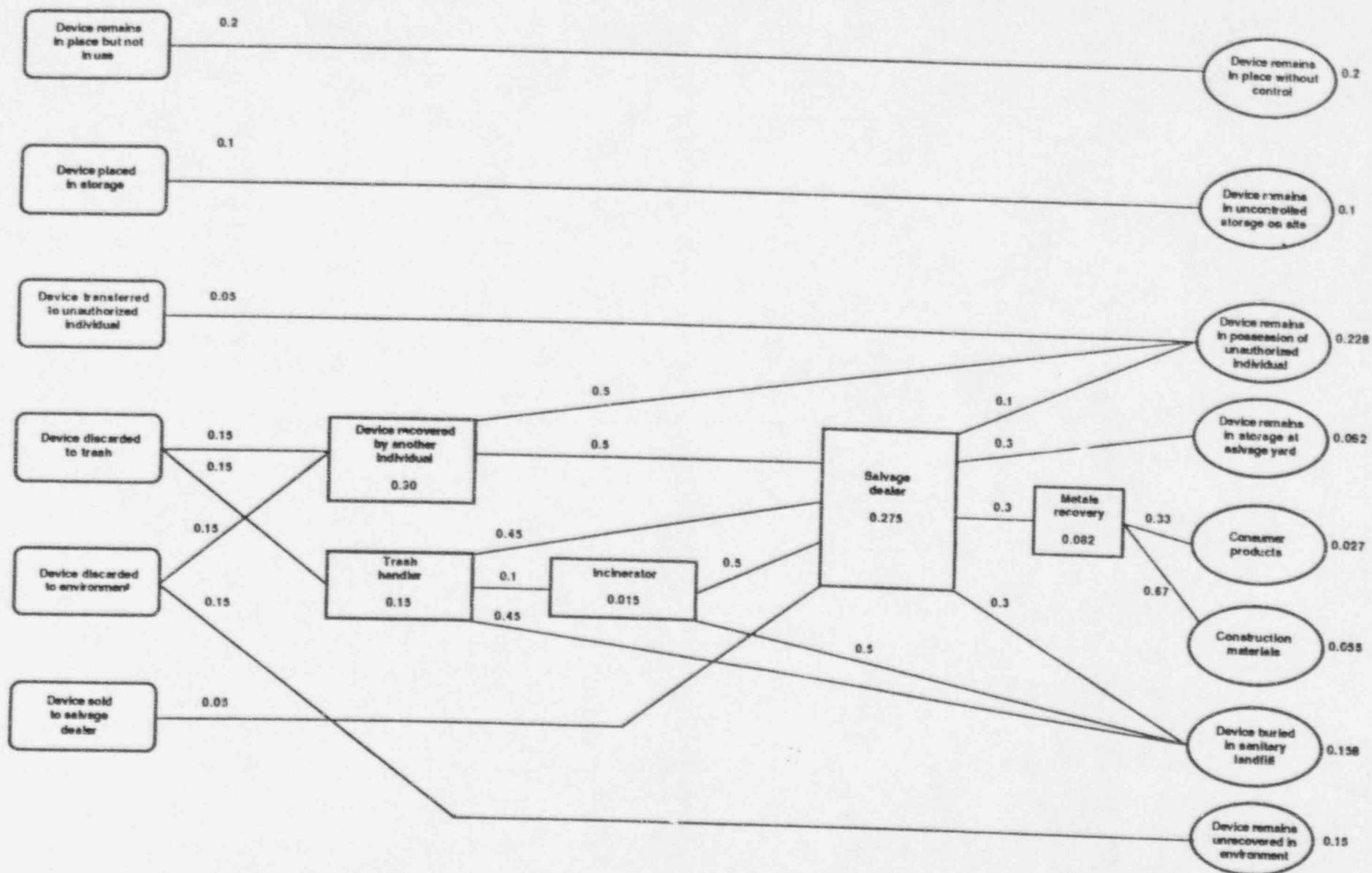


Figure 4 Scenario development for device class A-3 , static eliminators/detectors

The overriding concern in the dosimetry of immersion in clouds of noble gases is the dose to the organs from external irradiation (ICRP 1978). The doses to the major organs from immersion in a cloud of 2 mCi (74 MBq) of Kr-85 in a 100 m³ room are listed table 20. These dose equivalent rates are also extremely low, and no significant internal hazard exists from mishandling of these devices.

Internal dose - realistic individual

The values given by Wilmot for release of various radionuclides cannot be directly applied to tritium. The largest value he gives is 10⁻¹. Using 10⁻¹ with a value of 10⁻³ for uptake, the fraction of the original source activity which would be taken in under this model is 10⁻⁶. For intake of 0.025 mCi (0.925 MBq) of tritiated water, the dose equivalent to the soft tissues of the whole body would be 1.6 x 10⁻³ rem (1.6 x 10⁻⁵ Sv). Dose equivalents for exposure to Kr-85 were very low for the maximum case, so realistic estimates would not be of significance.

3.3.3.2) Dispersed Source

3.3.3.2.1) Incineration

The model of Buckley et al predicts an uptake of 1.7 x 10⁻⁶ mCi/yr (6.4 x 10⁻⁵ Bq/yr) for incineration of 5 devices per year containing 250 mCi (9250 MBq) of H-3, if 100% of the activity is released from the device. This would result in a dose equivalent of 1.1 x 10⁻⁷ rem/yr (1.1 x 10⁻⁹ Sv/yr) to the maximum individual, 5.5 x 10⁻⁸ rem/yr (5.5 x 10⁻¹⁰ Sv/yr) to the average individual, and 4.0 x 10⁻³ person-rem/yr (4.0 x 10⁻⁵ person-Sv/yr) to the population near the incinerator. For Kr-85, 100% release from the device was assumed and no removal of the activity from the stack gases was assumed. These assumptions resulted in a concentration of 1.9 x 10⁻¹¹ mCi/m³ (7.0 x 10⁻⁴ Bq/m³) downwind from the incinerator. Corresponding annual dose equivalents to the maximum and average individual and population from continuous exposure to this concentration are listed in table 21.

3.3.3.2.2) Metals Recycling

As stated in section 2.3.2.2, a general result for Co-60 gauges is the only result afforded by the available data. See that section for a discussion of this result.

3.3.3.2.3) Burial in Landfill

The model of Buckley for transport of radionuclides from a landfill predicts an intake of 0.866 mCi (32 MBq) of H-3 at the withdrawal point near a landfill in which a 250 mCi (9250 MBq) source was buried. Intake of this amount of activity would result in a dose equivalent of 0.054 rem (5.4 x 10⁻⁴ Sv) to the soft tissues of the whole body. No groundwater pathway for Kr-85 was envisioned.

TABLE 20. DOSE EQUIVALENT RATES FOR IMMERSION IN

2 mCi (7.4×10^7 Bq) OF Kr-85 IN A 100 M³ ROOM

Organ	Dose Equivalent (rem/hr)
Gonads	8.9×10^{-7}
Breast	8.1×10^{-7}
Red Marrow	9.6×10^{-7}
Lungs	7.1×10^{-7}
Bone Surfaces	1.0×10^{-6}
Stomach Wall	7.2×10^{-7}
Kidneys	6.7×10^{-7}
Liver	6.3×10^{-7}
Spleen	7.4×10^{-7}
Adrenals	6.6×10^{-7}
Skin	3.4×10^{-3}
Lens of the eye	1.5×10^{-6}

TABLE 21. ANNUAL DOSE EQUIVALENTS FOR INHALATION

OF Kr-85 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)	Average Individual (rem)	Population (Person-rem)
Gonads	3.2×10^{-10}	1.6×10^{-10}	1.2×10^{-5}
Breast	2.8×10^{-10}	1.4×10^{-10}	1.0×10^{-5}
Red Marrow	3.5×10^{-10}	1.7×10^{-10}	1.3×10^{-5}
Lungs	2.6×10^{-10}	1.3×10^{-10}	9.6×10^{-6}
Bone Surfaces	3.8×10^{-10}	1.9×10^{-10}	1.4×10^{-5}
Stomach Wall	2.6×10^{-10}	1.3×10^{-10}	9.6×10^{-6}
Kidneys	2.4×10^{-10}	1.2×10^{-10}	9.0×10^{-6}
Liver	2.3×10^{-10}	1.2×10^{-10}	8.5×10^{-6}
Spleen	2.8×10^{-10}	1.4×10^{-10}	1.0×10^{-5}
Adrenals	2.4×10^{-10}	1.2×10^{-10}	8.8×10^{-6}
Skin	2.9×10^{-8}	1.4×10^{-8}	1.1×10^{-3}
Lens of the eye	3.6×10^{-10}	1.8×10^{-10}	1.3×10^{-5}

3.4) CLASS B - GAMMA GAUGES

3.4.1) Device Description

Gamma gauges contain sources which are mounted with a detector on a production line for monitoring and quality control. The emitted or reflected rays from a radioactive source are sensed by a detector, and the intensity of the signal may be translated into a measure of the thickness, density, etc. of the material studied. The limitation of this study to sources less than 20 mCi (740 MBq) removes some of the popular devices from consideration. Radionuclides commonly used include Co-60, Cs-137, Am-241, and Ra-226. It is estimated that some 50,000 industrial gauging devices are probably in use today (Peters 85).

Gamma gauges are normally not removed from the process lines. Because of the work environments commonly encountered, there is significant potential for damage of the devices and sources in the workplace. The corrosive atmosphere of industrial lines, rough treatment, or poor maintenance could hide or destroy labeling which indicates the presence of a radioactive source. These labels may also be painted over or removed. Gamma gauges are often hard to distinguish from other parts on the processing line and vary greatly in size (Peters 1985). Devices are often lost because the persons familiar with them have left the plant or the plant has closed. These lost gauges may be discarded in landfills or included with scrap metal (Lubenau 1986).

Several incidents have occurred in which steel products have become contaminated by radioactivity assumed to be from a gamma gauge. In 1984, significant levels of Co-60 were detected in steel fittings imported from Taiwan; the source was traced to the inadvertent mixing of a source with scrap feed. In 1984, a Cs-137 gauge in a South Carolina steel plant was struck and melted by a stream of molten steel. No radioactive products were released, and all contamination was confined to the plant site. An event in California in 1985 involved a Cs-137 gauge mixed with scrap metal, again in which most of the activity was kept on the plant site. Four Cs-137 level and density gauges were lost by a licensee in Alabama in 1985; one gauge was recovered from a nearby scrapyard and contaminated steel products were detected at another location (Lubenau 1986).

3.4.2) Scenario Development

Figure 5 shows the probabilities assigned to the events and pathways for gamma gauges. Highest probabilities for pathways leading from initial events were assigned to the device remaining in place (because these devices are designed to be fixed on a process line) and to the device being recovered by a trash handler after being discarded into the trash (0.3). Lower values (0.1) were assigned to the device being placed in storage or transferred to an unauthorized individual or directly to the salvage dealer. The apparent value to persons not familiar with their use was thought to be in the recoverable metals (steel and lead). Therefore, the pathway leading from recovery by an

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CLASS B - GAMMA GAUGES

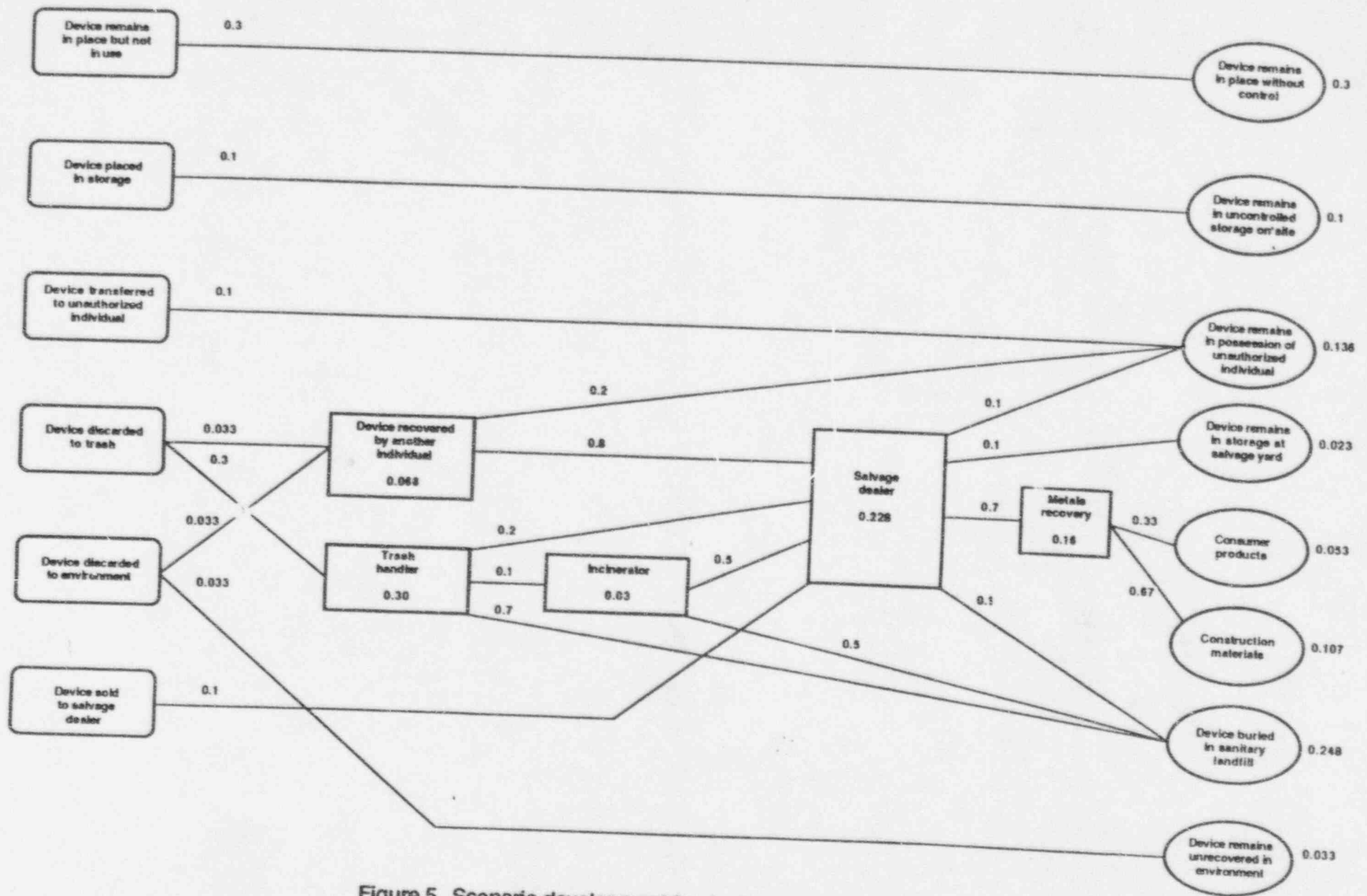


Figure 5 Scenario development for device class B, gamma gauges

individual to the salvage dealer was weighted more heavily (0.8) than that for the individual keeping the device (0.2). If the device were discarded into the trash, the most probable event was assumed to be disposal in a landfill (0.7) if not recovered from the trash; if recovered, the pathway leading to the salvage dealer was again weighted more heavily. Because of the high metal content, the pathway from the salvage dealer to metals recycling was weighted the most heavily (0.7), with the other pathways weighted less heavily (0.1 each). Of materials sent to metals recycling, approximately two-thirds (0.67) go to construction materials and one-third (0.33) go to consumer products (Metal Statistics 1986). No adequate information was available to assess the probability regarding the flow of materials after incineration, so probabilities of 0.5 were assigned to both pathways leading from this point.

Predicted final probabilities are highest for burial in sanitary landfill (0.25), remaining in place without control (0.30), and passing through metal recycling (0.16 total). These results show some of the trends noted by Lubenau et al. (Lubenau 1986) for final disposition of these devices if mishandled.

3.4.3) Dose Assessment

3.4.3.1) 'Intact' source

Gamma gauges typically employ radioactive sources enclosed in a well-shielded geometry with a mechanical shutter which moves back and forth to expose or cover the source. Because a built-in mechanism exposes the source, a person not familiar with safe handling of these devices could conceivably expose the source and receive a significant radiation dose.

External dose - maximum individual

Table 6 shows the estimated radiation doses for exposure to these sources at 100 cm for 20 weeks or on contact for three hours.

External dose - realistic case

External exposure rates around several Cs-137 gauges with sources containing more than 20 mCi (740 MBq) were quoted in the NRC's Sealed Source and Device Registry. The devices were shielded so that the external exposure rates were less than 10 mR hr^{-1} ($2.6 \text{ uC kg}^{-1} \text{ hr}^{-1}$) on contact and less than 5 mR hr^{-1} ($1.3 \text{ uC kg}^{-1} \text{ hr}^{-1}$) at 1 foot (30.5 cm) from the device with the shutter closed. Because of the lower activity of the devices considered in this report, continuous exposure to the shuttered device would probably not result in annual dose equivalents exceeding 500 mrem/yr (5 mSv/yr). If the shutter were briefly opened, dose equivalents for photon exposure would be less than those quoted in the previous section for a 20 week (3400 hour) exposure, in proportion to the time of exposure.

Internal dose - maximum individual

For Co-60, ICRP 30 (ICRP 1979) assigns an f_1 value (fraction of material in the gastrointestinal system which passes into the bloodstream) of 0.05 to oxides and hydroxides, and an f_1 value of 0.3 to all other compounds. Oxides, hydroxides, halides, and nitrates were assigned to inhalation class Y; all others were assigned to inhalation class W. For Cs-137, all classes were assigned an f_1 value of 1.0 and all compounds were assigned to inhalation class D. For Ra-226, all compounds were assigned an f_1 value of 0.2 and an inhalation class of W. For Am-241, all compounds were assigned an f_1 value of 5×10^{-4} and an inhalation class of W. As discussed in section 3.2.3.1, a fraction of 0.3 was chosen as the maximum fraction of the source activity that an individual might take in. Tables 22-25 list the organ dose equivalents resulting from this assumption.

Internal dose - realistic case

In most circumstances, the activity in the gauges would not be released. If a gauge were exposed to a corrosive atmosphere or otherwise mistreated, a small fraction of the activity might be lost from the source in the form of removable contamination. Tables 26-29 list the committed dose equivalents which would result from intake of 10^{-4} of the assumed source activity for Co-60, and 10^{-6} of the source activity for Cs-137, Ra-226, and Am-241.

3.4.3.2) Dispersed source

3.4.3.2.1) Incineration

From table 5, stack release fractions of 0.273 and 0.525 were assigned to Co-60 and Cs-137, respectively, for use in the model of Buckley et al. for incineration. Fractions of 0.5 were assigned to Ra-226 and Am-241, because this was the approximate maximum for nuclides other than those for which very high release fractions were observed (H-3, C-14, I-125, and Tl-204). The total activity inhaled, based on the model of Buckley et al. (NUREG 1980), is 1.9×10^{-8} mCi (7.0×10^{-7} MBq) of Co-60, 7.3×10^{-8} mCi (2.7×10^{-6} MBq) of Cs-137, 6.9×10^{-8} mCi (2.6×10^{-6} MBq) of Am-241, and 3.5×10^{-8} mCi (1.3×10^{-6} MBq) of Ra-226. Tables 30-33 list the maximum individual, average individual, and population committed dose equivalents from inhalation of these activity levels.

3.4.3.2.2) Metals Recycling

As stated in section 2.3.2.2, a general result for Co-60 gauges is the only result afforded by the available data. See that section for a discussion of this result.

3.4.3.2.3) Burial in Landfill

The total activity reaching the withdrawal point was calculated to be very low for Cs-137, Co-60, and Am-241 (as the equation predicted radioactive decay before reaching the withdrawal point) and 0.033 mCi

(1.2 MBq) of Ra-226. Table 34 lists committed dose equivalents for ingestion of 0.033 mCi (1.2 MBq) of Ra-226.

TABLE 22. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION

OF 3 mCi (111 MBq) OF Co-60

Organ	Dose Equivalent (rem)			
	Ingestion		Inhalation	
	$f_1=0.05$	$f_1=0.3$	Class W	Class Y
Gonads	36	80	44	-
Breast	12	57	47	-
Red Marrow	14	61	47	-
Lungs	9.7	56	400	3800
Small Intestine	40	91	-	-
Upper Large Intestine	63	110	-	-
Lower Large Intestine	120	160	-	-
Liver	26	140	91	-
Remainder of body	23	97	100	-
			89	-

TABLE 23. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION

OF 6 mCi (222 MBq) OF Cs-137

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Gonads	310	200
Breast	270	170
Red Marrow	290	180
Lungs	290	200
Thyroid	290	180
Bone Surfaces	290	180
Small Intestine	310	200
Upper Large Intestine	310	200
Lower Large Intestine	310	200
Remainder of body	330	210

TABLE 24. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION
OF 3 mCi (111 MBq) OF Ra-226

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Gonads	1000	-
Red Marrow	6700	-
Bone Surfaces	75000	84000
Lungs	-	180000

TABLE 25. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION
OF 6 mCi (222 MBq) of Am-241

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Gonads	3100	710000
Red Marrow	19000	4400000
Bone Surfaces	240000	56000000
Liver	51000	12000000

TABLE 26. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION

OF 1 μCi (37 kBq) OF Co-60

Organ	Dose Equivalent (mrem)			
	Ingestion $f_1=0.05$	Ingestion $f_1=0.3$	Inhalation Class W	Inhalation Class Y
Gonads	12	27	15	-
Breast	4.1	19	16	-
Red Marrow	4.8	20	16	-
Lungs	3.2	18	130	1300
Small Intestine	13	30	-	-
Upper Large Intestine	21	36	-	-
Lower Large Intestine	41	52	30	-
Liver	8.5	48	34	-
Remainder of body	7.8	32	30	-

TABLE 27. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION

OF 20 nCi (740 Bq) OF Cs-137

Organ	Dose Equivalent (mrem)	
	Ingestion	Inhalation
Gonads	1.0	0.65
Breast	0.89	0.58
Red Marrow	0.96	0.61
Lungs	0.96	0.65
Thyroid	0.96	0.58
Bone Surfaces	0.96	0.58
Small Intestine	1.0	0.67
Upper Large Intestine	1.0	0.67
Lower Large Intestine	1.0	0.67
Remainder of body	1.1	0.70

TABLE 28. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION

OF 10 nCi (370 Bq) OF Ra-226

Organ	<u>Dose Equivalent (mrem)</u>	
	Ingestion	Inhalation
Gonads	3.4	-
Red Marrow	22	-
Bone Surfaces	250	280
Lungs	-	590

TABLE 29. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION

OF 20 nCi (740 Bq) of Am-241

Organ	<u>Dose Equivalent (mrem)</u>	
	Ingestion	Inhalation
Gonads	10	2400
Red Marrow	62	15000
Bone Surfaces	810	180000
Liver	170	41000

TABLE 30. COMMITTED DOSE EQUIVALENTS FOR INHALATION

OF Co-60 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (urem)		Average Individual (urem)		Population (person-rem)	
	Class W	Class Y	Class W	Class Y	Class W	Class Y
Gonads	0.30	-	0.15	-	0.011	-
Breast	0.31	-	0.16	-	0.012	-
Red Marrow	0.31	-	0.16	-	0.012	-
Lungs	2.7	25	1.3	13	0.095	0.95
Lower Large Intestine	0.61	-	0.30	-	0.022	-
Liver	0.68	-	0.34	-	0.025	-
Remainder of body	0.60	-	0.30	-	0.022	-

TABLE 31. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Cs-137 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (urem)	Average Individual (urem)	Population (person-rem)
Gonads	2.4	1.2	0.087
Breast	2.1	1.1	0.077
Red Marrow	2.2	1.1	0.082
Lungs	2.4	1.2	0.087
Thyroid	2.1	1.1	0.078
Bone Surfaces	2.1	1.1	0.078
Small Intestine	2.5	1.2	0.090
Upper Large Intestine	2.4	1.2	0.089
Lower Large Intestine	2.5	1.2	0.090
Remainder of body	2.6	1.3	0.094

TABLE 32. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Ra-226 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (mrem)	Average Individual (mrem)	Population (person-rem)
Lungs	2.1	1.0	76
Bone Surfaces	0.99	0.49	36

TABLE 33. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Am-241 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (mrem)	Average Individual (mrem)	Population (person-rem)
Gonads	8.3	4.1	300
Red Marrow	52	26	1900
Bone Surfaces	650	320	24000
Liver	140	71	5200

TABLE 34. COMMITTED DOSE EQUIVALENTS FOR INGESTION OF
0.033 mCi (1.2 MBq) OF Ra-226

Organ	Dose Equivalent (rem)
Gonads	11
Red Marrow	72
Bone Surfaces	820

3.5) CLASS C-1 - BETA GAUGES: BACKSCATTER TYPE

3.5.1) Device Description

These devices are widely used in production facilities in monitoring process lines or in measuring thickness, density, or composition of such materials as plastic, paper, steel sheets, precious metal platings, plating of circuit boards, and plastic coatings. Devices can be permanently mounted or portable. Nuclides common to the beta backscatter gauges include Sr-90, C-14, Pm-147, Tl-204, Ru-106, and Pb-210. Radiotoxicity ranges from moderate to very high. Approximately 1,200 beta backscatter gauges are distributed per year.

Because these devices may be portable with the sources being interchangeable, the beta backscatter gauges may be easily misplaced. They may be lost or discarded in the trash. If found, they may appear to have some salvage value, and thus may be transferred to a salvage dealer.

3.5.2) Scenario Development

The probabilities assigned to the various pathways (Figure 6) were based on the appearance of the beta backscatter gauges, their size, their portability, and their apparent salvage value. The highest probabilities were assigned to the pathways leading from the initial event of the device being discarded to the trash (0.21 each that the device would be recovered by another individual or remain in the trash) and to the pathways in which the device remains in place or in storage on site (0.15 each). Because their external appearance might lead some to believe that the devices might have some salvage value, the pathway leading from the device being recovered by an individual to the salvage dealer was weighted more heavily (0.8). It was thought about equally probable that the device would be kept by the salvage dealer, sent to metals recycling, or buried in a landfill, but less probable that the salvage dealer would give it to another individual. Therefore, the probability of 0.3 was assigned to each of the former pathways and 0.1 to the latter.

These assumptions indicate that the highest probabilities are that the devices would be buried in a sanitary landfill (0.28), remain in place without control (0.15), or remain uncontrolled in storage (0.15). The predicted probability of possession by an unauthorized individual is also relatively high (0.15) due to their attractive appearance. The interchangeable sources and portability of the gauges also raises the probability of possession by unauthorized individuals. All other events appear to be less likely. These events include becoming used in consumer products or construction materials (0.12 total) or remaining in storage at the salvage yard (0.12).

CLASS C-1 BETA BACKSCATTER GAUGES

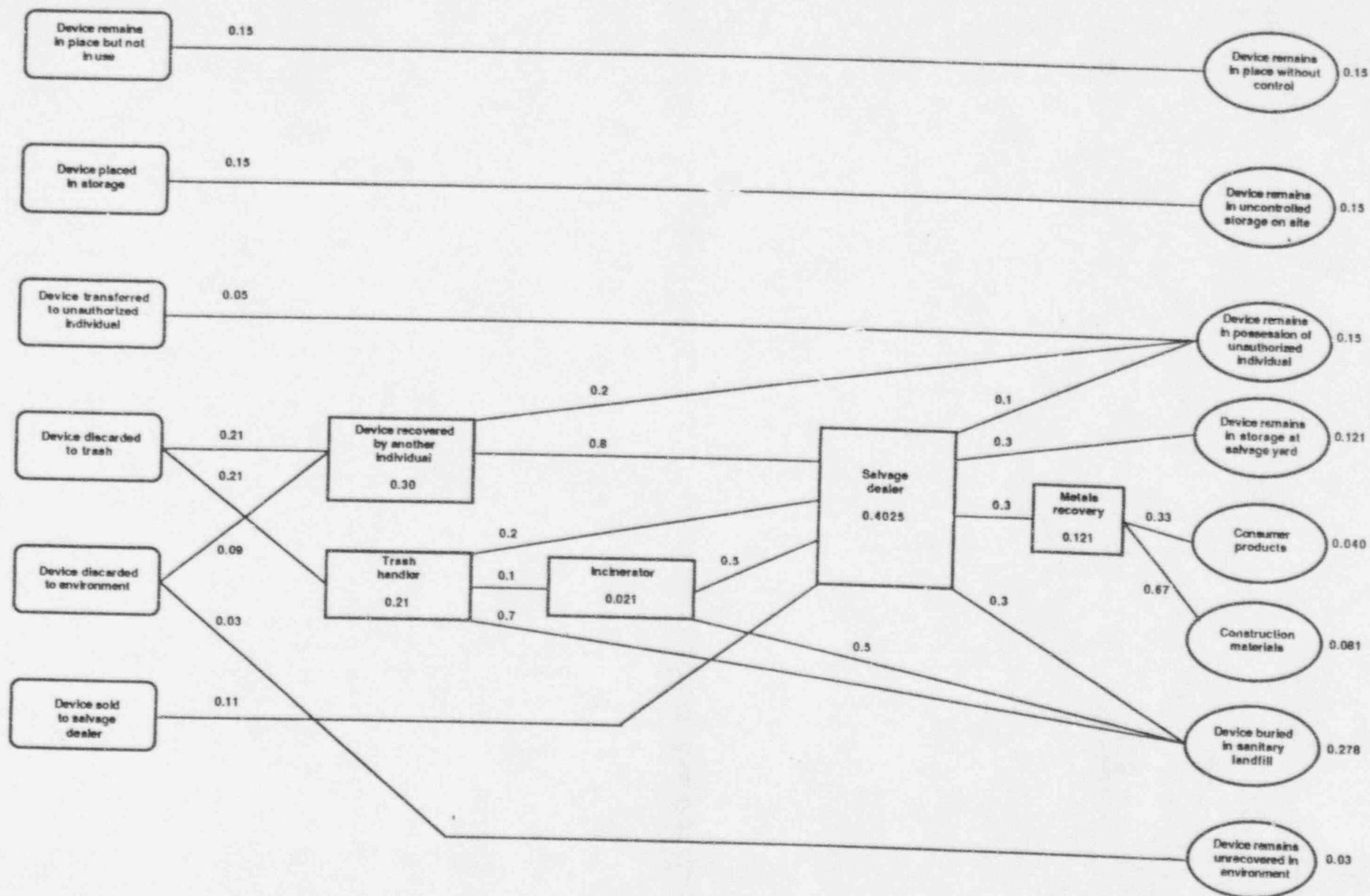


Figure 6 Scenario development for device class C-1, beta backscatter gauges

3.5.3) Dose Assessment

3.5.3.1) 'Intact' Source

External dose - maximum individual

Dose equivalents for exposure to point sources for 20 weeks at 100 cm (photon) or in contact for three hours (beta) at the activity levels stated in section 3.5.1 for Sr-90, Tl-204, Ru-106, Pm-147, C-14, and Pb-210 are shown in table 6.

External dose - realistic case

External photon dose rates near the devices were very low, as listed in the NRC registry. The beta dose rates listed in table 6 are high enough for transitory erythema or ulceration and moist desquamation to occur, based on the estimates in NCRP 39 (NCRP 1971). More realistic estimates could be based on other arbitrary estimates of contact time less than three hours, with the estimates in table 6 scaled accordingly. It is still possible that estimates might exceed the limits for radiation workers for dose to the extremities (18.75 rem per calendar quarter). These devices present an appreciable hazard for external irradiation.

Internal dose - maximum individual

ICRP 30 assigns all soluble compounds of Sr-90 to inhalation class D with an f_1 of 0.3. SrTiO_3 is assigned to class Y with an f_1 of 0.01. All compounds of carbon are assumed to be instantaneously and uniformly distributed throughout all organs and tissues of the body where they are retained indefinitely. For Pm-147, oxides, hydroxides, carbides, and fluorides are assigned to inhalation class Y and all others are assigned to class W. All compounds are assigned an f_1 of 0.0003. All compounds of thallium are assigned to class D with an f_1 of 1.0. All compounds of Ru-106 are assigned an f_1 of 0.05. Oxides and hydroxides of ruthenium are assigned to class Y, halides to class W, and all others to class D. All commonly occurring compounds of lead are assigned to class D, with an f_1 of 0.2.

As for the previously discussed devices with industrial applications, a fraction of 0.3 was assigned as the maximum amount of the total source activity which might be inadvertently taken into the body. Tables 35-39 list the dose equivalents corresponding to intake of 0.3 of the source activities listed in table 1. The committed dose equivalent for intake of 0.015 mCi (0.56 MBq) of C-14 is 0.031 rem (0.31 mSv) to any organ.

Internal dose - realistic case

Intake values from the study of Wilmot were used unless they were lower than the 10^{-6} value suggested by Ricks et al., in which case an intake value of 10^{-6} was assigned. This scheme results in the assignment of fractions of 10^{-6} to all nuclides except C-14, for which a fraction of 10^{-4} was used. Tables 40-44 list the dose equivalents for

intake of these fractions of the initial source activity. The committed dose equivalent for intake of 5×10^{-6} mCi (1.85×10^{-4} MBq) of C-14 is 1.0×10^{-5} rem (1.0×10^{-7} Sv) to any organ.

3.5.3.2) Dispersed Source

3.5.3.2.1) Incineration

The model of Buckley predicts the intake of 8.7×10^{-11} mCi (3.2×10^{-9} MBq) of Sr-90, 6.9×10^{-10} mCi (2.6×10^{-8} MBq) of Tl-204, 2.9×10^{-11} mCi (1.1×10^{-9} MBq) of Ru-106, 1.7×10^{-10} mCi (6.4×10^{-9} MBq) of Pm-147, 3.5×10^{-10} mCi (1.3×10^{-8} MBq) of C-14, and 3.5×10^{-11} mCi (1.3×10^{-9} MBq) of Pb-210 by the maximum individual. Tables 45-49 list the dose equivalents corresponding to these intakes to the maximum and average individuals and the population near the incinerator. The committed dose equivalents from inhalation of C-14 to the maximum individual, average individual, and population are 7.3×10^{-10} rem (7.3×10^{-12} Sv), 3.6×10^{-10} rem (3.6×10^{-12} Sv), and 2.6×10^{-5} person-rem (2.6×10^{-7} person-Sv), respectively.

3.5.3.2.2) Metals Recycling

As stated in section 2.3.2.2, a general result for Co-60 gauges is the only result afforded by the available data. See that section for a discussion of this result.

3.5.3.2.3) Burial in Landfill

The model of Buckley did not list values of K for several radionuclides considered in this section. A value of 1.0 was assigned in all such cases. The model then predicted intakes of 1.8×10^{-4} mCi (6.7×10^{-3} MBq) of Sr-90, 6.2×10^{-5} mCi (2.3×10^{-3} MBq) of Tl-204, 8.7×10^{-8} mCi (3.2×10^{-6} MBq) of Ru-106, 1.24×10^{-3} mCi (4.6×10^{-2} MBq) of C-14, and 6.0×10^{-5} mCi (2.2×10^{-3} MBq) of Pb-210 at the withdrawal point. The model predicted that all of the Pm-147 would decay before reaching the withdrawal point. Dose equivalents corresponding to these intakes are listed in tables 50-53. The committed dose equivalent for ingestion of this amount of C-14 is 2.6×10^{-3} rem (2.6×10^{-5} Sv) to any organ.

TABLE 35. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION OF

0.0075 mCi (2.8×10^5 Bq) Sr-90

Organ	Dose Equivalent (rem)			
	Ingestion		Inhalation	
	$f_1=0.3$	$f_1=0.01$	Class D	Class Y
Red Marrow	5.3×10^0	1.8×10^{-1}	9.2×10^0	-
Bone Surfaces	1.2×10^1	3.9×10^{-1}	2.0×10^1	-
Upper Large Intestine	-	1.7×10^{-1}	-	-
Lower Large Intestine	-	7.2×10^{-1}	-	-
Lungs	-	-	-	8.0×10^1

TABLE 36. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION

OF 0.03 mCi (1.1×10^6 Bq) OF Tl-204

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Gonads	7.3×10^{-2}	4.6×10^{-2}
Breast	7.3×10^{-2}	4.6×10^{-2}
Red Marrow	7.3×10^{-2}	4.7×10^{-2}
Lungs	7.3×10^{-2}	1.2×10^{-1}
Stomach Wall	1.0×10^{-1}	5.1×10^{-2}
Small Intestine	7.3×10^{-2}	4.6×10^{-2}
Kidneys	5.1×10^{-1}	3.2×10^{-1}
Remainder of Body	7.3×10^{-2}	4.6×10^{-2}

TABLE 37. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION
OF 0.0075 mCi (2.8×10^5 Bq) OF Ru-106

Organ	Dose Equivalent (rem)			
	Ingestion	Class D	Inhalation Class W	Class Y
Gonads	-	3.9×10^{-1}	-	-
Breast	-	3.9×10^{-1}	-	-
Red Marrow	-	3.9×10^{-1}	-	-
Lungs	-	5.0×10^{-1}	5.8×10^0	2.8×10^1
Stomach Wall	-	3.9×10^{-1}	-	-
Small Intestine	-	4.2×10^{-1}	-	-
Upper Large Intestine	6.9×10^{-1}	4.7×10^{-1}	-	-
Lower Large Intestine	2.0×10^0	6.9×10^{-1}	-	-
Thyroid	-	3.9×10^{-1}	-	-
Bone Surfaces	-	3.9×10^{-1}	-	-
Remainder of Body	-	3.9×10^{-1}	-	-

TABLE 38. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION
OF 0.015 mCi (5.6×10^5 Bq) OF Pm-147

Organ	Dose Equivalent (rem)		
	Ingestion	Inhalation Class W	Class Y
Upper Large Intestine	6.1×10^{-2}	-	-
Lower Large Intestine	1.8×10^{-1}	-	-
Red Marrow	-	4.6×10^{-1}	-
Bone Surfaces	-	5.6×10^0	-
Lungs	-	5.4×10^{-1}	4.3×10^0
Liver	-	1.5×10^0	-

TABLE 39. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION
OF $0.003 \text{ mCi } (1.1 \times 10^5 \text{ Bq})$ OF Pb-210

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Kidneys	3.1×10^1	7.9×10^1
Liver	6.8×10^1	1.7×10^2
Red Marrow	1.6×10^1	4.1×10^1
Bone Surfaces	2.4×10^2	6.1×10^2

TABLE 40. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION
OF $2.5 \times 10^{-8} \text{ mCi } (0.925 \text{ MBq})$ Sr-90

Organ	Dose Equivalent (rem)			
	Ingestion		Inhalation	
	$f_1=0.3$	$f_1=0.01$	Class D	Class Y
Red Marrow	1.8×10^{-5}	5.9×10^{-7}	3.1×10^{-5}	-
Bone Surfaces	3.9×10^{-5}	1.3×10^{-6}	6.8×10^{-5}	-
Lungs	-	-	-	2.7×10^{-4}
Upper Large Intestine	-	5.6×10^{-7}	-	-
Lower Large Intestine	-	2.4×10^{-6}	-	-

TABLE 41. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION
OF $5 \times 10^{-8} \text{ mCi } (1.85 \text{ Bq})$ OF Pm-147

Organ	Dose Equivalent (rem)		
	Ingestion	Inhalation	
		Class W	Class Y
Upper Large Intestine	2.0×10^{-7}	-	-
Lower Large Intestine	5.9×10^{-7}	-	-
Red Marrow	-	1.5×10^{-6}	-
Bone Surfaces	-	1.8×10^{-5}	-
Lungs	-	1.8×10^{-6}	1.4×10^{-5}
Liver	-	5.0×10^{-6}	-

TABLE 42. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION

OF 1×10^{-7} mCi (3.7 Bq) OF Tl-204

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Gonads	2.4×10^{-7}	1.5×10^{-7}
Breast	2.4×10^{-7}	1.5×10^{-7}
Red Marrow	2.4×10^{-7}	1.6×10^{-7}
Lungs	2.4×10^{-7}	4.1×10^{-7}
Stomach Wall	3.4×10^{-7}	1.7×10^{-7}
Small Intestine	2.4×10^{-7}	1.5×10^{-7}
Kidneys	1.7×10^{-6}	1.1×10^{-6}
Remainder of Body	2.4×10^{-7}	1.5×10^{-7}

TABLE 43. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION

OF 2.5×10^{-8} mCi (0.925 MBq) OF Ru-106

Organ	Dose Equivalent (rem)			
	Ingestion	Class D	Inhalation Class W	Class Y
Stomach Wall	-	1.3×10^{-6}	-	-
Small Intestine	-	1.4×10^{-6}	-	-
Upper Large Intestine	2.3×10^{-6}	1.6×10^{-6}	-	-
Lower Large Intestine	6.6×10^{-6}	2.3×10^{-6}	-	-
Lungs	-	1.7×10^{-6}	1.9×10^{-5}	9.2×10^{-5}
Gonads	-	1.3×10^{-6}	-	-
Breast	-	1.3×10^{-6}	-	-
Red Marrow	-	1.3×10^{-6}	-	-
Thyroid	-	1.3×10^{-6}	-	-
Bone Surfaces	-	1.3×10^{-6}	-	-
Remainder	-	1.3×10^{-6}	-	-

TABLE 44. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION

OF 2×10^{-5} mCi (0.37 MBq) OF Pb-210

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Kidneys	1.0×10^{-4}	2.6×10^{-4}
Liver	2.3×10^{-4}	5.5×10^{-4}
Red Marrow	5.6×10^{-5}	1.4×10^{-4}
Bone Surfaces	8.1×10^{-4}	2.0×10^{-3}

TABLE 45. COMMITTED DOSE EQUIVALENTS FOR INHALATION

OF Sr-90 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)		Average Individual (rem)		Population (Person-rem)	
	Class D	Class Y	Class D	Class Y	Class D	Class Y
Red Marrow	1.1×10^{-7}	-	5.3×10^{-8}	-	3.9×10^{-3}	-
Bone Surfaces	2.3×10^{-7}	-	1.2×10^{-7}	-	8.6×10^{-3}	-
Lungs	-	9.3×10^{-7}	-	4.7×10^{-7}	-	3.4×10^{-2}

TABLE 46. COMMITTED DOSE EQUIVALENTS FOR INHALATION

OF Tl-204 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)		Average Individual (rem)		Population (Person-rem)	
Gonads	1.1×10^{-9}		5.3×10^{-10}		3.8×10^{-5}	
Breast	1.1×10^{-9}		5.3×10^{-10}		3.8×10^{-5}	
Red Marrow	1.1×10^{-9}		5.4×10^{-10}		3.9×10^{-5}	
Lungs	2.8×10^{-9}		1.4×10^{-9}		1.0×10^{-4}	
Stomach Wall	1.2×10^{-9}		5.9×10^{-10}		4.3×10^{-5}	
Small Intestine	1.1×10^{-9}		5.3×10^{-10}		3.8×10^{-5}	
Kidney	7.4×10^{-9}		3.7×10^{-9}		2.7×10^{-4}	
Remainder of Body	1.1×10^{-9}		5.3×10^{-10}		3.8×10^{-5}	

TABLE 47. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Ru-106 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)		Average Individual (rem)		Population (Person-rem)	
	Class W	Class Y	Class W	Class Y	Class W	Class Y
Lungs	2.3×10^{-8}	1.1×10^{-7}	1.1×10^{-8}	5.4×10^{-8}	8.2×10^{-4}	3.9×10^{-3}

TABLE 48. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Pm-147 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)		Average Individual (rem)		Population (Person-rem)	
	Class W	Class Y	Class W	Class Y	Class W	Class Y
Red Marrow	5.3×10^{-9}	-	2.6×10^{-9}	-	1.9×10^{-4}	-
Lungs	6.2×10^{-9}	1.0×10^{-8}	3.1×10^{-9}	2.5×10^{-8}	2.3×10^{-4}	1.8×10^{-3}
Bone Surfaces	6.4×10^{-5}	-	3.2×10^{-8}	-	2.3×10^{-3}	-
Liver	1.7×10^{-8}	-	8.7×10^{-9}	-	6.3×10^{-4}	-

TABLE 49. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Pb-210 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)	Average Individual (rem)	Population (Person-rem)
Red Marrow	4.8×10^{-7}	2.4×10^{-7}	1.7×10^{-2}
Bone Surfaces	7.1×10^{-6}	3.5×10^{-6}	2.6×10^{-1}
Kidneys	9.1×10^{-7}	4.6×10^{-7}	3.3×10^{-2}
Liver	1.9×10^{-6}	9.6×10^{-7}	7.0×10^{-2}

TABLE 50. COMMITTED DOSE EQUIVALENTS FOR INGESTION OF

1.8×10^{-4} mCi OF Sr-90

Organ	Dose Equivalent (rem)	
	$f_1 = 0.3$	$f_1 = 0.01$
Red Marrow	0.13	4.3×10^{-3}
Bone Surfaces	0.28	9.5×10^{-3}
Upper Large Intestine	-	4.1×10^{-3}
Lower Large Intestine	-	1.8×10^{-2}

TABLE 51. COMMITTED DOSE EQUIVALENTS FOR INGESTION OF

6.2×10^{-5} mCi OF ^{222}Rn

Organ	Dose Equivalent (rem)
Gonads	1.5×10^{-4}
Breast	1.5×10^{-4}
Red Marrow	1.5×10^{-4}
Lungs	1.5×10^{-4}
Stomach Wall	2.1×10^{-4}
Small Intestine	1.5×10^{-4}
Kidneys	1.1×10^{-3}
Remainder of Body	1.5×10^{-4}

TABLE 52. COMMITTED DOSE EQUIVALENTS FOR INGESTION OF

8.7×10^{-8} mCi OF Ru-106

Organ	Dose Equivalent (rem)
Upper Large Intestine	8.0×10^{-6}
Lower Large Intestine	2.3×10^{-5}

TABLE 53. COMMITTED DOSE EQUIVALENTS FOR INGESTION OF

6.0×10^{-5} mCi OF Pb-210

Organ	Dose Equivalent (rem)
Red Marrow	0.33
Bone Surfaces	4.8
Kidneys	0.62
Liver	1.4

3.6) CLASS C-2 - BETA GAUGES: TRANSMISSION TYPE

3.6.1) Device Description

Devices of this type are used to measure thickness, density, or composition of materials on process lines. These transmission beta gauges are typically mounted permanently as opposed to the more portable backscatter beta gauges. Sr-90 is the source most often used in these devices. A number of gauges are excluded from this study by limiting the activity of the source to less than 20 mCi (740 MBq). The physical characteristics of these devices are very similar to those of the gamma gauges, except that more metal is probably needed for shielding in some of the gamma gauges.

3.6.2) Scenario Development

As with the gamma gauges, these devices are meant to be permanently mounted on production lines until they are no longer of use. For these reasons, the same probabilities assigned to the gamma gauges were assigned to the beta transmission gauges, and the resultant probabilities for final device status were identical (Figure 7).

3.6.3) Dose Assessment

3.6.3.1) 'Intact' Source

External dose - maximum individual

Table 6 shows dose equivalents for exposure to point sources at the activity levels stated in section 3.6.1 for Sr-90 in contact for three hours.

External dose - realistic case

As with the beta backscatter gauges, the dose equivalents for skin contact are very high even with a short exposure, and may produce permanent effects to the skin should an exposure of more than a few minutes take place. Because the devices are designed so that the activity may be exposed, it is conceivable that someone not knowledgeable about the hazards might receive a significant exposure.

Internal dose - maximum individual

As with other gauges, a fraction of 0.3 was arbitrarily assigned as the maximum amount of the source activity which might be inadvertently taken into the body. Table 54 lists dose equivalents for intake of 6 mCi (222 MBq) of Sr-90.

Internal dose - realistic case

An intake value of 10^{-6} was assigned to Sr-90, as described in section 3.5.3.1. Dose estimates for intake of 2×10^{-5} mCi (7.4×10^{-4} MBq) of Sr-90 are listed in table 55.

CLASS C-2 BETA TRANSMISSION GAUGES

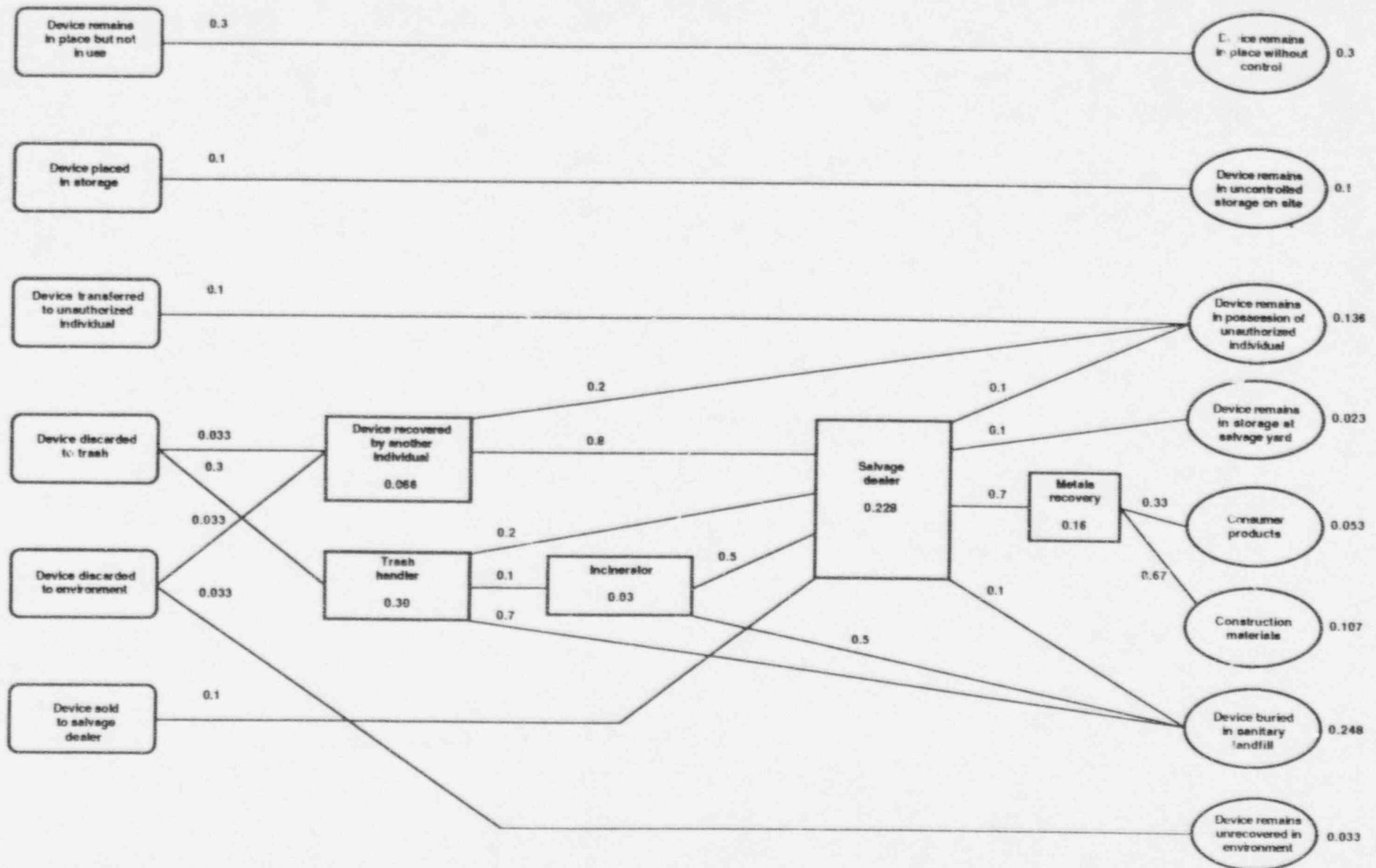


Figure 7 Scenario development for device class C-2, beta transmission gauges

3.6.3.2) Dispersed Source

3.6.3.2.1) Incineration

Based on the model of Buckley, the intake by the maximum individual near an incinerator which receives 5 devices per year of this type would be 7.0×10^{-8} mCi (2.6×10^{-6} MBq). Dose estimates for the maximum and average individual and the population near an incinerator are listed in table 56.

3.6.3.2.2) Metals Recycling

As stated in section 2.3.2.2, a general result for Co-60 gauges is the only result afforded by the available data. See that section for a discussion of this result.

3.6.3.2.3) Burial in Landfill

The model of Buckley predicts that the intake at the withdrawal point near a landfill which receives one of these devices would be 0.147 mCi (5.4 MBq). Dose estimates for this amount of activity are listed in table 57.

TABLE 54. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION

OF 6 mCi (2.2×10^8 Bq) OF Sr-90

Organ	Dose Equivalent (rem)			
	Ingestion		Inhalation	
	$f_1=0.3$	$f_1=0.01$	Class D	Class Y
Red Marrow	4.2×10^3	1.4×10^2	7.3×10^3	-
Bone Surfaces	9.3×10^3	3.1×10^2	1.6×10^4	-
Upper Large Intestine	-	1.4×10^2	-	-
Lower Large Intestine	-	5.8×10^2	-	-
Lungs	-	-	-	6.4×10^4

TABLE 55. COMMITTED DOSE EQUIVALENTS FOR INHALATION OR INGESTION

OF 2×10^{-5} mCi (7.4×10^{-4} MBq) OF Sr-90

Organ	Dose Equivalent (rem)			
	Ingestion		Inhalation	
	$f_1=0.3$	$f_1=0.01$	Class D	Class Y
Red Marrow	1.4×10^{-2}	4.7×10^{-4}	2.4×10^{-2}	-
Bond Surfaces	3.1×10^{-2}	1.0×10^{-3}	5.4×10^{-2}	-
Upper Large Intestine	-	4.5×10^{-4}	-	-
Lower Large Intestine	-	1.9×10^{-3}	-	-
Lungs	-	-	-	2.1×10^{-1}

TABLE 56. COMMITTED DOSE EQUIVALENTS FOR INHALATION

OF Sr-90 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)		Average Individual (rem)		Population (Person-rem)	
	Class D	Class Y	Class D	Class Y	Class D	Class Y
Red Marrow	8.5×10^{-5}	-	4.2×10^{-5}	-	3.1	-
Bone Surfaces	1.8×10^{-4}	-	9.4×10^{-5}	-	6.8	-
Lungs	-	7.4×10^{-4}	-	3.7×10^{-4}	-	27

TABLE 57. COMMITTED DOSE EQUIVALENTS FOR INGESTION OF

0.147 mCi OF Sr-90

Organ	Dose Equivalent (rem)	
	$f_1 =$	$f_1 = 0.01$
Red Marrow	100	3.5
Bone Surfaces	230	7.6
Upper Large Intestine	-	3.3
Lower Large Intestine	-	14

3.7) CLASS D - GAS CHROMATOGRAPHS

3.7.1) Device Description

Gas chromatographs are laboratory analytical instruments containing ionization sources in detector cells or electron capture detectors. Electron capture detectors are used to analyze the chemical composition of gas samples. Instruments may have interchangeable detector cells. Common nuclides and activities are Ni-63 with 20 mCi (740 MBq) and H-3 with 1000 mCi (37 GBq). Approximately 900 of the Ni-63 detectors are sold per year.

These devices are complex, expensive instruments used mainly in laboratories by trained, knowledgeable personnel. Licensees were more aware of the general license program and requirements than with many other devices (NRC 1987). Vendor information about gas chromatographs appears to be more comprehensive, thereby improving knowledge of license regulations. Several manufacturers make the sources almost impossible to access by ordinary means.

3.7.2) Scenario Development

Pathways for gas chromatographs contain a very high probability of remaining in place without control (0.8), as shown in Figure 8. This value was high due to the assumption that unused gas chromatographs are likely to remain in laboratories. Their size and expense combined with the awareness of regulations by licensees make this outcome more likely than any of the others. A smaller probability is assigned to the devices remaining in place without control (0.15) due to the possibility that they may be stored for future use. Other pathways and final conditions have probabilities which are insignificant. Some gas chromatographs may be discarded or sold to salvage dealers but these probabilities are minimal given the characteristics of the devices and regulation awareness of the licensees.

3.7.3) Dose Assessment

3.7.3.1) 'Intact' Source

External dose - maximum individual

Neither H-3 nor Ni-63 has any significant photon emissions; their beta energies are too low to penetrate the dead layer of the skin. Therefore, these nuclides pose no threat as external sources.

Internal dose - maximum individual

These devices are designed so that the sources are extremely difficult to reach or remove. It would take a significant effort to expose any of the source activity. As with other devices in which the sources are not readily accessible, a fraction of 0.3 was arbitrarily assigned as the maximum amount of the source activity which might be inadvertently taken into the body. Intake of 300 mCi (11,000 MBq) of H-3 would result in a dose equivalent of 19 rem (0.19 Sv) to the soft

CLASS D GAS CHROMATOGRAPHS

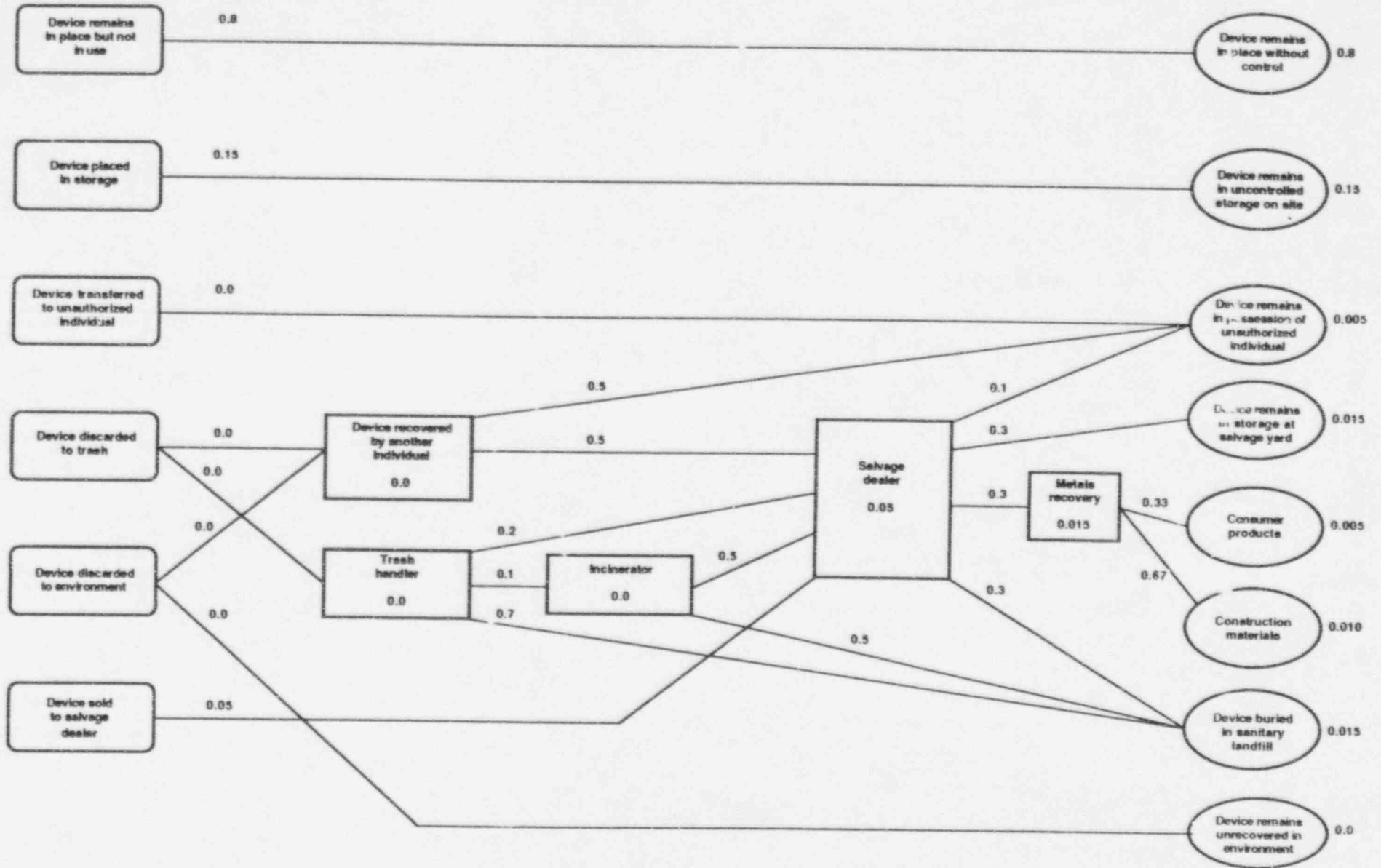


Figure 8 Scenario development for device class D, gas chromatographs

tissues of the whole body.

ICRP 30 assigns oxides, hydroxides, and carbides of nickel to inhalation class W and all other compounds to inhalation class D, with all compounds having an f_1 of 0.05. Table 58 lists dose equivalent commitments from ingestion or inhalation of 6 mCi (222 MBq) of Ni-63.

Internal dose - realistic case

Based on the study by Wilmot, values of 10^{-4} were assigned to both H-3 and Ni-63 as intake fractions for the realistic case. For H-3, this would result in a dose equivalent of 6.3×10^{-3} rem (6.3×10^{-2} mSv) to the soft tissues of the whole body. Table 59 lists dose estimates for intake of 2×10^{-3} mCi (7.4×10^4 Bq) of Ni-63.

3.7.3.2) Dispersed Source

3.7.3.2.1) Incineration

The model of Buckley predicts an intake by the maximum individual near an incinerator site of 7.0×10^{-5} mCi (2.6×10^{-3} MBq) of H-3 and 7.0×10^{-8} mCi (2.6×10^{-6} MBq) of Ni-63. This would result in dose equivalents of 4.4×10^{-6} rem (4.4×10^{-8} Sv), 2.2×10^{-6} rem (2.2×10^{-8} Sv), and 0.16 person-rem (0.0016 person-Sv) to the maximum individual, average individual, and population, respectively from H-3. Estimates for Ni-63 are shown in table 60.

3.7.3.2.2) Metals Recycling

As stated in section 2.3.2.2, a general result for Co-60 gauges is the only result afforded by the available data. See that section for a discussion of this result.

3.7.3.2.3) Burial in Landfill

Based on the model of Buckley, the intake at the withdrawal point near a landfill receiving these sources would be 3.5 mCi (130 MBq) of H-3, and no Ni-63 will reach the withdrawal point. The intake of this amount of H-3 would result in a dose equivalent of 0.22 rem (0.0022 Sv).

TABLE 58. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION

OF 6 mCi ($2.2 \times 10^8 \text{ Bq}$) OF Ni-63

Organ	Ingestion	Dose Equivalent (rem)	
		Inhalation Class D	Class W
Gonads	1.9×10^0	1.8×10^1	5.6×10^0
Breast	1.9×10^0	1.8×10^1	5.6×10^0
Red Marrow	1.9×10^0	1.8×10^1	-
Lungs	1.9×10^0	1.9×10^1	6.9×10^1
Stomach Wall	2.2×10^0	1.8×10^1	-
Small Intestine	2.9×10^0	1.8×10^1	-
Upper Large Intestine	8.0×10^0	1.9×10^1	-
Lower Large Intestine	2.0×10^1	2.1×10^1	1.5×10^1
Kidneys	-	1.3×10^1	-
Thyroid	-	1.8×10^1	-
Bone Surfaces	-	1.8×10^1	-

TABLE 59. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION

OF $2 \times 10^{-3} \text{ mCi}$ ($7.4 \times 10^4 \text{ Bq}$) OF Ni-63

Organ	Ingestion	Dose Equivalent (rem)	
		Inhalation Class D	Class W
Gonads	6.3×10^{-4}	6.1×10^{-3}	1.8×10^{-3}
Breast	6.3×10^{-4}	6.1×10^{-3}	1.8×10^{-3}
Red Marrow	6.3×10^{-4}	6.1×10^{-3}	-
Lungs	6.3×10^{-4}	6.4×10^{-3}	2.3×10^{-2}
Stomach Wall	7.4×10^{-4}	6.1×10^{-3}	-
Small Intestine	9.6×10^{-4}	6.1×10^{-3}	-
Upper Large Intestine	2.7×10^{-3}	6.4×10^{-3}	-
Lower Large Intestine	6.8×10^{-3}	7.0×10^{-3}	5.0×10^{-3}
Kidneys	-	6.1×10^{-3}	-
Thyroid	-	6.1×10^{-3}	-
Bone Surfaces	-	6.1×10^{-3}	-

TABLE 60. COMMITTED DOSE EQUIVALENTS FOR INHALATION

OF Ni-63 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)		Average Individual (rem)		Population (Person-rem)	
	Class D	Class W	Class D	Class W	Class D	Class W
Gonads	2.1×10^{-7}	6.4×10^{-8}	1.1×10^{-7}	3.2×10^{-8}	7.7×10^{-3}	2.3×10^{-3}
Breast	2.1×10^{-7}	6.4×10^{-8}	1.1×10^{-7}	3.2×10^{-8}	7.7×10^{-3}	2.3×10^{-3}
Red Marrow	2.1×10^{-7}	-	1.1×10^{-7}	-	7.7×10^{-3}	-
Lungs	2.2×10^{-7}	8.0×10^{-7}	1.1×10^{-7}	4.0×10^{-7}	8.2×10^{-3}	2.9×10^{-2}
Thyroid	2.1×10^{-7}	-	1.1×10^{-7}	-	7.7×10^{-3}	-
Bone						
Surfaces	2.1×10^{-7}	-	1.1×10^{-7}	-	7.7×10^{-3}	-
Stomach						
Wall	2.1×10^{-7}	-	1.1×10^{-7}	-	7.8×10^{-3}	-
Small						
Intestine	2.1×10^{-7}	-	1.1×10^{-7}	-	7.8×10^{-3}	-
Upper Large						
Intestine	2.2×10^{-7}	-	1.1×10^{-7}	-	8.2×10^{-3}	-
Lower Large						
Intestine	2.4×10^{-7}	1.7×10^{-7}	1.2×10^{-7}	8.6×10^{-8}	8.9×10^{-3}	6.3×10^{-3}
Kidneys	2.1×10^{-7}	-	1.1×10^{-7}	-	7.7×10^{-3}	-

3.8) CLASS E-1 - HIGH TOXICITY X-RAY FLUORESCENCE ANALYZERS

3.8.1) Device Description

X-Ray fluorescence analyzers (XRF analyzers) with high toxicity radionuclides are used in laboratories, on process lines, or in field use to determine the elemental composition of samples. Radioactive sources emit soft X-rays which excite atoms in the material of interest, which in turn emit other, characteristic X-rays. Nuclides common to these instruments include Am-241 with activity of 30 mCi (1100 MBq) and Cm-244 with activity of 100 mCi (3700 MBq). Approximately 90 of these analyzers are distributed per year.

In accountability, this device is somewhat similar to the gas chromatograph. X-ray fluorescence analyzers are generally large, expensive, complex, and are used by trained individuals. These factors are responsible for high accountability. Four of six licensees surveyed by the NRC were aware of license regulations, kept records, and performed leak tests.

3.8.2) Scenario Development

Pathways for the XRF analyzers are similar to the gas chromatograph pathways. Because they are complex, expensive, and used by trained individuals, it is more likely that they would remain in place if unused (0.7). These devices could be left in laboratories after they close or in plants no longer in operation. XRF analyzers may also be placed in storage and forgotten (0.2). Transfer of these devices may occur due to transfer of equipment among laboratories (0.05). Because they can be portable and are used in the field, it is presumed that some of the instruments may be discarded to the environment (0.05).

Final conditions reflect some of the same probabilities mentioned in the previous paragraph. The highest probabilities occur for the device remaining in place without control (0.7) and remaining in storage (0.2). The highest category after these is for the device remaining in possession of an unauthorized individual (0.08). This results from the assumed probability for transfer directly to an unauthorized individual (0.05) and from transfer of the device after it is discarded (0.03). Other final probabilities are very small as noted in Figure 9.

3.8.3) Dose Assessment

3.8.3.1) 'Intact' Source

External dose - maximum individual

Table 6 shows dose equivalents for exposure to point sources at the activity levels stated in section 3.8.1 for Am-241 and Cm-244 for 20 weeks at 100 cm. The table also shows the dose equivalent to the skin from contact to the Am-241 source for three hours (Cm-244 has no beta emissions with energy sufficient to penetrate the dead layer).

April 10, 1987

CLASS E-1 HIGH TOXICITY XRF ANALYZERS

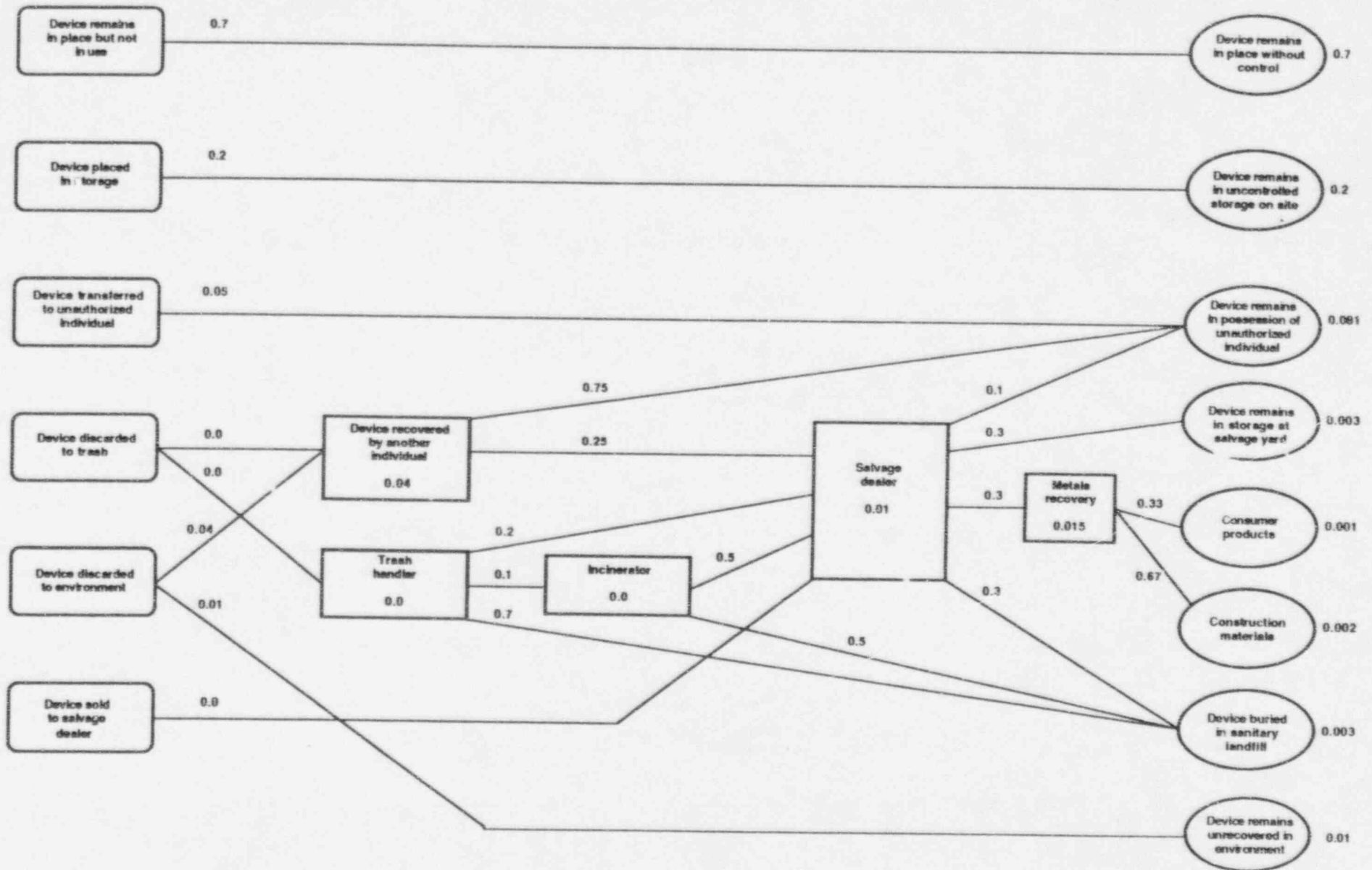


Figure 9 Scenario development for device class E-1, high toxicity XRF analyzers

External dose - realistic case

Values quoted in the NRC registry indicate that external exposure rates would not exceed 10 mR/hr (2.6 uC/kg-hr) to either the whole body or skin. These values were recorded very near to the devices, with the shutter open, and would fall off quickly with distance. The values listed for the maximum dose equivalents could produce transitory effects to the skin (contact with nonencapsulated source), but due to the nature of the device enclosures, these sources would not be expected to be a significant external hazard.

Internal dose - maximum individual

Both Am-241 and Cm-244 represent extremely high internal hazards because of their alpha emissions and, in the case of Cm-244, spontaneous fission events. ICRP 30 assigns all compounds of both of these elements to class W with an f_1 of 0.0005. As with other devices in which the sources are not readily accessible, a fraction of 0.3 was arbitrarily assigned as the maximum amount of the source activity which might be inadvertently taken in. Tables 61 and 62 show the committed dose equivalent predicted by the ICRP 30 models for ingestion of 30 percent of the activity associated with these sources. As with the Am-241 gamma gauges, the dose equivalents for intake of even small amounts of the activity are significant, and any tampering with the sources which results in an intake could have serious consequences. But unauthorized individuals are not likely to have access to these devices because they are usually effectively controlled.

Internal dose - realistic case

Based on the values cited in the paper by Wilmot, intake fractions of 10^{-6} were assigned to Am-241 and Cm-244. Tables 63 and 64 show dose estimates for intake of these amounts of activity.

3.8.3.2) Dispersed Source

3.8.3.2.1) Incineration

The model of Buckley predicts an intake of 1×10^{-7} mCi (3.7×10^{-6} MBq) of Am-241 and 3.5×10^{-7} mCi (1.3×10^{-5} MBq) of Cm-244. Tables 65 and 66 list dose equivalents corresponding to intake of these amounts of activity.

3.8.3.2.2) Metals Recycling

As stated in section 2.3.2.2, a general result for Co-60 gauges is the only result afforded by the available data. See that section for a discussion of this result.

3.8.3.2.3) Burial in Landfill

The model of Buckley predicts that no Am-241 will reach the withdrawal point. A value of K is not listed for Cm-244. It seems

unreasonable to assign a value of 1 because of the nature of the radionuclide. The value for Am-241 was assigned; the model again predicts that none of the activity will reach the withdrawal point.

TABLE 61. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION OF

9 mCi (3.3×10^8 Bq) OF Am-241

Organ	<u>Dose Equivalent (rem)</u>	
	Ingestion	Inhalation
Gonads	4.7×10^3	1.1×10^6
Red Marrow	2.8×10^4	6.7×10^6
Bone Surfaces	3.7×10^5	8.3×10^7
Liver	7.7×10^4	1.8×10^7

TABLE 62. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION OF

30 mCi (1.1×10^9 Bq) OF Cm-244

Organ	<u>Dose Equivalent (rem)</u>	
	Ingestion	Inhalation
Gonads	7.3×10^3	1.8×10^6
Red Marrow	4.9×10^4	1.1×10^7
Bone Surfaces	6.0×10^5	1.4×10^8
Liver	1.4×10^5	3.3×10^7

TABLE 63. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION

OF 3×10^{-5} mCi (1100 Bq) of Am-241

Organ	<u>Dose Equivalent (rem)</u>	
	Ingestion	Inhalation
Gonads	0.0016	3.6
Red Marrow	0.093	22
Bone Surfaces	1.2	280
Liver	0.26	61

TABLE 64. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION
OF 1×10^{-4} mCi (3700 Bq) of Cm-244

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Gonads	0.024	5.9
Red Marrow	0.16	37
Bone Surfaces	2.0	480
Liver	0.48	110

TABLE 65. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Am-241 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)	Average Individual (rem)	Population (Person-rem)
Gonads	1.2×10^{-2}	6.2×10^{-3}	450
Red Marrow	7.7×10^{-2}	3.9×10^{-2}	2800
Bone Surfaces	9.6×10^{-1}	4.8×10^{-1}	35000
Liver	2.1×10^{-1}	1.1×10^{-1}	7700

TABLE 66. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Cm-244 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)	Average Individual (rem)	Population (Person-rem)
Gonads	2.1×10^{-2}	1.0×10^{-2}	750
Red Marrow	1.3×10^{-1}	6.4×10^{-2}	4700
Bone Surfaces	1.7×10^0	8.4×10^{-1}	61000
Liver	3.9×10^{-1}	1.9×10^{-1}	14000

3.9) CLASS E-2 - MODERATE TOXICITY X-RAY FLUORESCENCE ANALYZERS

3.9.1) Device Description

X-ray fluorescence analyzers (XRF analyzers) with moderate toxicity nuclides are similar to the E-1 class of XRF analyzers except that they employ different radionuclides. Nuclides included here are Cd-109 with activity of 20 mCi (740 MBq) and Fe-55 with activity of 100 mCi (3700 MBq). These XRF analyzers are used in laboratories, process lines, and in the field. Approximately 90 of these are distributed each year. The physical characteristics and accountability of these devices are similar to the E-1 class of XRF analyzers.

3.9.2) Scenario Development

Because of their similarity to the other XRF analyzers, the same probabilities were assigned to this class as to class E-1, and the resultant probabilities for the final device status are identical (Figure 10).

3.9.3) Dose Assessment

3.9.3.1) 'Intact' source

External dose - maximum individual

Table 6 shows dose equivalents for exposure to point sources of Cd-109 at the activity levels stated in section 3.9.1 for 20 weeks at 100 cm. A specific gamma constant for Fe-55 could not be located, so one was calculated, including all photons capable of penetrating the skin dead layer (Fe-55 has several very weak L-shell X-rays of about 0.6 keV as well as K-shell X-rays of about 6 keV). Table 6 thus also lists an estimated dose equivalent for exposure to a 100 mCi Fe-55 source at 100 cm for 20 weeks. Neither nuclide has beta particles with energy sufficient to penetrate the skin dead layer.

External dose - realistic case

Values quoted in the NRC registry for dose equivalent rates near these devices were all below 5 mrem/hr (0.005 mSv/hr) with the shutter closed. With the shutter open, however, the values were as high as 26 mrem/hr (0.26 mSv/hr) whole body and 163 mrem/hr (1.63 mSv/hr) to the skin for Cd-109 and 55 mrem/hr (0.55 mSv/hr) whole body and 5200 mrem/hr (52 mSv/hr) to the skin for Fe-55. If an individual were to work with the device often, opening and closing the shutter, he could exceed the 500 mrem/yr (0.005 Sv/yr) limit for the general public for an exposure time of 20 hours near the Cd-109 source or 10 hours near the Fe-55 source. Skin dose equivalents under these situations are predicted to be 3.3 rem (0.033 Sv) for Cd-109 and 52 rem (0.52 Sv) for Fe-55.

CLASS E-2 MODERATE TOXICITY XRF ANALYZERS

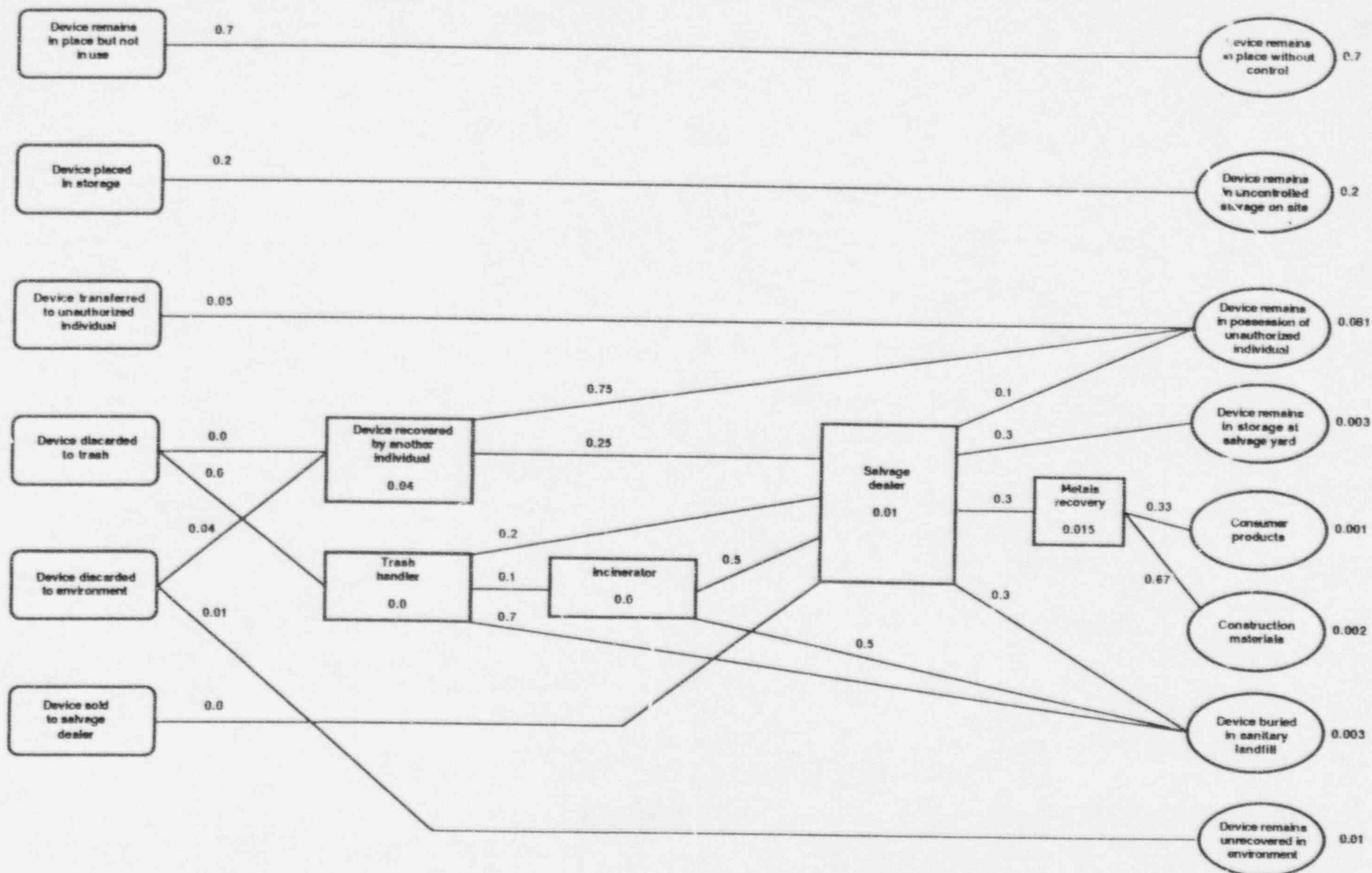


Figure 10 Scenario development for device class E-2, moderate toxicity XRF analyzers

Internal dose - maximum individual

For internal exposures, ICRP 30 assigns oxides and hydroxides of Cd-109 to class Y, sulphides, halides, and nitrates to class W, and all others to class D. All compounds are assigned an f_1 of 0.05. For Fe-55, ICRP 30 assigns oxides, hydroxides, and halides to class W, and all others to class D. All compounds are assigned an f_1 of 0.1. As with other devices in which the sources are not readily accessible, a fraction of 0.3 was arbitrarily assigned as the maximum amount of the source activity which might be inadvertently taken into the body. Tables 67 and 68 list dose estimates for intake of this fraction of the maximum source activity listed in table 1.

Internal dose - realistic case

Values of 10^{-6} were assigned to Cd-109 and 10^{-4} to Fe-55 for intake of material in the realistic case. Tables 69 and 70 list dose estimates corresponding to these levels of activity.

3.9.3.2) Dispersed source

3.9.3.2.1) Incineration

The model of Buckley predicts an intake by the maximum individual near an incinerator which receives 5 devices of this type per year of 7.0×10^{-8} mCi (2.6×10^{-6} MBq) of Cd-109 and 3.5×10^{-7} mCi (1.3×10^{-5} MBq) of Fe-55. Tables 71 and 72 list dose estimates for the maximum and average individual and population near an incinerator.

3.9.3.2.2) Metals Recycling

As stated in section 2.3.2.2, a general result for Co-60 gauges is the only result afforded by the available data. See that section for a discussion of this result.

3.9.3.2.3) Burial in Landfill

The model of Buckley predicts an intake of 2.8×10^{-4} mCi (1.0×10^{-2} MBq) of Cd-109 and 0.027 mCi (1 MBq) of Fe-55. Tables 73 and 74 list dose estimates for ingestion of this amount of activity.

TABLE 67. COMMITTED DOSE EQUIVALENTS FROM INGESTION AND INHALATION
OF 6 mCi (2.2×10^8 Bq) OF Cd-109

Organ	Ingestion	Dose Equivalent (rem)		
		Class D	Inhalation	Class Y
			Class W	
Kidneys	9.1×10^2	8.7×10^3	2.4×10^3	7.5×10^2
Liver	1.6×10^2	1.6×10^3	4.7×10^2	-
Lungs	-	-	3.3×10^2	1.7×10^3
Lower Large Intestine	1.0×10^2	-	-	-

TABLE 68. COMMITTED DOSE EQUIVALENTS FROM INGESTION AND INHALATION
OF 30 mCi (1.1×10^9 Bq) OF Fe-55

Organ	Ingestion	Dose Equivalent (rem)	
		Class D	Inhalation
			Class W
Gonads	1.1×10^1	5.7×10^1	2.0×10^1
Breast	1.1×10^1	5.7×10^1	1.9×10^1
Red Marrow	1.1×10^1	5.7×10^1	2.0×10^1
Lungs	1.1×10^1	5.7×10^1	1.2×10^2
Small Intestine	1.3×10^1	-	-
Upper Large Intestine	1.9×10^1	5.7×10^1	-
Lower Large Intestine	3.3×10^1	6.0×10^1	3.1×10^1
Liver	3.7×10^1	1.9×10^2	6.4×10^1
Spleen	6.2×10^1	3.1×10^2	1.1×10^2
Remainder of Body	-	5.7×10^1	-

TABLE 69. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION
OF 2×10^{-5} mCi (740 Bq) OF Cd-109

Organ	Ingestion	Dose Equivalent (rem)		
		Class D	Inhalation	
			Class W	Class Y
Kidneys	3.0×10^{-3}	2.9×10^{-2}	8.1×10^{-3}	2.5×10^{-3}
Liver	5.5×10^{-4}	5.2×10^{-3}	1.6×10^{-3}	-
Lungs	-	-	1.1×10^{-3}	5.8×10^{-3}
Lower Large Intestine	3.4×10^{-4}	-	-	-

TABLE 70. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION
OF 0.01 mCi (3.7×10^5 Bq) of Fe-55

Organ	Ingestion	Dose Equivalent (rem)	
		Class D	Inhalation
			Class W
Gonads	4.1×10^{-3}	1.9×10^{-2}	6.7×10^{-3}
Breast	3.7×10^{-3}	1.9×10^{-2}	6.7×10^{-3}
Red Marrow	4.1×10^{-3}	1.9×10^{-2}	6.7×10^{-3}
Lungs	3.7×10^{-3}	1.9×10^{-2}	6.7×10^{-3}
Small Intestine	4.4×10^{-3}	-	-
Upper Large Intestine	6.3×10^{-3}	1.9×10^{-2}	-
Lower Large Intestine	1.1×10^{-3}	2.0×10^{-2}	1.0×10^{-2}
Liver	1.3×10^{-2}	6.3×10^{-2}	2.1×10^{-2}
Spleen	2.1×10^{-2}	1.0×10^{-1}	3.5×10^{-2}
Remainder of Body	-	1.9×10^{-2}	-

TABLE 71. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Cd-109 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)		Average Individual (rem)		Population (Person-rem)	
	Class D		Class D		Class D	
Kidneys	1.0×10^{-4}		5.0×10^{-5}		3.7×10^0	
Liver	1.8×10^{-5}		9.1×10^{-6}		6.7×10^{-1}	

Organ	Maximum Individual (rem)		Average Individual (rem)		Population (Person-rem)	
	Class W	Class Y	Class W	Class Y	Class W	Class Y
Lungs	3.9×10^{-6}	2.0×10^{-5}	1.9×10^{-6}	1.0×10^{-5}	1.4×10^{-1}	7.3×10^{-1}
Kidneys	2.8×10^{-5}	8.7×10^{-6}	1.4×10^{-5}	4.4×10^{-6}	1.0×10^0	3.2×10^{-1}
Liver	5.4×10^{-6}	-	2.7×10^{-6}	-	2.0×10^{-1}	-

TABLE 72. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Fe-55 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)		Average Individual (rem)		Population (Person-rem)	
	Class D	Class W	Class D	Class W	Class D	Class W
Gonads	6.7×10^{-7}	2.3×10^{-7}	3.3×10^{-7}	1.2×10^{-7}	2.4×10^{-2}	8.4×10^{-3}
Breast	6.6×10^{-7}	2.2×10^{-7}	3.3×10^{-7}	1.1×10^{-7}	2.4×10^{-2}	8.0×10^{-3}
Red Marrow	6.7×10^{-7}	2.3×10^{-7}	3.3×10^{-7}	1.2×10^{-7}	2.4×10^{-2}	8.4×10^{-3}
Lungs	6.7×10^{-7}	1.4×10^{-6}	3.3×10^{-7}	7.1×10^{-7}	2.4×10^{-2}	5.2×10^{-2}
Upper Large Intestine	6.7×10^{-7}	-	3.3×10^{-7}	-	2.4×10^{-2}	-
Lower Large Intestine	6.9×10^{-7}	3.6×10^{-7}	3.5×10^{-7}	1.8×10^{-7}	2.5×10^{-2}	1.3×10^{-2}
Liver	2.2×10^{-6}	7.5×10^{-7}	1.1×10^{-6}	3.7×10^{-7}	8.0×10^{-2}	2.7×10^{-2}
Spleen	3.6×10^{-6}	1.2×10^{-6}	1.8×10^{-6}	6.1×10^{-7}	1.3×10^{-1}	4.4×10^{-2}
Remainder of Body	6.7×10^{-7}	-	3.3×10^{-7}	-	2.4×10^{-2}	-

TABLE 73. COMMITTED DOSE EQUIVALENTS FOR INGESTION OF
 2.8×10^{-4} mCi OF Cd-109

Organ	Dose Equivalent (rem)
Lower Large Intestine	4.6×10^{-3}
Kidneys	4.3×10^{-2}
Liver	7.7×10^{-3}

TABLE 74. COMMITTED DOSE EQUIVALENTS FOR INGESTION OF
0.0274 mCi OF Fe-55

Organ	Dose Equivalent (rem)
Gonads	1.1×10^{-2}
Breast	1.0×10^{-2}
Red Marrow	1.1×10^{-2}
Lungs	1.0×10^{-2}
Small Intestine	1.2×10^{-2}
Upper Large Intestine	1.7×10^{-2}
Lower Large Intestine	3.0×10^{-2}
Liver	3.4×10^{-2}
Spleen	5.7×10^{-2}

3.10) CLASS F - CALIBRATION OR REFERENCE SOURCES - Cs-137, Co-60,
Ra-226, Sr-90

3.10.1) Device Description

This class includes sources often supplied by the detector manufacturer and are used to check instrument performance in the field or as calibration/analytical standards. Small in size, these sources are susceptible to loss. Nuclides commonly used as calibration sources with activities include Am-241, 0.01 mCi (0.37 MBq), Ra-226, 0.004 mCi (0.15 MBq), Cs-137, 0.10 mCi (3.7 MBq), Co-60, 0.01 mCi (0.37 MBq), Sr-90, 0.001 mCi (0.037 MBq), and plutonium, 0.005 mCi (0.185 MBq). Covered under this section are such devices regulated under 10CFR31.5. Americium-241, regulated under 10CFR31.8, and Pu-239, regulated under 10CFR70.19, are covered as class I and K, respectively. No estimates are available on the number of these sources distributed per year.

Detector sources are used in laboratories to calibrate instruments. These sources are used by knowledgeable personnel, inventoried periodically, and are usually kept in a locked cabinet. This system allows for a close check on the sources. However, these small sources are also taken out into the field to be used as calibration standards, which increases their susceptibility to loss and theft.

3.10.2) Scenario Development

Pathway probabilities were estimated taking into account the two environments in which these sources are used (Figure 11). The pathway believed to be the most probable is for the device remaining in place but not in use (0.5). This accounts for those sources used in laboratories or stored for field purposes which remain in place even if the lab is shut down, unused, or abandoned. The device also may remain in storage (0.1) without use in the laboratory. Due to the field use of some sources, higher probabilities are assigned to the device being discarded in the trash (0.15 total) or discarded to the environment (0.20 total). A smaller probability was estimated for the transfer of the source to an unauthorized individual (0.05).

Final condition results remain the same for the device being in place or in storage. Other conditions change as they are affected by the various pathways. As a result of these detector sources being discarded to the trash or the environment, some devices will be buried in a sanitary landfill (0.12). A higher probability now occurs when the device remains in the possession of an unauthorized individual (0.12) as several pathways affect this final result. These detector sources also may remain unrecovered in the environment (0.14) due to the field use of these sources. Small probabilities appear in the remaining conditions (storage at the salvage yard and consumer or construction products).

CLASS F CALIBRATION OR REFERENCE SOURCES

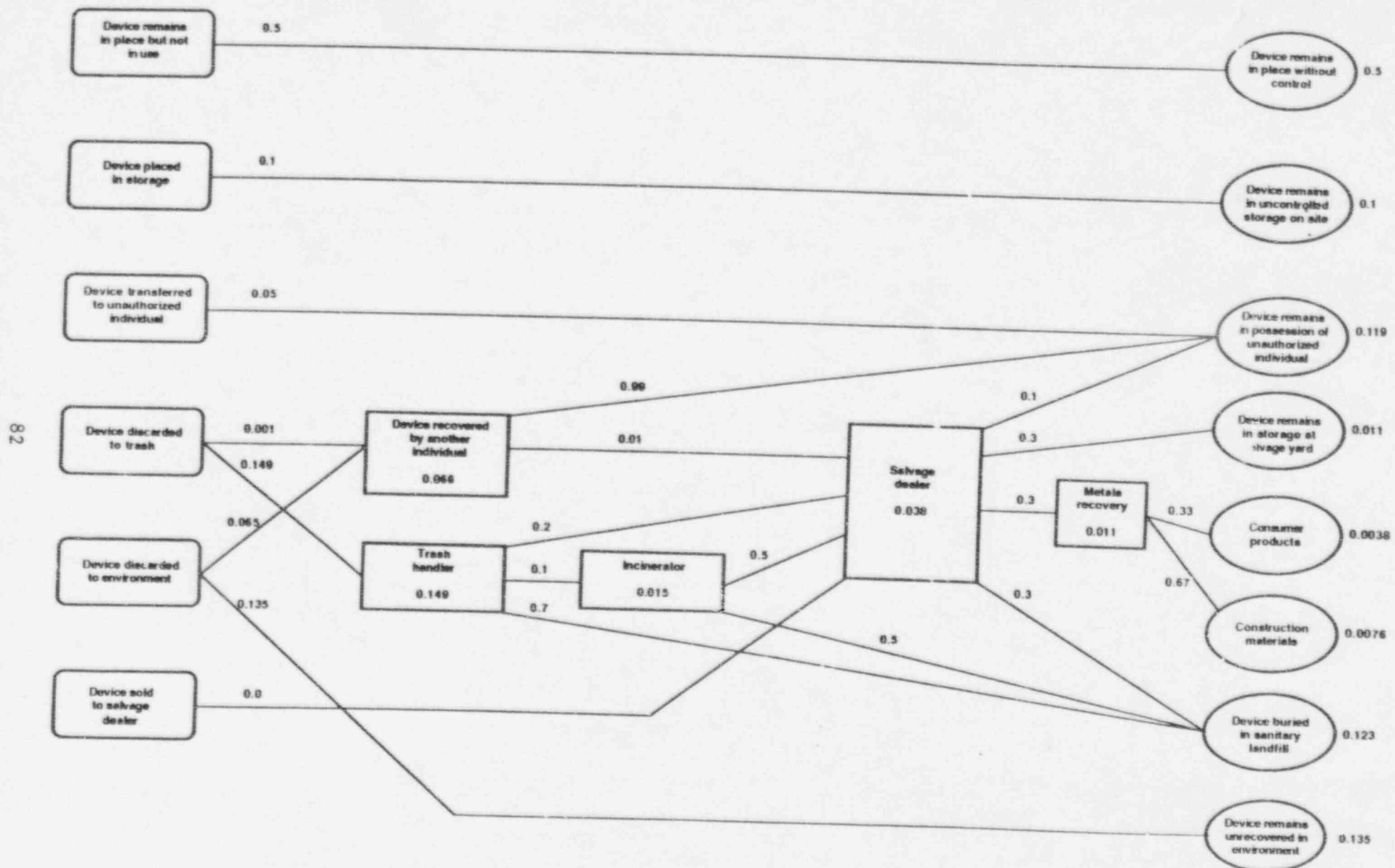


Figure 11 Scenario development for device class F, calibration or reference sources

3.10.3) Dose Assessment

3.10.3.1) 'Intact' Source

External dose - maximum individual

Table 6 shows dose equivalents for exposure to point sources at the activity levels stated in section 3.10.1 for Ra-226, Cs-137, Co-60, and Sr-90 for 20 weeks at 100 cm or in contact for three hours. Because these sources are small and designed to be portable, longer contact times could be envisioned if the source were to be kept in a pocket. In these cases, some of the beta energy would be attenuated by the person's clothing before reaching the skin. Dose equivalents received from gamma emissions could be considerably higher than those listed, as the distance between the source and the person would be less, and the exposure rate changes in proportion to the square of the distance.

External dose - realistic case

In most cases, individuals who find a check source would only be exposed to them for a short period of time, and would rarely be exposed to the beta emissions, because the sources are usually enclosed. The sources are labeled as being radioactive, so most people would turn the source over to a local authority or discard it. Dose equivalents for a one hour contact time to an encapsulated source would be at most 5 rem (0.05 Sv), for Cs-137.

Internal dose - maximum individual

Because of the size of the sources, it is not inconceivable that someone might inadvertently ingest the entire source. Indeed such a case is discussed in section 3.14.3.1. However, the amount of material which might be freed from the source and available for uptake into body organs would not usually be 100% of that ingested. For this reason, a maximum fraction of 0.3 was again assigned as the maximum amount of free activity which might be ingested. For inhalation, the assumption used in previous sections that 0.3 would be the maximum that might be inhaled was again used, as it would also take a significant effort to dislodge the source activity. Tables 75-78 list dose estimates for intake of 30% of the maximum source activity, as listed in table 1.

Internal dose - realistic case

On the basis of the recommendations of Wilmot, fractions of 10^{-6} were assigned to Ra-226, Cs-137, and Sr-90 and a fraction of 10^{-4} was assigned to Co-60. Tables 79-82 list dose estimates for these amounts of ingestion or inhalation.

3.10.3.2) Dispersed Source

3.10.3.2.1) Incineration

The model of Buckley predicts an intake by the maximum individual of 1.4×10^{-11} mCi (5.2×10^{-10} MBq) of Ra-226, 3.6×10^{-10} mCi ($1.3 \times$

10^{-8} MBq) of Cs-137, 1.9×10^{-11} mCi (7.0×10^{-10} MBq) of Co-60, and 3.5×10^{-12} mCi (1.3×10^{-10} MBq) of Sr-90. Tables 83-86 list the dose estimates for the maximum and average individual and the population near an incinerator which receives 5 of these sources per year.

3.10.3.2.2) Metals Recycling

As stated in section 2.3. .2, a general result for Co-60 gauges is the only result afforded by the available data. See that section for a discussion of this result.

3.10.3.2.3) Burial in Landfill

The model of Buckley for landfill leaching predicts that 7.6×10^{-5} mCi (2.8×10^{-3} MBq) of Ra-226 and 7.2×10^{-6} mCi (2.7×10^{-4} MBq) of Sr-90 will reach the withdrawal point, and that neither of the other radionuclides will reach the withdrawal point before removal by radioactive decay. Tables 87 and 88 list the dose equivalents corresponding to ingestion of these levels of activity of Ra-226 and Sr-90.

TABLE 75. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION
OF 0.0012 mCi (4.4×10^4 Bq) OF Ra-226

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Gonads	4.1×10^{-1}	-
Red Marrow	2.7×10^0	-
Bone Surfaces	3.0×10^1	3.4×10^1
Lungs	-	7.1×10^1

TABLE 76. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION
OF 0.03 mCi (1.11×10^6 Bq) OF Cs-137

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Gonads	1.6×10^0	9.8×10^{-1}
Breast	1.3×10^0	8.7×10^{-1}
Red Marrow	1.4×10^0	9.2×10^{-1}
Lungs	1.4×10^0	9.8×10^{-1}
Thyroid	1.4×10^0	8.8×10^{-1}
Bone Surfaces	1.4×10^0	8.8×10^{-1}
Small Intestine	1.6×10^0	1.0×10^0
Upper Large Intestine	1.6×10^0	1.0×10^0
Lower Large Intestine	1.6×10^0	1.0×10^0
Remainder of Body	1.7×10^0	1.1×10^0

TABLE 77. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION
OF 0.003 mCi (1.11×10^5 Bq) OF Co-60

Organ	Dose Equivalent (rem)			
	Ingestion $f_1=0.05$		Inhalation Class W Class Y	
Gonads	3.6×10^{-2}	8.0×10^{-2}	4.4×10^{-2}	3.8×10^0
Breast	1.2×10^{-2}	5.7×10^{-2}	4.7×10^{-2}	-
Red Marrow	1.4×10^{-2}	6.1×10^{-2}	4.7×10^{-2}	-
Lungs	9.6×10^{-3}	5.6×10^{-2}	4.0×10^{-1}	-
Small Intestine	4.0×10^{-2}	9.1×10^{-2}	-	-
Upper Large Intestine	6.3×10^{-2}	1.1×10^{-1}	-	-
Lower Large Intestine	1.2×10^{-1}	1.6×10^{-1}	9.1×10^{-2}	-
Liver	2.6×10^{-2}	1.4×10^{-1}	1.0×10^{-1}	-
Remainder of Body	2.3×10^{-2}	9.6×10^{-2}	8.9×10^{-2}	-

TABLE 78. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION
OF 0.0003 mCi (1.1×10^4 Bq) OF Sr-90

Organ	Dose Equivalent (rem)			
	Ingestion $f_1=0.3$		Inhalation Class D Class Y	
Red Marrow	2.1×10^{-1}	7.1×10^{-3}	3.7×10^{-1}	-
Bone Surfaces	4.7×10^{-1}	1.6×10^{-2}	8.1×10^{-1}	-
Upper Large Intestine	-	6.8×10^{-3}	-	-
Lower Large Intestine	-	2.9×10^{-2}	-	-
Lungs	-	-	-	3.2×10^0

TABLE 79. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION

OF 4×10^{-9} mCi (0.15 Bq) OF Ra-226

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Gonads	1.4×10^{-6}	-
Red Marrow	8.9×10^{-6}	-
Bone Surfaces	1.0×10^{-4}	1.1×10^{-4}
Lungs	-	2.3×10^{-4}

TABLE 80. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION

OF 10^{-7} mCi (3.7 Bq) OF Cs-137

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Gonads	5.2×10^{-6}	3.3×10^{-6}
Breast	4.4×10^{-6}	2.9×10^{-6}
Red Marrow	4.8×10^{-6}	3.1×10^{-6}
Lungs	4.8×10^{-6}	3.3×10^{-6}
Thyroid	4.8×10^{-6}	2.9×10^{-6}
Bone Surfaces	4.8×10^{-6}	2.9×10^{-6}
Small Intestine	5.2×10^{-6}	3.4×10^{-6}
Upper Large Intestine	5.2×10^{-6}	3.3×10^{-6}
Lower Large Intestine	5.2×10^{-6}	3.4×10^{-6}
Remainder of Body	5.6×10^{-6}	3.5×10^{-6}

TABLE 81. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION
OF 10^{-6} mCi (37 Bq) OF Co-60

Organ	Dose Equivalent (rem)			
	Ingestion		Inhalation	
	$f_1=0.05$	$f_1=0.3$	Class W	Class Y
Gonads	1.2×10^{-5}	2.7×10^{-5}	1.5×10^{-5}	-
Breast	4.1×10^{-6}	1.9×10^{-5}	1.6×10^{-5}	-
Red Marrow	4.8×10^{-6}	2.0×10^{-5}	1.6×10^{-5}	-
Lungs	3.2×10^{-6}	1.8×10^{-5}	1.3×10^{-4}	1.2×10^{-3}
Small Intestine	1.3×10^{-5}	3.0×10^{-5}	-	-
Upper Large Intestine	2.1×10^{-5}	3.6×10^{-5}	-	-
Lower Large Intestine	4.1×10^{-5}	5.2×10^{-5}	3.0×10^{-5}	-
Liver	8.5×10^{-6}	4.8×10^{-5}	3.4×10^{-5}	-
Remainder of Body	7.8×10^{-6}	3.2×10^{-5}	3.0×10^{-5}	-

TABLE 82. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION
OF 1×10^{-9} mCi (0.037 Bq) OF Sr-90

Organ	Dose Equivalent (rem)			
	Ingestion		Inhalation	
	$f_1=0.3$	$f_1=0.01$	Class D	Class Y
Red Marrow	7.0×10^{-7}	2.4×10^{-8}	1.2×10^{-6}	-
Bone Surfaces	1.6×10^{-6}	5.2×10^{-8}	2.7×10^{-6}	-
Upper Large Intestine	-	2.3×10^{-8}	-	-
Lower Large Intestine	-	9.6×10^{-8}	-	-
Lungs	-	-	-	1.1×10^{-5}

TABLE 83. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Ra-226 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)	Average Individual (rem)	Population (Person-rem)
Lungs	8.2×10^{-7}	4.1×10^{-7}	3.0×10^{-2}
Bone Surfaces	3.9×10^{-7}	2.0×10^{-7}	1.4×10^{-2}

TABLE 84. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Cs-137 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)	Average Individual (rem)	Population (Person-rem)
Gonads	1.2×10^{-8}	5.9×10^{-9}	4.3×10^{-4}
Breast	1.1×10^{-8}	5.3×10^{-9}	3.8×10^{-4}
Red Marrow	1.1×10^{-8}	5.6×10^{-9}	4.1×10^{-4}
Lungs	1.2×10^{-8}	5.9×10^{-9}	4.3×10^{-4}
Thyroid	1.1×10^{-8}	5.3×10^{-9}	3.9×10^{-4}
Bone Surfaces	1.1×10^{-8}	5.3×10^{-9}	3.9×10^{-4}
Small Intestine	1.2×10^{-8}	6.1×10^{-9}	4.5×10^{-4}
Upper Large Intestine	1.2×10^{-8}	6.1×10^{-9}	4.4×10^{-4}
Lower Large Intestine	1.2×10^{-8}	6.1×10^{-9}	4.5×10^{-4}
Remainder of Body	1.3×10^{-8}	6.4×10^{-9}	4.7×10^{-4}

TABLE 85. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Co-60 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)		Average Individual (rem)		Population (Person-rem)	
	Class W	Class Y	Class W	Class Y	Class W	Class Y
Gonads	2.8×10^{-10}	-	1.4×10^{-10}	-	1.0×10^{-5}	-
Breast	2.9×10^{-10}	-	1.5×10^{-10}	-	1.1×10^{-5}	-
Red Marrow	2.9×10^{-10}	-	1.5×10^{-10}	-	1.1×10^{-5}	-
Lungs	2.5×10^{-9}	2.4×10^{-8}	1.3×10^{-9}	1.2×10^{-8}	9.2×10^{-5}	8.7×10^{-4}
Lower Large Intestine	5.8×10^{-10}	-	2.9×10^{-10}	-	2.1×10^{-5}	-
Liver	6.5×10^{-10}	-	3.2×10^{-10}	-	2.4×10^{-5}	-
Remainder of Body	5.6×10^{-10}	-	2.8×10^{-10}	-	2.0×10^{-5}	-

TABLE 86. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Sr-90 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)		Average Individual (rem)		Population (Person-rem)	
	Class D	Class Y	Class D	Class Y	Class D	Class Y
Red Marrow	4.2×10^{-9}	-	2.1×10^{-9}	-	1.5×10^{-4}	-
Bone Surfaces	9.4×10^{-9}	-	4.7×10^{-9}	-	3.4×10^{-4}	-
Lungs	-	3.7×10^{-8}	-	1.9×10^{-8}	-	1.4×10^{-3}

TABLE 87. COMMITTED DOSE EQUIVALENTS FOR INGESTION OF

7.6×10^{-5} mCi OF Ra-226

Organ	Dose Equivalent (rem)
Gonads	2.6×10^{-2}
Red Marrow	1.7×10^{-1}
Bone Surfaces	1.9×10^0

TABLE 88. COMMITTED DOSE EQUIVALENTS FOR INGESTION OF

7.2×10^{-6} mCi OF Sr-90

Organ	Dose Equivalent (rem)	
	f = 0.3	f = 0.01
Red Marrow	5.1×10^{-3}	1.7×10^{-4}
Bone Surfaces	1.1×10^{-2}	3.7×10^{-4}
Upper Large Intestine	-	1.6×10^{-4}
Lower Large Intestine	-	6.9×10^{-4}

3.11) CLASS G-1 - SELF-LUMINOUS DEVICES

3.11.1) Device Description

This class of device includes lighted warning signs such as exit signs, emergency light sources, safety markers, and light wands. Lighted gunsights and reference photometric standards with very low activity are also included. The principal nuclides in these devices are H-3 with an activity of 5000 mCi (185 GBq), Kr-85 with activity up to 1700 mCi (62.9 GBq), and C-14 with an activity of 0.10 mCi (3.7 MBq). Approximately 50,000 of these devices are distributed per year.

Of the several devices included in this classification, self-luminous exit signs are most frequently cited and discussed. They will provide the frame of reference for discussion of this category. The light source for these signs consist of phosphor-coated tubes filled with tritium. They are cased in plastic and can be ordered with or without an aluminum frame. As with the self-luminous aircraft signs, manufacturers recommend returning exit signs at the end of their useful life or disposing of them in a licensed radioactive waste site. The useful life may be ten, fifteen, or twenty years depending on the type of sign. One manufacturer stated that records are kept of the end user, serial number of the signs, and date of manufacture. The signs are also labeled according to NRC regulations with the hazardous symbol, UL sign, activity level, and address for return. However, the survey by the NRC found that the placement of labels was inconsistent and that the label could not always be found. Recordkeeping was also not as consistent as the manufacturer stated. Seventy-six percent of licensees surveyed reported that they kept the required records, including record of purchase, installation, and transfer.

The NRC survey found widespread problems with transfer of exit signs. Vendors sell the signs to electrical distributors as general licensees and they resell the signs. The electrical distributors were found to be unaware of regulations (14 of 16 surveyed) and therefore could not make their customers aware of regulations. This resale transfer makes it impossible for vendors and regulatory bodies to keep track of signs (NRC 1987).

3.11.2) Scenario Development

The probabilities assigned to the pathways were based on the labeling of exit signs, their improper transfer by general licensees, recordkeeping by licensees, lack of awareness of regulations, their appeal to the general public, and their apparent dollar value if sold (Figure 12). Initial pathways which show higher probabilities based on these include transferring the device to an unauthorized individual (0.35), remaining in place but not in use (0.20), storage (0.15), and being discarded to trash (0.15). The outcome is also affected for these same reasons with a high probability that the devices will remain in the possession of unauthorized individuals (0.58) because of improper transfer among licensees and from the devices being discarded and recovered or sold to a salvage dealer who resells them. It is assumed they would be attractive to the public and that salvage dealers

CLASS G-1 SELF-LUMINOUS DEVICES

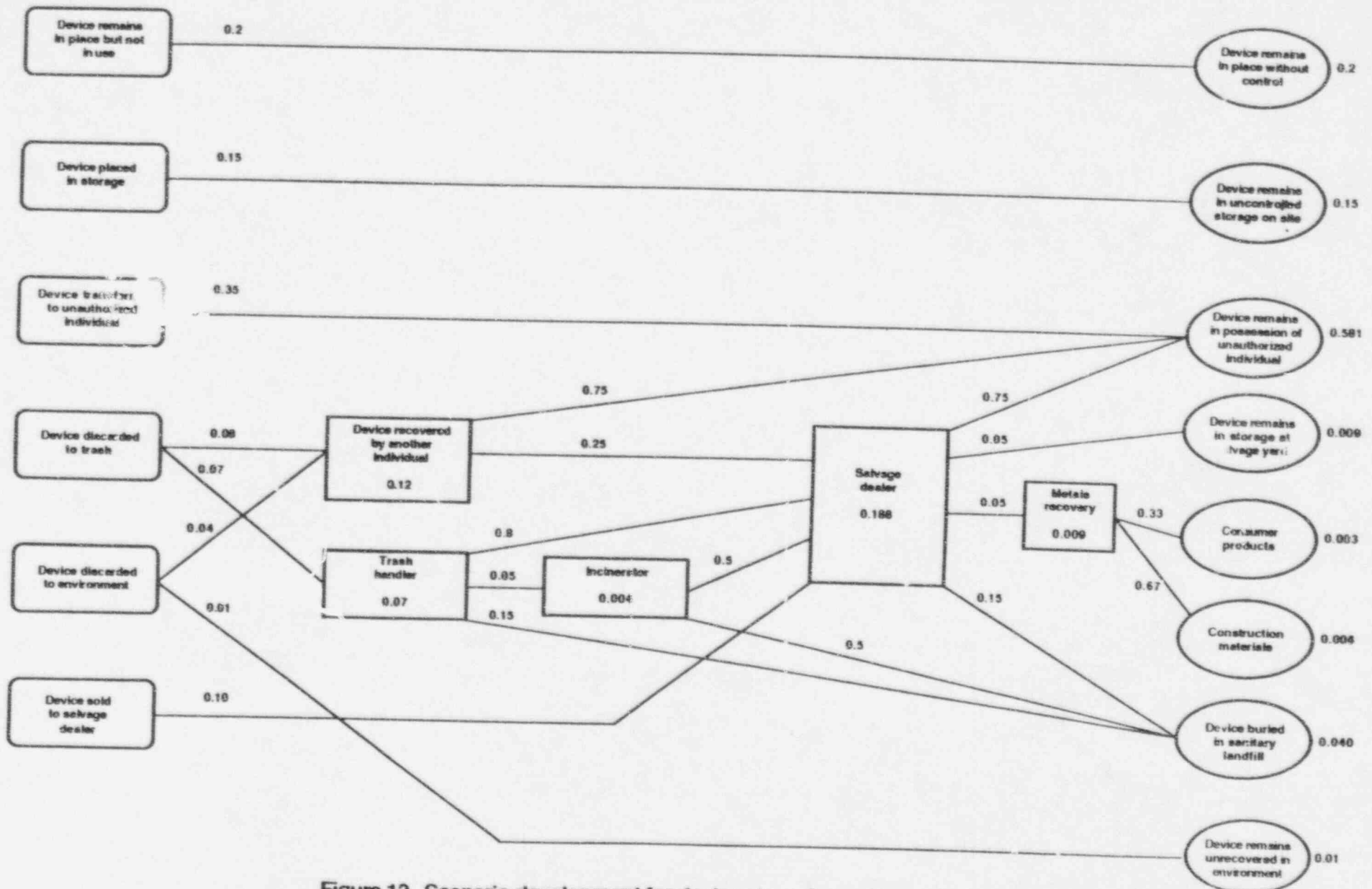


Figure 12 Scenario development for device class G-1, self-luminous devices

would make a greater profit by resale than by metals recycling. Metals recycling also depends on whether the sign is encased in aluminum or consists of plastic only. The other significant probabilities include the device remaining in place without control (0.20) and the device remaining in uncontrolled storage on site (0.15). These outcomes are due to lack of regulation awareness, abandoned buildings, and inconsistent labeling. Other outcomes had lower probabilities.

3.11.3) Dose Assessment

3.11.3.1) 'Intact' Source

External dose - maximum individual

Dose equivalents for exposure to point sources at the activity levels stated in section 3.10.1 for H-3, Kr-85, and C-14 for 20 weeks at 100 cm are shown in table 6 for Kr-85 and for contact for three hours for C-14. H-3 and C-14 do not present external photon hazards, and Kr-85 sources would not be a contact hazard because the Kr-85 would disperse if the activity were released.

External dose - realistic case

As neither of the maximum dose equivalents would produce any observable effects in the individual, estimates for a more realistic situation would also not be of concern for acute effects. The limit for the general public for whole body exposure (500 mrem) might be exceeded if a Kr-85 source were kept for a substantial period of time at 100 cm distance (more than 8 days) and the occupational limit for dose equivalent to the extremities (75 rem/year) could be exceeded if the C-14 source itself were contacted for more than 5 hours, but this is less likely than the case of exposure to Kr-85.

Internal dose - maximum individual

As with other devices in which the source is not normally exposed, fractions of 0.3 of the H-3 and C-14 activity were assumed to be ingested or inhaled in the maximum case. This would result in dose equivalents to the whole body or any organ of 94 rem (0.94 Sv) for H-3 and 0.062 rem (0.62 mSv) for C-14. For Kr-85, it was assumed that if the source were tampered with, 100% of the activity would be released into the smallest size room (100 m³) listed in the ICRP 30 dose estimate tables. Table 89 lists the dose equivalent rates to the organs listed in that report for release of this amount of activity.

Internal dose - realistic case

Based on the values given by Wilmot, intake values of 10^{-4} were assigned to H-3 and C-14 and a value of 10^{-1} was assigned to Kr-85 (although this represents a release, not an intake). This would result in dose equivalents to the whole body or any organ of 0.031 rem (0.31 mSv) for H-3 and 2.1×10^{-5} rem (2.1×10^{-7} Sv) for C-14. Table 90 lists dose equivalents for the 100 m³ room for exposure to 10% of the maximum activity listed in table 1 for Kr-85.

3.11.3.2) Dispersed Source

3.11.3.2.1) Incineration

The model of Buckley predicts an intake of 3.4×10^{-4} mCi (1.3×10^{-2} MBq) of H-3 and 7.0×10^{-10} mCi (2.6×10^{-8} MBq) of C-14 by the maximum individual. For Kr-85, the model predicts a maximum downwind concentration of 1.6×10^{-8} $\mu\text{Ci}/\text{m}^3$ (5.9×10^{-7} MBq/ m^3). Table 91 lists annual dose estimates for exposure to this concentration of Kr-85 for the maximum and average individual and population near an incinerator which receives 5 of these devices per year. The intakes listed for H-3 would result in dose commitments of 2.2×10^{-5} rem (2.2×10^{-7} Sv), 1.1×10^{-5} rem (1.1×10^{-7} Sv), and 0.80 person-rem (0.0080 person-Sv) to the maximum individual, average individual, and population, respectively. The intake listed for C-14 would result in dose commitments of 1.4×10^{-9} rem (1.4×10^{-11} Sv), 7.2×10^{-10} rem (7.2×10^{-12} Sv), and 5.3×10^{-5} person-rem (5.3×10^{-7} person-Sv) to the maximum individual, average individual, and population, respectively.

3.11.3.2.2) Metals Recycling

As stated in section 2.3.2.2, a general result for Co-60 gauges is the only result afforded by the available data. See that section for a discussion of this result.

3.11.3.2.3) Burial in Landfill

The model of Buckley for a landfill predicts 17.5 mCi (648 MBq) of H-3 and 0.0025 mCi (0.0925 MBq) of C-14 will reach the withdrawal point. No groundwater pathway is hypothesized for Kr-85. These intakes correspond to dose equivalents of 1.1 rem (0.011 Sv) and 0.0051 rem (0.051 mSv), respectively, to the whole body or any organ.

TABLE 89. DOSE EQUIVALENT RATES FOR IMMERSION IN 1700 mCi
 (6.3×10^{10}) Bq) OF Kr-85 IN A M^3 ROOM

Organ	Dose Equivalent (rem/hr)
Gonads	7.5×10^{-4}
Breast	6.9×10^{-4}
Red Marrow	8.2×10^{-4}
Lungs	6.0×10^{-4}
Bone Surfaces	8.8×10^{-4}
Stomach Wall	6.1×10^{-4}
Kidneys	5.7×10^{-4}
Liver	5.3×10^{-4}
Spleen	6.3×10^{-4}
Adrenals	5.6×10^{-4}
Skin	2.9×10^0
Lens of the eye	1.3×10^{-3}

TABLE 90. DOSE EQUIVALENT RATES FOR IMMERSION IN 170 mCi
 (6.3×10^9) Bq) OF Kr-85 IN A 100 M^3 ROOM

Organ	Dose Equivalent Rate (rem/hr)
Gonads	7.5×10^{-5}
Breast	6.9×10^{-5}
Red Marrow	8.2×10^{-5}
Lungs	6.0×10^{-5}
Bone Surfaces	8.8×10^{-5}
Stomach wall	6.1×10^{-5}
Kidneys	5.7×10^{-5}
Liver	5.3×10^{-5}
Spleen	6.3×10^{-5}
Adrenals	5.6×10^{-5}
Skin	2.9×10^{-1}
Lens of the eye	1.3×10^{-4}

TABLE 91. ANNUAL DOSE EQUIVALENTS FOR INHALATION

OF Kr-85 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)	Average Individual (rem)	Population (Person-rem)
Gonads	2.7×10^{-7}	1.4×10^{-7}	1.0×10^{-2}
Breast	2.4×10^{-7}	1.2×10^{-7}	8.6×10^{-3}
Red Marrow	3.0×10^{-7}	1.5×10^{-7}	1.1×10^{-2}
Lungs	2.3×10^{-7}	1.1×10^{-7}	8.2×10^{-3}
Bone Surfaces	3.2×10^{-7}	1.6×10^{-7}	1.2×10^{-2}
Stomach Wall	2.3×10^{-7}	1.1×10^{-7}	8.2×10^{-3}
Kidneys	2.0×10^{-7}	1.0×10^{-7}	7.3×10^{-3}
Liver	2.0×10^{-7}	1.0×10^{-7}	7.3×10^{-3}
Spleen	2.4×10^{-7}	1.2×10^{-7}	8.8×10^{-3}
Adrenals	2.1×10^{-7}	1.0×10^{-7}	7.7×10^{-3}
Skin	2.5×10^{-5}	1.2×10^{-5}	9.0×10^{-1}
Lens of the eye	3.1×10^{-7}	1.5×10^{-7}	1.1×10^{-2}

3.12) CLASS G-2 - SELF-LUMINOUS DEVICES IN AIRCRAFT

3.12.1) Device Description

Devices of this type include lighted warning signs which are permanently mounted. The sources most commonly found in these devices are H-3 with activities up to 5000 mCi (185 GBq) and Pm-147 with activities of up to 300 mCi (11 GBq). More than 30,000 of these devices are presently installed in aircraft with six or more in large commercial planes and four to six in the smaller commercial planes (NRC 1987). A potential for internal hazard exists if the device is damaged (NCRP 1977). The life of an aircraft exit sign as stated by one manufacturer is five years.

At least one manufacturer of these devices highly recommends returning the signs to the manufacturer as the first alternative for disposal. As a second alternative, the licensee may dispose of the devices in a licensed radioactive waste disposal facility. The manufacturer tries to keep track of each sign by knowing the end user, the serial number of the sign, and the date of manufacture. However, planes are frequently sold and their interior parts are often removed and transferred to another location. The survey done by the NRC found that although a large number of licensees (14 out of 15) kept proper records of receipt, they could only access records dating back one month. Transfer records were not well kept; only 10 out of 15 kept accurate records (NRC 1987). The survey group's main concern was the final disposition of the signs if planes are stripped before resale to other countries. The devices may or may not end up in a licensed radioactive waste area or returned to the manufacturer.

3.12.2) Scenario Development

Probabilities leading from initial events (Figure 13) are highest in the pathways for the device being transferred to an unauthorized individual (0.25) and for the device being sold to a salvage dealer (0.25). The probability of transfer to an unauthorized individual is high due to the fact that aircraft are transferred from one company to another and devices from one plane to another. The device itself would appear to have some value for a salvage dealer to resell to other individuals. The probability of the device being placed in storage is also high (0.15). Total probabilities for the device being discarded to the trash (0.20) are considerable, as the devices will not always be returned to the manufacturer as required. Probabilities for other initial pathways are lower. These include the device remaining in place but not in use (0.05) and the device being discarded to the environment (0.10 total). These were considered to be less likely because most airplanes do not sit idle for long periods of time and because the nature of these devices is such that they would not often be discarded.

The final status probabilities are highest for the device remaining in the possession of an unauthorized individual (0.47) because of the high probabilities assigned to such pathways as transfer to an unauthorized individual, recovery by another individual, and

CLASS G-2 AIRCRAFT SELF-LUMINOUS DEVICES

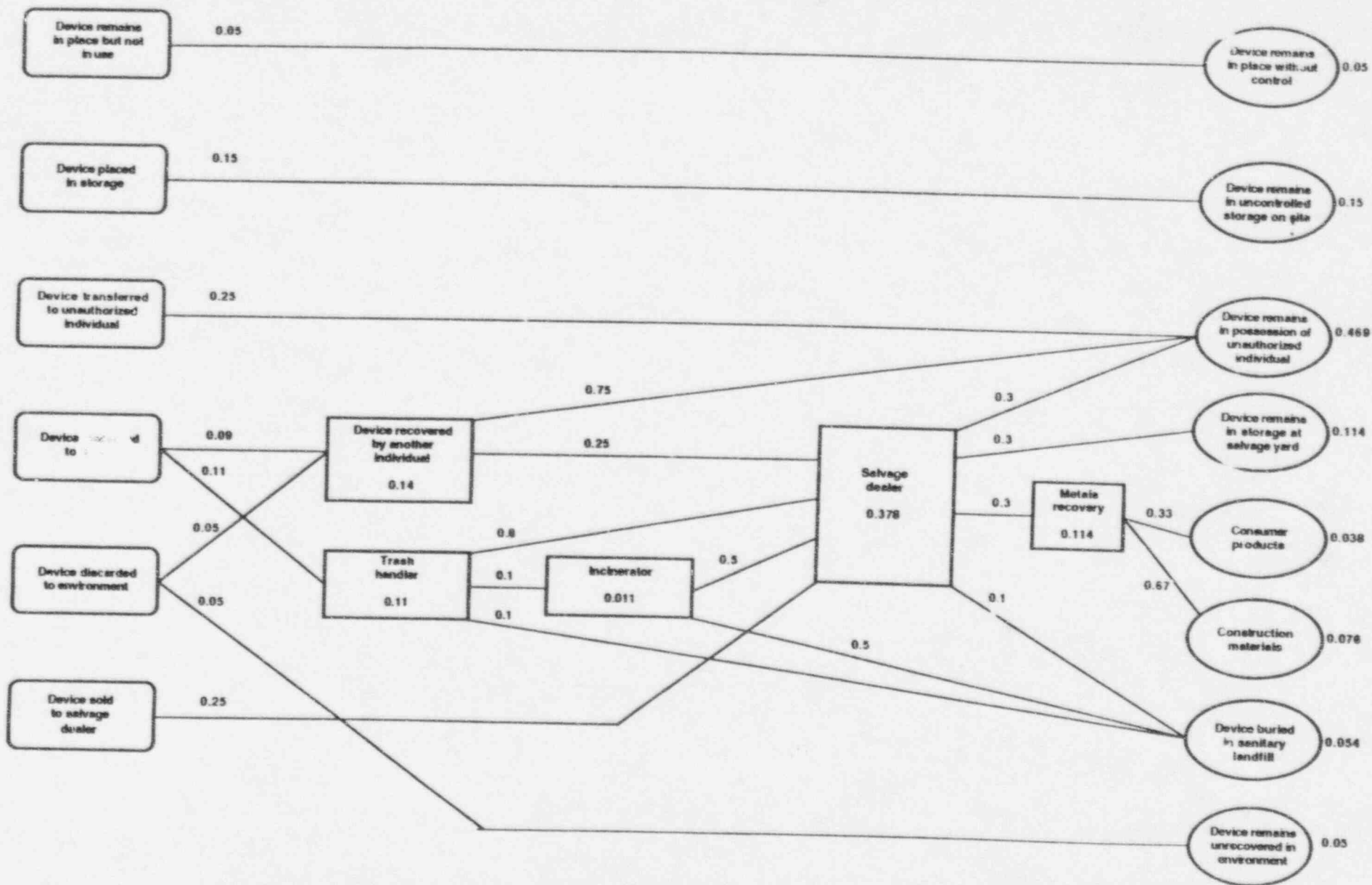


Figure 13 Scenario development for device class G-2, aircraft self-luminous devices

transfer from the salvage dealer. Other final condition probabilities are much lower for this device and do not appear to be as important.

3.12.3) Dose Assessment

3.12.3.1) 'Intact' Source

External dose - maximum individual

Table 6 shows dose equivalents for exposure to point sources at the activity levels stated in section 3.12.1 for Pm-147 for 20 weeks at 100 cm or in contact for three hours.

External dose - realistic case

The external photon dose equivalent for Pm-147 is very low for the maximum case, but the contact dose from beta emissions is very high. This indicates that the devices will not be an external hazard to the public if they are not broken open. Contact to the Pm-147 sources could produce observable skin damage in a short period of time.

Internal dose - maximum individual

As with other devices in which the activity is not easily accessible, a fraction of 0.3 of the maximum source activity was arbitrarily assigned as the maximum amount which might be inadvertently taken into the body. Intake of this much activity would result in a dose equivalent of 94 rem (0.94 Sv) to the soft tissues of the whole body for H-3. Table 92 lists dose estimates for intake of 90 mCi (3300 MBq) of Pm-147.

Internal dose - realistic case

From the study of Wilmot, an intake value of 10^{-6} was assigned to Pm-147 and a value of 10^{-4} was assigned to H-3. The dose equivalent from intake of this much H-3 would be 0.031 rem (0.31 mSv). Table 93 lists dose estimates for intake of 3×10^{-4} mCi (0.011 MBq) of Pm-147.

3.12.3.2) Dispersed Source

3.12.3.2.1) Incineration

The model of Buckley predicts an intake of 3.4×10^{-4} mCi (0.013 MBq) of H-3 and 1.0×10^{-6} mCi (3.7×10^{-5} MBq) of Pm-147. The intake of H-3 would result in a dose equivalent to the soft tissues of the whole body of 2.2×10^{-5} rem (2.2×10^{-7} Sv), 1.1×10^{-5} rem (1.1×10^{-7} Sv), and 0.80 person-rem (0.008 person-Sv) for the maximum individual, average individual, and population, respectively. Table 94 lists corresponding values for Pm-147.

3.12.3.2.2) Metals Recycling

As stated in section 2.3.2.2, a general result for Co-60 gauges is the only result afforded by the available data. See that section for a

discussion of this result.

3.12.3.2.3) Burial in Landfill

The model of Buckley for the landfill predicts that none of the Pm-147 will reach the withdrawal point, and that 17.5 mCi (648 MBq) of H-3 will reach the withdrawal point. The H-3 intake would result in a dose equivalent of 1.1 rem (0.011 Sv).

TABLE 92. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION OF
90 mCi (3.3×10^9 Bq) OF Pm-147

Organ	Ingestion	Dose Equivalents (rem)	
		Inhalation	
		Class W	Class Y
Upper Large Intestine	3.7×10^2	-	-
Lower Large Intestine	1.1×10^3	-	-
Red Marrow	-	2.7×10^3	-
Lungs	-	3.2×10^3	2.6×10^4
Bone Surfaces	-	3.3×10^4	-
Liver	-	9.0×10^3	-

TABLE 93. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION OF
 3×10^{-4} mCi (1.1×10^4 Bq) OF Pm-147

Organ	Ingestion	Dose Equivalent (rem)	
		Inhalation	
		Class W	Class Y
Upper Large Intestine	1.2×10^{-3}	-	-
Lower Large Intestine	3.6×10^{-3}	-	-
Red Marrow	-	9.1×10^{-3}	-
Lungs	-	1.1×10^{-2}	8.5×10^{-2}
Bone Surfaces	-	1.1×10^{-1}	-
Liver	-	3.0×10^{-2}	-

TABLE 94. COMMITTED DOSE EQUIVALENTS FOR INHALATION

OF Pm-147 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)		Average Individual (rem)		Population (Person-rem)	
	Class W	Class Y	Class W	Class Y	Class W	Class Y
Red Marrow	3.2×10^{-5}	-	1.6×10^{-5}	-	1.2	-
Lungs	3.7×10^{-5}	3.0×10^{-4}	1.9×10^{-5}	1.5×10^{-4}	1.4	11
Bone Surfaces	3.9×10^{-4}	-	1.9×10^{-4}	-	14	-
Liver	1.0×10^{-4}	-	5.2×10^{-5}	-	3.8	-

3.13) CLASS H - ANALYTICAL INSTRUMENTS WITH CALIBRATION SOURCES

3.13.1) Device Description

This class of device mainly includes liquid scintillation counters although a few mass spectrometers and ion mobility detectors are also included. The device usually contains Cs-137 with an activity of 0.040 mCi (1.5 MBq) and is well shielded inside a cabinet. Some devices may use 15 mCi (555 MBq) Ni-63 sources. These devices are used in analytical laboratories to set and test the calibration of the analytical equipment. Approximately 600 of these devices are distributed per year.

Liquid scintillation counters are quite similar in many respects to the gas chromatographs and XRF analyzers. They are large, complex, expensive laboratory instruments used by trained personnel. All but one licensee among twenty-three surveyed (eight used liquid scintillation counters) were aware of regulations, performed leak tests, and kept proper records (NRC 1987). Compliance with regulations is not a major problem with these instruments.

3.13.2) Scenario Development

Pathways for these devices appear similar to the three previous device classes (Figure 14). Because these laboratory instruments would probably stay in a laboratory no longer in use, the pathway to remaining in place but not in use was weightest the most heavily (0.8). Because personnel are more aware of regulations and because these instruments are expensive, transfer to unauthorized individuals would be unlikely. A probability of 0.1 was assigned to placement in storage, as some instruments are stored for later laboratory use. The device may be sold to a salvage dealer (0.1) since many of its parts are of value. The highest final probabilities included remaining in place without control and remaining in uncontrolled storage on site. Smaller final probabilities resulted for the device remaining in the possession of an unauthorized individual (0.01), remaining in storage at the salvage yard (0.03), being included in a consumer product (0.01) or construction material (0.02), or being buried in a landfill (0.03).

3.13.3) Dose Assessment

3.13.3.1) 'Intact' Source

External dose - maximum individual

Table 6 shows dose equivalents for exposure to point sources at the activity levels stated in section 3.13.1 for Cs-137 for 20 weeks at 100 cm or contact for three hours. As previously demonstrated, Ni-63 poses no external hazard.

External dose - realistic case

The external photon dose rates near one of these devices would be

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CLASS H ANALYTICAL INSTRUMENTS W/CALIBRATION SOURCES

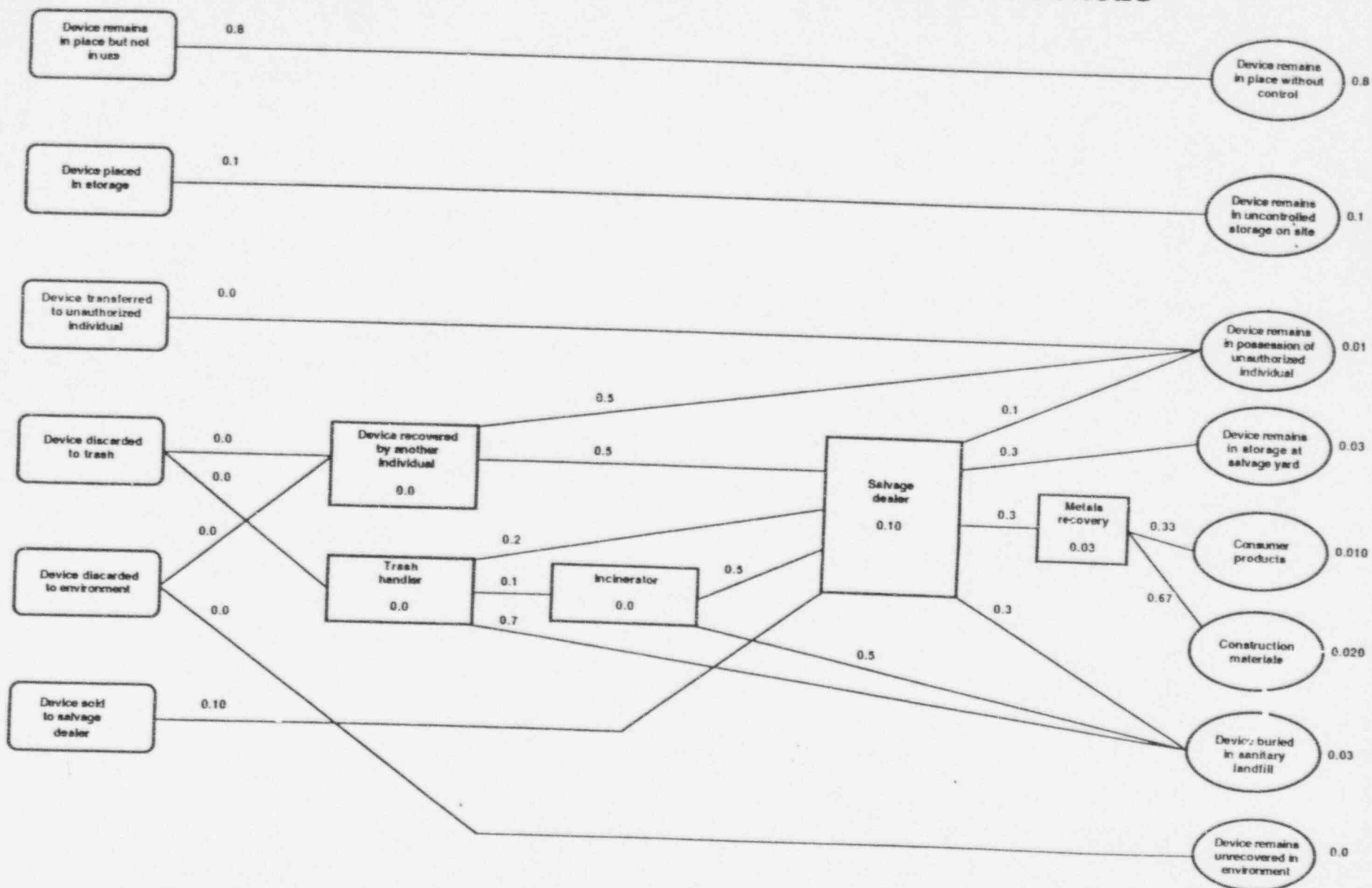


Figure 14 Scenario development for device class H, analytical instruments w/calibration sources

insignificant, because the rate for the unenclosed source at this activity level was only 51 mrem (0.51 mSv) for a 20 week exposure. The beta emissions would not be a hazard unless the unit were broken into and the source exposed, which is unlikely in a realistic scenario, because the manufacturers of these units take care to make the source inaccessible.

Internal dose - maximum individual

As with other devices in which the activity is not easily accessible, a fraction of 0.3 of the maximum source activity was arbitrarily assigned as the maximum amount which might be inadvertently taken into the body. Tables 95 and 96 lists dose estimates for intake of 0.012 mCi (0.44 MBq) of Cs-137 or 4.5 mCi (170 MBq) of Ni-63.

Internal dose - realistic case

The study of Wilmot suggests an intake value of 10^{-6} for Cs-137. In the absence of a value for Ni-63, a value of 10^{-4} was assigned. Tables 97 and 98 list dose estimates for intake of these fractions of the maximum source activity, as listed in table 1.

3.13.3.2) Dispersed Source

3.13.3.2.1) Incineration

The model of Buckley predicts an intake by the maximum individual near an incinerator which receives 5 of these devices per year to be 1.5×10^{-10} mCi (5.6×10^{-9} MBq) for Cs-137 and 5.2×10^{-8} mCi (1.9×10^{-6} MBq) for Ni-63. Tables 99 and 100 list the corresponding dose equivalents for the maximum and average individual and population.

3.13.3.2.2) Metals Recycling

As stated in section 2.3.2.2, a general result for Co-60 gauges is the only result afforded by the available data. See that section for a discussion of this result.

3.13.3.2.3) Burial in Landfill

The landfill model proposed by Buckley predicts that neither Cs-137 nor Ni-63 will reach the withdrawal point before they are removed by radioactive decay.

TABLE 95. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION

OF 0.012 mCi (4.4×10^5 Bq) OF Cs-137

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Gonads	6.2×10^{-1}	3.9×10^{-1}
Breast	5.3×10^{-1}	3.5×10^{-1}
Red Marrow	5.8×10^{-1}	3.7×10^{-1}
Lungs	5.8×10^{-1}	3.9×10^{-1}
Thyroid	5.8×10^{-1}	3.5×10^{-1}
Bone Surfaces	5.8×10^{-1}	3.5×10^{-1}
Small Intestine	6.2×10^{-1}	4.0×10^{-1}
Upper Large Intestine	6.2×10^{-1}	4.0×10^{-1}
Lower Large Intestine	6.2×10^{-1}	4.0×10^{-1}
Remainder of Body	6.7×10^{-1}	4.2×10^{-1}

TABLE 96. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION

OF 4.5 mCi (1.7×10^8 Bq) OF Ni-63

Organ	Dose Equivalent (rem)		
	Ingestion	Inhalation Class D	Class W
Gonads	1.4×10^0	1.4×10^1	4.2×10^0
Breast	1.4×10^0	1.4×10^1	4.2×10^0
Red Marrow	1.4×10^0	1.4×10^1	-
Lungs	1.4×10^0	1.4×10^1	5.2×10^1
Stomach Wall	1.7×10^0	1.4×10^1	-
Small Intestine	2.2×10^0	1.4×10^1	-
Upper Large Intestine	6.0×10^0	1.4×10^1	-
Lower Large Intestine	1.5×10^1	1.6×10^1	1.1×10^1
Kidneys	-	1.4×10^1	-
Thyroid	-	1.4×10^1	-
Bone Surfaces	-	1.4×10^1	-

TABLE 97. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION

OF 4×10^{-8} mCi (1.48 Bq) of Cs-137

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Gonads	2.1×10^{-6}	1.3×10^{-6}
Breast	1.8×10^{-6}	1.2×10^{-6}
Red Marrow	1.9×10^{-6}	1.2×10^{-6}
Lungs	1.9×10^{-6}	1.3×10^{-6}
Thyroid	1.9×10^{-6}	1.2×10^{-6}
Bone Surfaces	1.9×10^{-6}	1.2×10^{-6}
Small Intestine	2.1×10^{-6}	1.3×10^{-6}
Upper Large Intestine	2.1×10^{-6}	1.3×10^{-6}
Lower Large Intestine	2.1×10^{-6}	1.3×10^{-6}
Remainder of Body	2.2×10^{-6}	1.4×10^{-6}

TABLE 98. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION

OF 1.5×10^{-3} mCi (5.5×10^4 Bq) of Ni-63

Organ	Ingestion	Dose Equivalent (rem)	
		Inhalation Class D	Class W
Gonads	4.7×10^{-4}	4.6×10^{-3}	1.4×10^{-3}
Breast	4.7×10^{-4}	4.6×10^{-3}	1.4×10^{-3}
Red Marrow	4.7×10^{-4}	4.6×10^{-3}	-
Lungs	4.7×10^{-4}	4.8×10^{-3}	1.8×10^{-2}
Stomach Wall	5.6×10^{-4}	4.6×10^{-3}	-
Small Intestine	7.2×10^{-4}	4.6×10^{-3}	-
Upper Large Intestine	2.0×10^{-3}	4.8×10^{-3}	-
Lower Large Intestine	5.1×10^{-3}	5.3×10^{-3}	3.7×10^{-3}
Thyroid	-	4.6×10^{-3}	-
Bone Surfaces	-	4.6×10^{-3}	-
Kidneys	-	4.6×10^{-3}	-

TABLE 99. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Cs-137 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)	Average Individual (rem)	Population (Person-rem)
Gonads	4.8×10^{-9}	2.4×10^{-9}	1.7×10^{-4}
Breast	4.2×10^{-9}	2.1×10^{-9}	1.5×10^{-4}
Red Marrow	4.5×10^{-9}	2.2×10^{-9}	1.6×10^{-4}
Lungs	4.8×10^{-9}	2.4×10^{-9}	1.7×10^{-4}
Thyroid	4.3×10^{-9}	2.1×10^{-9}	1.6×10^{-4}
Bone Surfaces	4.3×10^{-9}	2.1×10^{-9}	1.6×10^{-4}
Small Intestine	4.9×10^{-9}	2.5×10^{-9}	1.8×10^{-4}
Upper Large Intestine	4.9×10^{-9}	2.5×10^{-9}	1.8×10^{-4}
Lower Large Intestine	4.9×10^{-9}	2.5×10^{-9}	1.8×10^{-4}
Remainder of Body	5.3×10^{-9}	2.6×10^{-9}	1.9×10^{-4}

TABLE 100. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Ni-63 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)		Average Individual (rem)		Population (Person-rem)	
	Class D	Class W	Class D	Class W	Class D	Class W
Gonads	1.6×10^{-7}	4.8×10^{-8}	7.9×10^{-8}	2.4×10^{-8}	5.8×10^{-3}	1.6×10^{-3}
Breast	1.6×10^{-7}	4.8×10^{-8}	7.9×10^{-8}	2.4×10^{-8}	5.8×10^{-3}	1.6×10^{-3}
Red Marrow	1.6×10^{-7}	-	7.9×10^{-8}	-	5.8×10^{-3}	-
Lungs	1.7×10^{-7}	6.0×10^{-7}	8.4×10^{-8}	3.0×10^{-7}	6.1×10^{-3}	2.2×10^{-2}
Thyroid	1.6×10^{-7}	-	7.9×10^{-8}	-	5.8×10^{-3}	-
Bone Surfaces	1.6×10^{-7}	-	7.9×10^{-8}	-	5.8×10^{-3}	-
Stomach Wall	1.6×10^{-7}	-	8.0×10^{-8}	-	5.8×10^{-3}	-
Small Intestine	1.6×10^{-7}	-	8.0×10^{-8}	-	5.8×10^{-3}	-
Upper Large Intestine	1.7×10^{-7}	-	8.4×10^{-8}	-	6.1×10^{-3}	-
Lower Large Intestine	1.8×10^{-7}	1.3×10^{-7}	9.2×10^{-8}	6.4×10^{-8}	6.7×10^{-3}	4.7×10^{-3}
Kidneys	1.6×10^{-7}	-	7.9×10^{-8}	-	5.8×10^{-3}	-

3.14) CLASS I - CALIBRATION OR REFERENCE SOURCES - Am-241

3.14.1) Device Description

These devices are identical in nature to those described in section 3.10, except that they contain Am-241. They are covered in a separate section in this report because they are regulated under a separate section in the Code of Federal Regulations (see table 1). Source activities may be as high as 0.005 mCi (0.185 MBq).

3.14.2) Scenario Development

Because the sources are identical in nature to those in section 3.10, the same set of scenario probabilities are assumed to apply (Figure 15).

3.14.3) Dose Assessment

3.14.3.1) 'Intact' Source

External dose - maximum individual

Table 6 shows dose equivalents for exposure to point sources at the activity levels stated in section 3.14.1 for Am-241 for 20 weeks at 100 cm or in contact for three hours.

External dose - realistic case

Both of the dose estimates for external exposure to these sources were very low in the maximum case, and will not be significant in a realistic situation.

Internal dose - maximum individual

As discussed for the sources in section 3.10, 0.3 of the source activity was assumed to be free if the entire source was swallowed. Table 101 lists dose estimates for intake of 0.0015 mCi (0.056 MBq) of Am-241.

Internal dose - realistic case

From the study of Wilmot, a intake value of 10^{-6} was assigned to Am-241. Table 102 lists dose estimates for intake of 5×10^{-9} mCi (1.85×10^{-7} MBq) of Am-241.

In an incident in which a technologist inadvertently swallowed a ceramic disk containing 0.0028 mCi (0.10 MBq) of Am-241, Smith et al. (Smith 1983) found that the disk passed through the gastrointestinal system with almost no loss of activity from the disk. They estimated a dose equivalent to the lower large intestine is 17 rem (0.17 Sv), and assumed that only 1.6×10^{-7} mCi (5.8 Bq) were absorbed into the body system, which would result in a dose equivalent of 13 rem (0.13 Sv) to the bone surfaces, based on the ICRP 30 model. This incident did not involve a source with characteristics similar to the ones commonly used

CLASS I CALIBRATION OR REFERENCE SOURCES (Am-241)

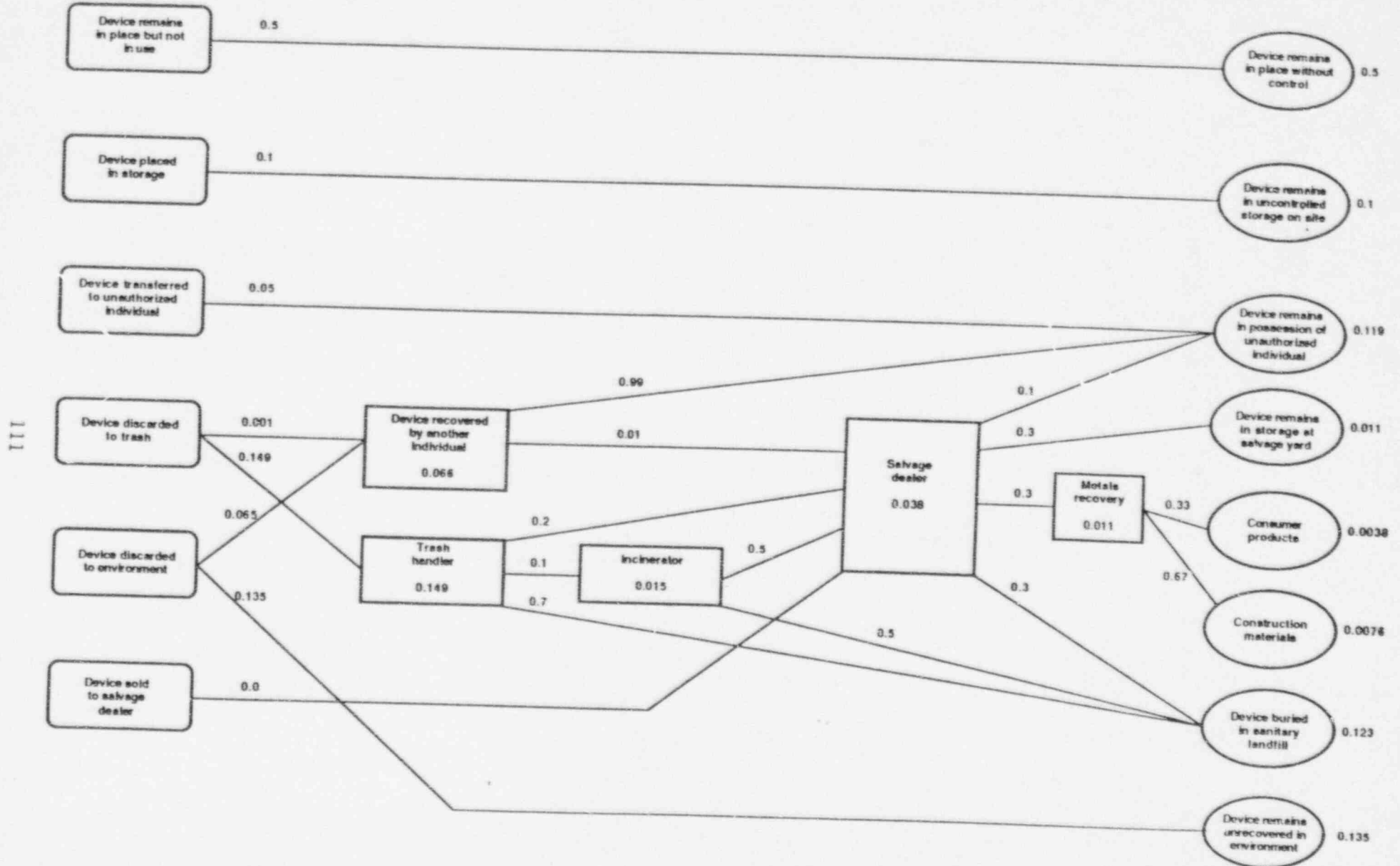


Figure 15 Scenario development for device class I, calibration or reference sources (Am-241)

for detector calibration, but the case study does demonstrate that some decrease in the radiation dose can occur if the source passes through the intestines without much of the activity leaching off, as would be expected in most cases.

3.14.3.2) Dispersed Source

3.14.3.2.1) Incineration

The model of Buckley predicts an intake of 3.5×10^{-11} mCi (1.3×10^{-9} MBq) of Am-241 from incineration of five sources per year. Table 103 lists dose estimates for the maximum and average individual and the population from intakes corresponding to this estimate.

3.14.3.2.2) Metals Recycling

As stated in section 2.3.2.2, a general result for Co-60 gauges is the only result afforded by the available data. See that section for a discussion of this result.

3.14.3.2.3) Burial in Landfill

The model of Buckley for the landfill predicts that none of the Am-241 will reach the withdrawal point.

TABLE 101. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION

OF 0.0015 mCi (5.6×10^4 Bq) OF Am-241

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Gonads	7.7×10^{-1}	1.8×10^2
Red Marrow	4.7×10^0	1.1×10^3
Bone Surfaces	6.1×10^1	1.4×10^4
Liver	1.3×10^1	3.0×10^3

TABLE 102. COMMITTED DOSE EQUIVALENTS FROM INGESTION OR INHALATION

OF 5×10^{-9} mCi (1.85×10^{-7} Bq) OF Am-241

Organ	Dose Equivalent (rem)	
	Ingestion	Inhalation
Gonads	2.6×10^{-6}	5.9×10^{-4}
Red Marrow	1.6×10^{-5}	3.7×10^{-3}
Bone Surfaces	2.0×10^{-4}	4.6×10^{-2}
Liver	4.2×10^{-5}	1.0×10^{-2}

TABLE 103. COMMITTED DOSE EQUIVALENTS FOR INHALATION

OF Am-241 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)	Average Individual (rem)	Population (Person-rem)
Gonads	4.1×10^{-6}	2.1×10^{-6}	0.15
Red Marrow	2.6×10^{-5}	1.3×10^{-5}	0.94
Bone Surfaces	3.2×10^{-4}	1.6×10^{-4}	12 -
Liver	7.1×10^{-5}	3.5×10^{-5}	2.6

3.15) CLASS J - SMALL QUANTITIES OF SOURCE MATERIAL

Included in this class are two items in the NRC registry and both are calibration standards for logging tools. The sources are encased in stainless steel and are attached to the end of a long cable. Nuclides used and typical activities are natural uranium, 0.005 mCi (0.185 MBq) and Th-232, 0.005 mCi (0.185 MBq).

Companies which receive this material to produce these sources are limited by 10CFR40.22 to reception of 15 pounds of material at any one time, and no more than 150 pounds per calendar year. No restrictions have been placed on disposal up to this activity limit.

The sources themselves represent minimal external hazards. External dose rates near these encapsulated sources are very low. Risks from intake of the activity if released through machining to produce the sources or damage to an existing source are higher from the chemical toxicity than from the radiotoxicity. No published data could be located which would help with either scenario development or dose assessment, so a detailed analysis, as was done for the other device classes, was not performed.

3.16) CLASS K - CALIBRATION OR REFERENCE SOURCES - Pu-239

3.16.1) Device Description

These devices are identical in nature to those described in sections 3.10 and 3.14, except that they contain Pu-239. They are covered in a separate section in this report because they are covered in a separate section in the Code of Federal Regulations (see table 1). Source activities may be as high as 0.005 mCi (0.185 MBq).

3.16.2) Scenario Development

Because the sources are identical in nature to those in section 3.10 and 3.14, the same set of scenario probabilities are assumed to apply (Figure 16).

3.16.3) Dose Assessment

3.16.3.1) 'Intact' Source

External dose - maximum individual

Table 6 shows dose equivalents for exposure to point sources at the activity levels stated in section 3.16.1 for Pu-239 for 20 weeks at 100 cm or in contact for three hours.

External dose - realistic case

The dose estimate for external exposure to these sources is very low for the maximum case, and will not be significant in a realistic situation.

Internal dose - maximum individual

As with the sources in section 3.10 and 3.14, 0.3 of the source activity was assumed to be free if the entire source was swallowed. Table 104 lists dose estimates for intake of 0.0015 mCi (0.056 MBq) of Pu-239.

Internal dose - realistic case

From the study of Wilmot, a intake value of 10^{-6} was assigned to Pu-239. Table 105 lists dose estimates for intake of 5×10^{-9} mCi (1.85×10^{-7} MBq) of Pu-239.

3.16.3.2) Dispersed Source

3.16.3.2.1) Incineration

The model of Buckley predicts an intake of 1.7×10^{-11} mCi (6.4×10^{-10} MBq) of Pu-239 from incineration of five sources per year. Table 106 lists dose estimates for the maximum and average individual and the population from inhalation of this amount of activity.

CLASS K CALIBRATION OR REFERENCE SOURCES (Pu-239)

116

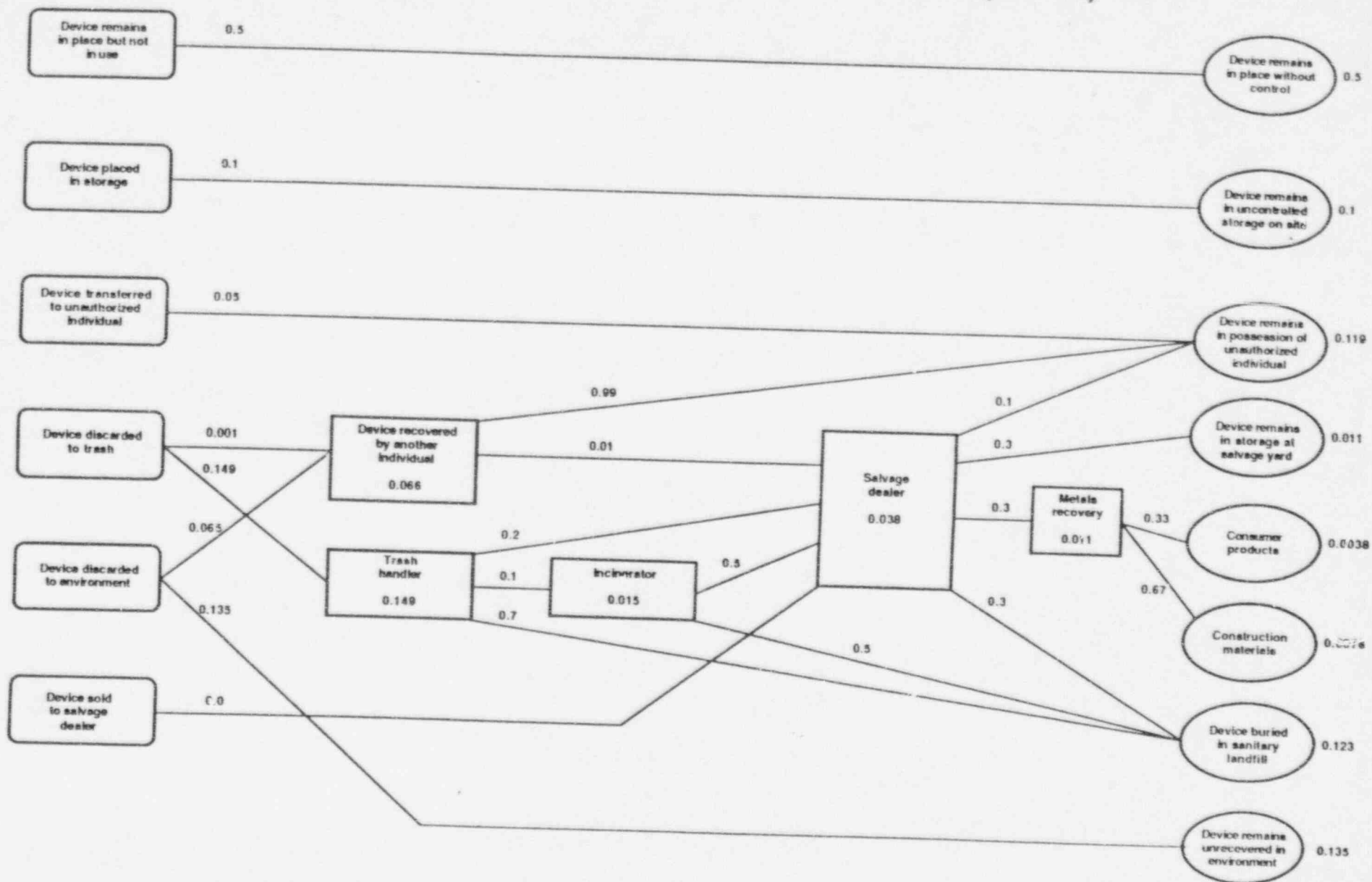


Figure 16 Scenario development for device class K, calibration or reference sources (Pu-239)

3.16.3.2.2) Metals Recycling

As stated in section 2.3.2.2, a general result for Co-60 gauges is the only result afforded by the available data. See that section for a discussion of this result.

3.16.3.2.3) Burial in Landfill

The model of Buckley for the landfill predicts an intake at the withdrawal point of 9.07×10^{-6} mCi (3.4×10^{-4} MBq) of Pu-239. Table 107 lists dose estimates for ingestion of this amount of activity.

TABLE 104. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION
OF 0.0015 mCi (0.0555 MBq) OF Pu-239

Organ	Dose Equivalent (rem)			
	Ingestion $f_1=1 \times 10^{-4}$	$f_1=1 \times 10^{-5}$	Inhalation Class W	Class Y
Gonads	0.14	0.014	180	-
Red Marrow	0.89	0.089	1100	420
Bone Surfaces	12	1.2	14000	5300
Liver	2.4	0.24	2900	1200
Lungs	-	-	-	1800
Upper Large Intestine	-	0.094	-	-
Lower Large Intestine	-	0.29	-	-

TABLE 105. COMMITTED DOSE EQUIVALENTS FOR INGESTION OR INHALATION
OF 5×10^{-9} mCi (1.85×10^{-7} MBq) OF Pu-239

Organ	Dose Equivalent (rem)			
	Ingestion $f_1=1 \times 10^{-4}$	$f_1=1 \times 10^{-5}$	Inhalation Class W	Class Y
Gonads	4.8×10^{-7}	4.8×10^{-8}	5.9×10^{-4}	-
Red Marrow	3.0×10^{-6}	3.0×10^{-7}	3.7×10^{-3}	1.4×10^{-3}
Bone Surfaces	3.9×10^{-5}	3.9×10^{-6}	4.6×10^{-2}	1.8×10^{-2}
Liver	8.1×10^{-6}	8.1×10^{-7}	9.8×10^{-3}	3.9×10^{-3}
Lungs	-	-	-	5.9×10^{-3}
Upper Large Intestine	-	3.1×10^{-7}	-	-
Lower Large Intestine	-	9.8×10^{-7}	-	-

TABLE 106. COMMITTED DOSE EQUIVALENTS FOR INHALATION
OF Pu-239 FROM INCINERATOR EMISSIONS

Organ	Maximum Individual (rem)		Average Individual (rem)		Population (person-rem)	
	Class W	Class Y	Class W	Class Y	Class W	Class Y
Gonads	2.1×10^{-6}	-	1.0×10^{-6}	-	0.077	-
Red Marrow	1.3×10^{-5}	4.9×10^{-6}	6.5×10^{-6}	2.4×10^{-6}	0.47	0.18
Bone Surfaces	1.6×10^{-4}	6.1×10^{-5}	8.0×10^{-5}	3.0×10^{-5}	5.8	2.2
Liver	3.4×10^{-5}	1.3×10^{-5}	1.7×10^{-5}	6.5×10^{-6}	1.2	0.47
Lungs	-	2.1×10^{-5}	-	1.0×10^{-5}	-	0.77

TABLE 107. COMMITTED DOSE EQUIVALENTS FOR INGESTION OF
 9.07×10^{-6} mCi (3.4×10^{-4} MBq) OF Pu-239

Organ	Dose Equivalent (rem)	
	$f_1=1 \times 10^{-4}$	$f_1=1 \times 10^{-5}$
Gonads	8.7×10^{-4}	8.7×10^{-5}
Red Marrow	5.4×10^{-3}	5.4×10^{-4}
Bone Surfaces	7.0×10^{-2}	7.0×10^{-3}
Liver	1.5×10^{-2}	1.5×10^{-3}
Upper Large Intestine	-	5.7×10^{-4}
Lower Large Intestine	-	1.8×10^{-3}

4) Summary and Conclusions

Although the results generated through this analysis were based on a small amount of data, some conclusions about public health concerns are evident. The two major dosimetry concerns are external irradiation and internal irradiation. The former concern relates to exposure to the radiation field from the radioactive source in a device or contact to the source. Internal irradiation may occur from improper handling of the device, in which activity may be released from the source, or intake of radioactivity which has been dispersed over a wide area because of the device inadvertently being incinerated, passed through metals recycling, or buried in a landfill, with leaching of the radioactive material into public drinking water supplies.

Exposure to the radiation field was not a significant concern for most of the devices, even under a worst case assumption of exposure for 20 weeks at 100 cm. Devices for which exposures under this assumption were greater than 500 mrem (0.005 Sv) include all types of gamma gauges (class B), X-ray fluorescence analyzers (class E-1 and E-2), and self-luminous devices containing Kr-85 (class G-1). In almost all cases, contact to the radioactive source for three hours will produce localized radiation doses which are inappropriate for members of the general public. For many devices, the source is extremely inaccessible, and these dose equivalents would not be expected (e.g., gas chromatographs, analytical instruments with calibration or reference sources). In other device types, however, the source can be accessed through a built-in mechanism, and significant radiation doses can occur if the device is manipulated by persons unaware of the radiation hazards (see table 6).

Internal radiation doses were derived for several situations. A worst case situation was defined as intake of 30% of the radioactive material by inhalation or ingestion. More realistic numbers were generated by assuming a much smaller fraction, usually between 10^{-6} and 10^{-4} . Estimates were also generated for inhalation of radioactive material if a device were incinerated, and for ingestion of radioactive material which may have leached from a landfill site. For these situations, the results varied greatly, depending on the radiotoxicity and activity level of the radionuclide concerned. Significant estimates were derived for inhalation of Am-241, Ra-226, Cm-244, Pm-147, Pu-239, and, to a lesser extent, Po-210, Pb-210, Fe-55, Ni-63, Sr-90, Co-60, Cs-137, and Cd-109. High dose equivalents were estimated for ingestion of Am-241, Cm-244, Pm-147, and, to a lesser extent, Po-210, Pb-210, Fe-55, Sr-90, Co-60, Cs-137, and Cd-109. Dose equivalents from intake of material from the plume of an incinerator were highest for Am-241, Ra-226, and Cm-244. Ingestion of radioactive material from landfills were significant for Ra-226 and Sr-90. The model used for transport from the landfill predicted very long holdup times in the landfill soil matrix for some nuclides, which resulted in the prediction that no Am-241, Cm-244, or Pm-147 would be ingested. These nuclides were all significant hazards if ingested, and would be of concern if the migration rates were significantly faster than predicted by this model.

Although more literature was available for consequences of metal recycling of radionuclides than for any other category, the only quantitative information listed related to Co-60. Therefore, this analysis was performed only for Co-60 in gamma gauges. Inferences based on available literature showed that dose equivalents received by members of the general public who purchase contaminated products would most likely not exceed 500 mrem/yr (0.005 Sv/yr) in most cases.

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APPENDIX. DESCRIPTIONS

This Appendix contains brief descriptions of the devices considered in this report with typical nuclides and activities mentioned.

A-1 STATIC ELIMINATORS: HAND-HELD/PORTABLE/SMALL BRUSHES (10 CFR 31.3)

The 3M Company "Static Master" devices using Po-210 (138 day half-life) in microsphere encapsulation are small size, low cost, and have no salvage value. Studies have shown that microspheres can be dislodged from the foil source, but normally would not create an internal hazard unless microspheres are damaged. Activities when distributed are up to 0.50 mCi (18.5 MBq). Devices are supposed to be returned after a year when activity is less than 0.080 mCi (3.0 MBq).

A-2 STATIC ELIMINATORS OR DETECTORS: IN EQUIPMENT OR PROCESS LINE (VERY HIGH TOXICITY SOURCES) (10 CFR 31.5)

These ionization sources are used to remove static charge build-up in equipment or on production lines (conveyer belts, roller systems) or in air ducts, etc. Detectors are used to sense and measure static charge. Typical nuclides and activities include Po-210, up to 100 mCi (3700 MBq), Am-241, 0.0005 mCi (0.0185 MBq), and Ra-226, 0.0005 mCi (0.0185 MBq).

A-3 STATIC ELIMINATORS OR DETECTORS: IN EQUIPMENT OR ON PROCESS LINE (LOW TOXICITY SOURCES) (10 CFR 31.5)

Same description as A-2 but typical nuclides and activities are: H-3, 250 mCi (9250 MBq) and Kr-85, 2 mCi (74 MBq).

B GAMMA GAUGES: (10 CFR 31.5)

For this study these devices with activities less than 20 mCi (740 MBq) are included. Normally these are permanently mounted on the process line to measure thickness, level, density, etc. There is potential for damage in the normal work environment. Common nuclides with typical activities are: Co-60, 10 mCi (370 MBq), Cs-137, 20 mCi (740 MBq), Am-241, 20 mCi (740 MBq), and Ra-226, 10 mCi (370 MBq).

C-1 BETA GAUGES: BACKSCATTER TYPE. (10 CFR 31.5)

These devices are widely used in production facilities to monitor process lines or measure thickness, density, composition, etc. Devices are sometimes permanently mounted but are often portable and hand-held. There is potential for damage in the normal work environment. Common nuclides with typical activities are: Sr-90, 0.025 mCi (0.925 MBq), Tl-204, 0.10 mCi (3.7 MBq), Ru-106, 0.025 mCi (0.925 MBq), Pm-147, 0.050 (1.85 MBq), C-14, 0.050 mCi (1.85 MBq), and Pb-210, 0.010 mCi

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(0.37 MBq).

C-2 BETA GAUGES: TRANSMISSION TYPE (10 CFR 31.5)

For this study these devices with activities less than 20 mCi (740 MBq) are included. These devices are typically mounted permanently on the process line. There is potential for damage in normal work environment. Sources consist almost exclusively of Sr-90 sources of up to 20 mCi (740 MBq).

D GAS CHROMATOGRAPHS (10 CFR 31.5)

Devices consist of laboratory analytical instruments containing ionization sources in detector cells or electron capture detectors. Instruments may have interchangeable detector cells. Older units used Sr-90 or Ra-226, but there are no examples currently in the registry. They are subject to damage due to chemical environment or during cleaning. Common nuclides with activities are: Ni-63, 20 mCi (740 MBq) and H-3, 1000 mCi (37 GBq).

E-1 X-RAY FLUORESCENCE ANALYZERS: VERY HIGH TOXICITY SOURCES (10 CFR 31.5)

These analytical instruments using radioactive sources may be used in the laboratory or are designed to be portable for field use or to monitor a production line. They are supplied with source installed. Separate source/detector modules may be on hand. Source is well shielded in normal use. Common nuclides with activities include: Am-241, 30 mCi (1100 MBq) and Cm-244, 100 mCi (3700 MBq).

E-2 X-RAY FLUORESCENCE ANALYZERS: MODERATE TOXICITY SOURCES (10 CFR 31.5)

Same description as F-1, but common nuclides with activities are: Cd-109, 20 mCi (740 MBq) and Fe-55, 100 mCi (3700 MBq).

F SOURCES FOR CHECKING DETECTOR OPERATION OR CALIBRATION AND ANALYTICAL REFERENCE SOURCES (10 CFR 31.5)

These small sources, often supplied by the detector manufacturer, are used to check performance in the field or as calibration or analytical standards. Their small size makes them susceptible to loss or theft. Source could be damaged in work environment. Typical nuclides with activities are: Ra-226, 0.004 mCi (0.15 MBq), Cs-137, 0.10 mCi (3.7 MBq), Co-60, 0.01 mCi (0.37 MBq) and Sr-90, 0.001 mCi (0.037 MBq).

G-1 SELF-LUMINOUS DEVICES (10 CFR 31.5)

Lighted warning signs which include exit signs, emergency light sources, safety markers, and light wands are in this class. Also included are very low activity lighted gunsights or reference photometric standards. Common nuclides with activities in are: H-3, 5000 mCi (185 GBq), Kr-85, 1700 mCi (62.9 GBq), and C-14, 0.10 mCi (3.7

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MBq).

G-2 SELF-LUMINOUS DEVICES IN AIRCRAFT (10 CFR 31.7)

Lighted signs and safety markers are permanently mounted to minimize theft or vandalism. Nuclides include H-3 with up to 5000 mCi (185 GBq) of activity and Pm-147 with an activity of 300 mCi (11 GBq).

H ANALYTICAL INSTRUMENTS CONTAINING SMALL CALIBRATION OR REFERENCE SOURCES (10 CFR 31.5)

These are usually liquid scintillation counters with built in Cs-137, 0.040 mCi (1.5 MBq) or Ni-63, 15 mCi (555 MBq) reference sources but may also include a few mass spectrometers or ion mobility detectors. Devices are used in an analytical laboratory. Sources are well shielded.

I SOURCES FOR CHECKING DETECTOR OPERATION OR CALIBRATION AND ANALYTICAL REFERENCE SOURCES (10 CFR 31.8)

Same description as class F, but these include Am-241, 0.005 mCi (0.185 MBq).

J SMALL QUANTITIES OF SOURCE MATERIAL (10 CFR 40.22)

Only two items are in the registry and both represent calibration standards for logging tools. Nuclides and activities include: Th-232, 0.005 mCi (0.185 MBq) and natural uranium, 0.005 mCi (0.185 MBq).

K SOURCES FOR CHECKING DETECTOR OPERATION OR CALIBRATION AND ANALYTICAL REFERENCE SOURCES (10 CFR 70.19)

Same description as class F, but these include Pu-239, 0.005 mCi (0.185 MBq).

JK
 copy comment?
 SD

MEMORANDUM

DECEMBER 16, 1990

TO: Stephen Baggett, NMSS/NRC
 FROM: Craig Dean
 RE: Draft Report on Survey of 10 CFR §31.5 General Licensees

Enclosed is a draft of the subject report. It represents the preliminary results of the survey effort, in several respects:

- We have not yet prepared a summary of the report or final conclusions and recommendations. Several additional data analysis tasks must be completed before we can feel confident of what the final "message" will be, and we will also want to discuss those findings with you. We are still collating information, such as corrections to data taken from quarterly reports on types of devices held by survey respondents, that is not easily amenable to simple printouts. Similarly, we are still culling information that was presented in written supplements to the questionnaires, in letters, etc. Some of this may change slightly the picture presented from the tabulated yes/no answers.
- Several data tables that will eventually form appendices to the report still contain numerous duplications, due to different ways of spelling firm names, addresses, etc., and have not been included in this draft. We have given the general sense of such data in the text, and will supply you with printouts as soon as the cleanup of the data is completed.
- Estimates of the cost per survey respondent will be affected by the on-site visits that we are currently conducting. In addition, this draft reports data that are about 7 to 10 days old. Instead of calculating the cost per survey for this draft, we will provide you with a memorandum in the near future and will incorporate final cost information into the final draft.

We are looking forward to your comments on this preliminary discussion of results. A number of additional analytic points may occur to you as you read this draft. Please let me know at your convenience of any changes or additions that you would like us to make in future drafts.

DRAFT REPORT ON SURVEY OF GENERAL LICENSEES UNDER 10 CFR 31.5

Prepared by C.M. Dean, M. S. Lawrence, H. Lester

ICF Incorporated

December 1990

Prepared for
U.S. Nuclear Regulatory Commission

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DRAFT REPORT ON SURVEY OF GENERAL LICENSEES UNDER 10 CFR 31.5

Manuscript Completed: December 1990

Prepared by C.M. Dean, M.S. Lawrence, H. Lester


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Prepared for
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Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NRC FIN D 2554-0

[REDACTED]

ABSTRACT

The NRC for several years has been studying the regulatory framework for the licensing of the possession and use of certain measuring and gauging devices containing nuclear materials. In particular, the NRC has evaluated reports of incidents involving devices that are held under general licenses. This report describes the procedures that were followed and the results that were obtained from a survey of general licensees under 10 CFR §31.5. A sample of general licensees from all non-Agreement States for three categories of devices -- gauges, analytic instruments, and self-powered exit lights -- was contacted by mail with a questionnaire designed to obtain information about the respondents' knowledge of the regulatory requirements for general licensees and their practices and procedures concerning maintenance, testing, and disposition of the generally licensed devices. The response rates for the survey were between 85 and 95 percent, depending on the type of device. Although a high proportion of the general licensees displayed knowledge of the regulatory requirements and compliance with them, a significant number of discrepancies were identified.



SUMMARY

[To be provided]

[REDACTED]

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5.0 CONCLUSIONS AND RECOMMENDATIONS.[To be provided]

APPENDIX A: SURVEY QUESTIONNAIRES

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1.0 INTRODUCTION


1.1 Purpose of Report

This report describes the procedures that were followed and the results that were obtained from a mail survey conducted for the Office of Nuclear Materials Safety and Safeguards (NMSS) by ICF Incorporated (ICF) during 1990. The survey involved NRC general licensees under 10 CFR Part 31.5. A sample of general licensees from all non-Agreement States for three categories of devices containing nuclear materials licensed under Section 31.5 -- gauges, analytic devices, and self-powered exit signs -- was contacted by mail with a questionnaire designed to obtain information about the respondents' knowledge of the regulatory requirements for general licensees and their practices and procedures concerning maintenance, testing, and disposition of the generally licensed devices. This report presents the results of the survey.

1.2 Background

NRC has obtained information in the past about situations in which the regulatory requirements pertaining to general licensed materials have not been satisfied. This information has come primarily from inspection findings and from reports from general licensees of incidents involving devices containing radioactive isotopes. Such situations have included failures to conduct leakage tests and tests of the device on-off mechanism and indicator, if any, at the specified intervals; failure to maintain required records; failure to comply with reporting requirements; failure to comply with the requirements concerning transfers and custody of devices; and failure to comply with labeling requirements. Since about 1982, NRC has been collecting information and conducting studies of the general license regulatory framework. Between 1984 and 1986, NRC conducted an extensive study and prepared a report on this subject. The NRC report concluded that there were several areas of safety concern due to inadequate accountability of the devices and frequent lack of awareness on the part of the users of the contents and requirements of the pertinent regulations, particularly the rules on transfers, disposal, and recordkeeping.

In 1989, NRC contracted with ICF Incorporated to conduct another study of general licensees in the form of a mail survey. The survey was undertaken to accomplish at least two goals:

- (1) To obtain additional information about current general licensees. Survey respondents were firms that had obtained one or more devices during the period 1985 to 1990.
 - (2) To investigate whether mail survey techniques could usefully support NRC in informing general licensees of their responsibilities and determining the extent to which general licensees know, understand, and comply with the requirements in 10 CFR 31.5.
- 

1.3 Data Sources

The survey frame was provided to ICF by NRC from quarterly reports submitted by specific licensees during the period from about 1985 to early 1990. These quarterly reports list firms to which devices are shipped during the reporting period.

In addition to the firm names and addresses of general licensees provided on quarterly reports, information about members of the survey frame was also obtained by ICF through telephone calls and through library research. This information was developed after the initial distribution of the survey, when it became clear that certain names and/or addresses taken from the quarterly reports were no longer correct. ICF also conducted telephone followup to determine the correct name of the person who was serving as the radiation safety officer or other firm contact for the device, in cases when the name and address of the firm appeared to be correct but no survey response had been obtained. Finally, survey respondents provided additional information on survey forms about firm names, addresses, and contacts.


Information about the devices held by the general licensees was obtained initially from quarterly reports. The survey questionnaire requested that such information be verified and corrected if necessary by the respondent, and general licensees themselves provided information about the types and serial numbers of devices in their possession through survey responses. In a number of cases, respondents did correct the information on the questionnaire.

1.4 Limitations

Quarterly reports did not always contain complete correct addresses for firms to which devices were shipped. In addition, the reports only identified the firm to which a device was shipped initially; if the firm then transhipped the device the quarterly report did not indicate the ultimate recipient of the device. Finally, quarterly reports were in some cases unclear about whether they were describing devices that had been shipped to respondents or shipped back from respondents to specific licensees. In a number of cases, respondents reported that the device listed on the questionnaire had been returned to the vendor.

The list of potential respondents initially furnished by NRC did not include the names of 500 exit sign licensees, 1000 analytic device licensees, and 1500 gauge licensees, which was the goal for the final size of the survey. A supplementary sample therefore was chosen. Some members of the supplementary sample were chosen in error from Agreement States, and at the request of NRC those potential respondents were not followed up or included in the results. The final sample size was reduced by subtraction of these potential respondents.

Some potential respondents were vendors or electrical contractors who had shipped the devices to ultimate users. In some cases the vendor or contractor had no information about the ultimate destination and/or current



condition of the device. In other cases, the radiation safety officer was not familiar with all of the devices listed for a particular organization. This was particularly a problem with large organizations such as universities, with diffuse organization structures and rapid turnover of personnel.

Some potential respondents were impossible to locate, either because they were out of business, had left to forwarding address, or for other reasons could not be traced. A number of responses were received in which the respondent reported that the firm had been sold, entered bankruptcy, or had closed, and no remaining personnel were familiar with the disposition of the device.

A small group of potential respondents refused to cooperate. In general, however, response problems occurred due to lack of information rather than unwillingness to supply information.

1.5 Scope

This report discusses the following:

- i. Survey Design and Questionnaires (Chapter 2.0)
- ii. Survey Implementation (Chapter 3.0)
- iii. Findings of the study with respect to the general licensees authorized in 10 CFR §31.5 (Chapter 4.0); and
- iv. Overall general conclusions and recommendations (Chapter 5.0).

2.0 SURVEY DESIGN AND QUESTIONNAIRES

This section describes the initial design of the survey and the design and contents of the survey questionnaires. It first discusses the national population of general licensees, the survey frame (i.e., those members of the population from which the survey sample was chosen), and the survey sample. Next it discusses the development and contents of the survey questionnaires, including how the questions were chosen and how the questionnaires were tailored to reflect the different characteristics of the different types of general licensees included in the sample.

2.1 Survey Population, Frame, and Sample


The Office of Nuclear Materials Safety and Safeguards is responsible for administering the NRC's programs concerning the possession and use of nuclear materials. Licenses for possession and use of byproduct material constitute approximately 5,000 of the approximately 8,000 materials licenses currently administered by NRC (the 29 Agreement States administer another 16,000 licenses).¹ As 10 CFR §30.31 provides, the NRC's licenses for byproduct materials can take one of two forms: (1) the license can be a specific license issued to individually-named persons or organizations upon applications filed pursuant to the regulations in Part 30 and Parts 32 through 35 of 10 CFR, or (2) the license can be a general license, effective without the filing of an application with the Commission or the issuance of licensing documents to a particular person. As the 1989 NRC Annual Report explains:

General licensees include individuals or organizations that become licensees (without contacting the NRC) when they receive a byproduct source or a device containing a byproduct source from a specific licensee. These include certain measuring, gauging, illuminating, and controlling devices containing byproduct material with radioactivity ranging from microcuries to several curies. . . . General licensees are expected to be able to use the devices safely by following simple instructions -- without having radiological safety training or experience -- because safety is built into the devices.

Altogether, the population of general licensees numbers about 30,000 in non-Agreement States, using about 400,000 devices, with about 60,000 general licensees in Agreement States with about 800,000 devices.

NRC currently has issued about 17 general licenses. Seven pertain to the transportation, storage, and installation of radioactive material; ten allow the use of particular devices or materials. The survey frame was comprised only of general licensees under the general license established by 10 CFR §31.5.

¹ United States Nuclear Regulatory Commission, 1989 Annual Report, NUREG-1145, Vol. 6, July, 1990, p. 74.



Section 31.5 governs:

Any byproduct material contained in devices designed and manufactured for the purpose of detecting, measuring, gauging, or controlling thickness, density, level, interface location, radiation, leakage, or qualitative or quantitative chemical composition, or for producing light or an ionized atmosphere.

The identity of such general licensees is reported to NRC by the specifically licensed manufacturers of the devices containing the byproduct material. Manufacturers specifically licensed by NRC provide quarterly reports to NRC; Agreement State manufacturers or distributors must notify NRC when they sell generally licensed devices in non-Agreement States. The sampling frame for the survey was comprised of quarterly reports submitted by the manufacturers listed in Exhibits 2-1, 2-2 and 2-3, who produce a large proportion of all such devices, for the periods indicated in the exhibits.

The potential survey respondents were selected by NRC from the quarterly reports submitted by specific licensees. The quarterly reports are provided to NRC as proprietary business information under a claim of confidentiality. Therefore, the names of potential survey respondents chosen from those reports were treated as confidential information, and maintained under conditions of security, both in hard copy form and in computer records. Names of individual respondents are not included in this report.

Separate samples were selected for each of the three major categories of respondents, with the goal of identifying samples of the following sizes:

Gauges	1500 respondents
Analytic Devices	1000 respondents
Self-powered lights	500 respondents.

Initial review of the selected potential respondents revealed several factors that required a supplemental sample to be selected. These factors included:

- Duplication of firms;
- Missing addresses that could not be obtained in a timely manner;
- Other factors indicating that a potential respondent would not be appropriate, including the following:
 - (1) The possibility that the devices were installed on aircraft, and therefore correctly reported to NRC under section 31.xx of 10 CFR. In addition, because such devices would present particular problems of tracking and identification, they were removed from the sample.

EXHIBIT 2-1
SPECIFIC LICENSEES SUPPLYING DEVICES TO
GAUGE RESPONDENTS

<u>Firm Name</u>	<u>Dates Covered</u>
Combustion Engineering	1989-1990
Ohmart	1985, 1989-1990
Rosemount	1989
Industrial Dynamics	1987-1989
Sentrol	1985-1989
MPSI Technologies	1985-1988
Molins Richmond, Inc.	1989
Delphi Instruments	1988-1990
AEG-Telefunken	1985
RMD, Inc.	1989
Heuft	1989-1990
Honeywell	1988
AccuRay	1985-1988
Texas Nuclear	1985-1988
Kay Ray	1985-1989
Berthold Systems	1989-1990
Intergrated Industrial Systems	1989
Harrel	1989
United Technologies	1988-1989
Ahlstrom	1988-1989
CMI International	1987-1990
Ronan	1989
NDC Systems	1987-1989
Fife	1985-1986
Dosimeter Corporation	1985-1990
Victoreen	1989
Fairchild Weston	1985-1988
Data Measurement Corporation	1988-1989
AEonic Systems	1988-1990
Gamma Instruments	1988-1989
Peco Controls	1988-1990
Loral Control Systems	1989
Measurex	1985-1986
Barber Colman	1986-1989
TCI	1988-1990
UPA Technology	1985

EXHIBIT 2-2
SPECIFIC LICENSEES SUPPLYING DEVICES TO
ANALYTIC DEVICE RESPONDENTS

<u>Firm Name</u>	<u>Dates Covered</u>
NRD	1989
Chrompack	1988
Fischer Technology	1988-1989
Beckman	1989-1990
Perkin-Elmer	1987-1990
Varian	1988-1989
Packard Instrument	1989-1990
Hewlett-Packard	1989-1990
TSI	1989
Veeco	1988
Panametrics	1988-1990
PGT	1988-1989
Shimadzu	1989
CSI	1985-1989
ASOMA Instruments	1989
Valco Instruments	1988-1989
S-Cubed	1988-1989
ICN Micromedic Systems	1988-1989
Tracor Instruments	1988-1989
3M Electrical Specialties Division	1988-1990

EXHIBIT 2-3
SPECIFIC LICENSEES SUPPLYING DEVICES TO
SELF-POWERED LIGHTS LICENSEES

<u>Firm Name</u>	<u>Date Covered</u>
Stusser Electric	1989
Self-Powered Lighting	1989
JMJ Sales	1986-1989
Electric Distributors	1989
Safety Light	1988-1989
Isolite	1989
SRB Technologies	1989
Shielded Source	1989

(2) The possibility that the devices were installed on military ships, or shipped to military installations for other purposes.

- Indications that the number of firms initially identified was not sufficient to reach the goals for the size of the samples, even if other factors had not intervened.

A significant difference between the quarterly reports for gauges and analytic devices and the reports for self-powered lights was the number of such devices that could be included in a shipment. In most cases, gauges and analytic devices were described individually on the reports, with separate serial numbers or other identifiers. In contrast, the quarterly reports for self-powered lights sometimes described shipments of several lights of the same type, without other identifying characteristics. A typical report of this type might identify the firm to which the lights were shipped and the date shipped, give a model number, and then indicate that 24 such lights were shipped on the same day.

In some cases, potential respondents would have been required to provide information about large numbers of self-powered lights (For one firm, over one hundred such responses would have been required; others would have been required to make several dozen responses.) Because such a heavy burden would have significantly affected the likelihood of response, an additional sample of potential respondents for self-powered lights was chosen, and some of the previously chosen potential respondents with the largest numbers of devices were replaced by members of the supplemental sample with significantly fewer devices.

The supplemental samples were selected by ICF staff from quarterly reports provided by NRC. Both quarterly reports previously used and additional quarterly reports were used. In some cases, quarterly reports from earlier years (e.g., 1985-1988) were used to supplement the quarterly reports initially used by NRC (which dated primarily from 1989).

A number of firms located in Agreement States were mistakenly included in the supplemental sample. At the request of NRC, these potential respondents were dropped as soon as they were identified. Because almost all firms included on the quarterly reports were already part of the sample, however, the firms deleted from the sample could not be replaced.

The final sample sizes, following both supplementations and deletions, were as follows:

Gauges	1178 respondents
Analytic Devices	872 respondents
Self-powered lights	483 respondents.

The number of devices attributable to these respondents, however,

equalled or exceeded the goals for the survey.

Data concerning each respondent were entered into one of three databases, depending on the respondent type. Data included firm name and address; type of device, including serial number and amount and type of radionuclide, if available; and name and address of the vendor of the device. Data entry began in early May 1990. In addition, data were entered into an AlmPlus database for use in addressing and managing the survey mailings.

2.2 Survey Questionnaires

Survey questionnaires were designed to include both firm-specific and device-specific information, which was to be verified by the respondent, and questions to be answered by the respondent.


The questions included in the survey were of two types. Questions were based on the set of questions used by NRC in its study conducted between 1984 and 1986. In most cases, those questions could be answered by a "yes" or "no" response. When additional information, of a supplementary nature, was necessary, the questionnaire was designed to include space and instructions for narrative responses. Finally, "skip patterns" were included to ensure that related sets of questions were addressed properly. Copies of the questionnaires for gauges and analytic devices, and for self-powered lights are included in Appendix A.

For each respondent, a single firm-specific set of questions was prepared, with preprinted information on the firm name, address, contact person, phone number, and principal business. This section of the questionnaire also contained questions about knowledge of the general license conditions. Each respondent was assigned a separate respondent identification number, which was printed on the questionnaire forms.

For respondents with gauges and/or analytic devices, a second device-specific set of questions was prepared, with preprinted information on the device, for each device indicated on the quarterly reports to be in the possession of the general licensee. Thus, a licensee with a single gauge would receive one device-specific questionnaire; a licensee with three gauges would receive three such questionnaires. When possible, each device was identified by type and by serial number. When such identifiers could not be obtained from the quarterly reports, separate devices were identified by number (i.e., one of one, one of two, etc.).

For respondents with self-powered lights, a different type of questionnaire was prepared. Such respondents received only one device-specific questionnaire, irrespective of the number of lights they had been shipped of the same type. When a respondent had received more than one type of self-powered light, a different device-specific questionnaire was prepared for each type. Each device-specific questionnaire also included an entry giving the number of devices shipped and the date of the shipment.

In addition to the survey questionnaires, a cover letter was prepared for inclusion in the survey packages. Following review by NRC, the letter was printed on NRC letterhead. The cover letter explained the purpose of the survey, gave general instructions for responding to the survey, explained the confidential nature of responses, and encouraged all respondents to return the survey. A copy of the cover memorandum is included in Appendix B.



3.0 SURVEY IMPLEMENTATION

This section describes the steps taken to implement the survey of general licensees. It describes how the survey questionnaires were prepared and the contents of the packages that were sent to potential respondents; how responses and nonresponses were tracked; procedures for data collection and maintenance; and followup activities.

3.1 Preparation and Mailing of Survey Questionnaires

Letters containing questionnaires with preprinted firm-specific and device-specific information for each respondent, a cover letter, a return envelope, and a copy of 10 CFR §31.5 were prepared under secure conditions in a survey preparation room. Firm and device-specific information was obtained from a master database prepared from the contents of the quarterly reports. Address information was duplicated in an AlmPlus database, and address labels were printed out from that database. Address labels were printed in quadruplicate. One was attached to the envelope for each respondent, one to certified mail forms attached to the envelope, one to certified mail return receipts ("green cards"), and one to the appropriate entry in the respondent tracking notebook. Following preparation of each letter, it was reviewed by a different member of the survey staff as a check for the following:


- to ensure that the respondent name and address matched the mailing labels;
- to ensure that the firm identification number matched on all device-specific questionnaires;
- to ensure that proper number of device-specific questionnaires were included; and
- to ensure that the package was complete.

An entry was made for each questionnaire package in a questionnaire tracking notebook, organized by respondent number, at the time the package was prepared. A copy of the appropriate mailing label was placed in the notebook under the correct respondent identification number. The date the package was checked, the identity of the checker, and the date the package was mailed were all entered as those actions took place.

Mailing of questionnaires was done in three groups, by type of respondent, beginning in mid-June. The first group of surveys to be mailed was self-powered lights, followed by gauges and then by analytic devices. The group approach was adopted to enable the printing of the survey forms and the preparation of the questionnaires and survey packages to be spaced more evenly over time.

3.2 Tracking Procedures

Survey questionnaires were tracked in respondent notebooks and in the AlmPlus databases. The notebooks and databases were retained in the survey



preparation room. As certified mail receipts were received, the fact that the receipt had been sent back was entered into the respondent notebook and the receipts were filed by survey respondent identification number. This procedure enabled survey staff to verify whether the survey package had been accepted at the address to which it was sent, and by whom.

A number of survey packages were returned as undeliverable. That fact was entered into the AlmPlus survey tracking database. Special codes were used to indicate the reason for return (e.g., moved, left no forwarding address; no such address; addressee unknown, etc.) or action taken (e.g., address changed to new address and questionnaire remailed).

3.3 Data Collection and Maintenance Procedures

As completed survey responses were received, they were opened in the survey preparation room and logged in to the appropriate respondent notebook at the entry corresponding to that survey package, as indicated by the respondent identification number. Periodically thereafter, survey responses were entered in the appropriate survey results database. In addition, an entry indicating that the survey questionnaire had been returned was made in the AlmPlus survey tracking database. All returned survey questionnaires were filed by respondent number and retained in the survey preparation room.

The survey cover letter and the return envelopes both requested respondents to return completed surveys to the address of the survey contractor, in Vienna, Virginia. In some cases, however, questionnaires that could not be delivered to the respondent and/or completed questionnaires were mailed instead to the NRC Headquarters building in Rockville, Maryland. In such cases, the questionnaires were periodically transferred by messenger to ICF.

All survey response data were double key entered into the appropriate databases. The databases were maintained on computers located in the secure survey preparation room and were backed up on a daily basis. A separate copy of the databases was also maintained on a Bernoulli disk.

Certain respondents included letters with the questionnaire, or substituted letters for the questionnaires. In such cases, the letter was reviewed by the preparation room staff and either treated as a response or set aside for special attention. All letters were filed by respondent identification number and retained in the preparation room.

3.4 Followup Activities

In addition to the initial mailing, a number of followup activities were undertaken to obtain the highest possible response rate and, whenever possible, to obtain information concerning the ultimate disposition of the device even if it had been transferred or disposed of by the original general licensee. These followup activities, which are described in the following subsections, included a wave II mailing, use of a toll-free telephone hotline

to respond to questions by respondents, telephone followup initiated by ICF, and on-site visits.

Wave II Mailing.

Approximately 40-60 days after completion of the first wave mailing, a second mailing was made to all respondents who had not returned a completed questionnaire. This mailing included a slightly stronger cover letter and copies of all other documents (i.e., a survey form, return envelope, and copy of relevant portions of 10 CFR Part 31) that had been included in the first-wave mailing. This mailing was also sent by certified mail, return receipt requested. The addresses used for the Wave II mailing were corrected, when possible, with the results of the Wave I mailing and from other sources, described in this section.

Hotline Responses to Inquiries

Prior to the Wave I mailing, ICF established a dedicated telephone "hot line" and provided the number for that hotline in the cover memorandum that accompanied all survey questionnaires. This hotline was given a toll free 800 number. Software converted the single hotline into four incoming lines of the ICF telephone system, each of which was attached to a message storage and retrieval unit. ICF assigned staff to clear the "voice message boxes" and respond to incoming telephone calls on a daily basis. ICF sought to respond to telephone calls within one day, although this was not always accomplished, depending on the volume of incoming calls and the ability to call back and reach the respondent. Frequently, several calls back and forth would be placed before the respondent's questions could be answered. A phone log was kept of all calls placed to the hotline and the disposition of the inquiry. Periodic summaries were prepared of questions asked and answered most frequently. Those questions are described in Exhibit 3-1, which provides a sample of questions asked by potential respondents for the tritium exit sign survey. Similar questions were asked by respondents for the other two surveys.

Telephone Followup

ICF performed three types of followup activities by telephone in addition to the hotline service. First, a number of outstanding addresses were checked by telephone; second, as the period of time for responses came to a close, ICF contacted every outstanding potential respondent by telephone and sought to elicit a response from them; finally, in a number of cases, new





EXHIBIT 3-1
TYPICAL QUESTIONS ASKED BY
EXIT SIGN RESPONDENTS

1. We are a distributor, and do not keep signs in stock but ship them immediately to the customer, are we a general licensee and are we required to fill out the survey?
 2. We can't locate the sign--it is somewhere in the city--are we required to track it down and obtain the information?
 3. The proper contact person is out of town, or information is difficult to find--can we have an extension of time to prepare the questionnaire?
 4. What is a copy of the General License?
 5. We just took over this facility, and the signs were installed by a contractor; are we required to maintain records and supply information about them?
 6. Is the label clearly visible if it is 1 1/4" wide?
 7. The listed contact person is no longer an employee of the facility. Should he/she still fill out the survey?
 8. We have a broad-scope license, are we still required to fill out the survey?
 9. No serial number is given, and we can't locate which sign(s) we should be providing information about.
 10. The questionnaire must have been sent to us by mistake, we have no records of ever having received any signs.
 11. We sent the sign back for disposal; therefore, we no longer have it, do we still have to fill out the questionnaire?
 12. Is this survey mandatory?
- 

survey questionnaires were distributed to respondents and returned by them using telephone facsimile equipment. Each of these activities is described below.

Address checking was initiated when a survey package was returned by the postal service as undeliverable because the addressee was not at the indicated address or because of a problem with the address. Calls were made to the telephone information service for that city (in some cases for several different area codes associated with the general location) and inquiries were made about the telephone number of the potential respondent. If the respondent was listed, an inquiry was also made about the address. In some cases, address information was obtained from the telephone information service. If address information was not available from that source, the listed number was called and if the firm was reached new address information was obtained from the contact.

Approximately six weeks before the scheduled end of the survey, a list of all outstanding potential respondents was generated and a comprehensive telephone campaign was initiated to contact each of the firms listed. A protocol was developed to ensure that the person best suited to respond to the questionnaire was eventually contacted. Thus, once the firm itself was reached, the following persons were requested, in order:

- Radiation Safety Officer. In general, if a radiation safety officer had been designated by the firm, they were knowledgeable concerning the devices possessed by that firm. In some cases, however, the radiation safety officer had not received the survey questionnaire because it had gone to another person in the firm and had not been passed on properly to the RSO. In other cases, the RSO confirmed that the form had been received but that for some reason it had not yet been returned.
- Personnel Officer. If no RSO was available, most firms required all information about personnel and their duties to be obtained from the personnel office.
- Safety Officer. If no RSO was available, the next person contacted, through the personnel office, was usually the safety officer. Safety officers frequently were aware of devices and responsible for preparing survey responses.
- Facilities/Maintenance Office. If no RSO or SO was available, information about devices could sometimes be obtained from the facilities/maintenance office. In particular, the location of self-powered exit signs was frequently something that those personnel were familiar with or could determine.
- Management. For small firms, cases in which other personnel could not be contacted or were uncooperative, or when referred by other personnel, contacts were sometimes made with firm management.


Their responses was sometimes that they were personally not familiar with all the facts concerning the devices, but that they would determine the appropriate member of their staff to respond to the questionnaire.

During the course of the comprehensive followup, information from the same firm concerning responsibilities for preparing a survey response and whether or not such a response had been prepared sometimes varied significantly depending upon who was contacted.

The telephone followup frequently contacted a person who would ask for a survey questionnaire to be resent, stating that the initial form had been lost or discarded. On several occasions, respondents asked that the questionnaire be sent by telephone facsimile, and promised to respond immediately back by facsimile. This approach was extremely useful; however, it required that new survey questionnaires be printed very quickly so that they could be sent in a timely manner. Although most were sent the same day, some required additional time to prepare.

On-Site Followup

A limited number of site visits were performed when survey responses were not received from a significant number of respondents in a particular geographical area. These site visits were arranged in advance, and in the course of making such arrangements a number of additional survey responses were generated. Survey teams made from 4 to 6 visits per day. In some circumstances even with advance planning the necessary site representatives were not present and the device (particularly exit signs) could not be located by plant personnel. In other cases, filled out survey forms were obtained -- one of which had been delayed by review by the plant legal staff for about four months. In general, site visits added marginal response rates but were less productive than the other follow up activities described above.



4.0 RESULTS

This section presents the results obtained from the mail portion of the survey of general licensees. It first provides the response rates, by category of licensee. It then details the findings that can be derived from the survey responses. Supplementary data are presented in Appendices to this report.

4.1 Response Rates by Category of Licensee and by NRC Region

The response rates for all categories of general licensees included in the survey were at or above the rates normally obtained by mail surveys.¹ Exhibit 4-1 presents the response rates, by category of device and by Region, for each of the three broad types of devices included in the survey and each of the five NRC regions. As Exhibit 4-1 indicates, the response rates nationwide for both analytic device licensees and gauge licensees currently exceed 90 percent, as do the response rates by Region for almost all Regions. Response rates nationwide for self-powered light licensees are somewhat lower, at about 80 percent in total and for almost all Regions.²

In addition to the formal questionnaire responses received, a number of firms responded by letter. Approximately 40 letters must be added to the totals presented in Exhibit 4-1. Finally, followup activities are likely, if response rates continue at about their current rate, to elicit at least 100 additional responses.

Several hypotheses can be advanced for the high response rates. They include:

- A strong tradition of responsiveness on the part of NRC licensees to Commission and Staff inquiries;

¹ D.A. Dillman, Mail and Telephone Surveys, Wiley-Interscience, 1979, p. 51 suggests that response rates for mail surveys are usually lower than for surveys conducted by telephone or personal interview, ranging from 60 to 75 percent for surveys of the general public using lengthy questionnaires to 85 percent or higher for homogenous samples.

² Additional analysis is necessary to determine the number of devices that are reflected in this response rate. In some cases, the absence of only one respondent can significantly reduce the number of devices being reported upon. (For example, several potential respondents for exit signs submitted letters or survey questionnaires in which they stated that they were electrical supply companies, and therefore had transshipped all of the numerous signs delivered to them. In another case, a potential respondent advised that the plant to which 58 analytic devices had been shipped was operated under contract to the US Department of Energy, and therefore the questionnaires were not completed.)




EXHIBIT 4-1
 INTERIM SURVEY RESPONSE RATES
 BY CATEGORY OF GENERAL LICENSEE AND BY NRC REGION

Analytic	Region 1	Region 2	Region 3	Region 4	Region 5	Total
Nonagreement						
Received	368	54	306	47	10	785
Percent	89.10%	94.74%	90.27%	90.38%	90.91%	90.02%
Total	413	57	339	52	11	872
Gauge	Region 1	Region 2	Region 3	Region 4	Region 5	Total
Nonagreement						
Received	373	111	505	74	5	1068
Percent	90.31%	90.24%	90.66%	93.67%	83.33%	90.66%
Total	413	123	557	79	6	1178
Tritium	Region 1	Region 2	Region 3	Region 4	Region 5	Total
Nonagreement						
Received	140	32	163	18	24	377
Percent	77.78%	80.00%	79.90%	81.82%	64.86%	78.05%
Total	180	40	204	22	37	483
						12/08/90

- Statements in the questionnaire cover memorandum and in followup telephone calls that nonresponse could result in an on-site inspection of the device;
- Intensive mail and telephone followup to the initial survey mailing;
- Clear instructions and questionnaires that made response relatively easy;
- Availability of a telephone hotline, so that respondents' questions could be answered in a timely manner;
- Assistance provided to general licensees by vendors of the devices; and
- A relatively long period of time for responses and followup.

Additional research would be necessary to determine if the response rates are attributable to these or other factors.

Part IIA of each questionnaire requested the survey respondents to supply information concerning the principal business conducted at the facility. For analytic device and gauge respondents, 13 specified alternatives and an "other" category were provided; self-powered light respondents were given 9 alternatives and an "other" category. The descriptions of principal business activities are summarized in Exhibits 4-2, 4-3, and 4-4.




EXHIBIT 4-2
 PRINCIPAL BUSINESS ACTIVITIES SPECIFIED BY
 ANALYTIC DEVICE RESPONDENTS

<u>Type of Activity</u>	<u>Number of Respondents</u>	<u>Percent</u>
Agriculture	3	0.4
Canning/Bottling	2	0.3
Chemical	35	5.1
Food	15	2.2
Foundry	1	0.1
Laboratories	274	39.7
Mining	1	0.1
Petroleum	17	2.5
Power-Coal Feed	2	0.3
Pulp and Paper	2	0.3
State/Local Government	17	2.5
Steel	2	0.3
Waste Treatment	14	2.0
Other*	309	44.2

* The business activities in the "other" category are currently being tabulated. They include aerospace, electrical products, printed circuit board, instrumentation, medical electrical equipment, automotive parts, and pharmaceutical manufacturing.

EXHIBIT 4-3
PRINCIPAL BUSINESS ACTIVITIES SPECIFIED BY
GAUGE RESPONDENTS

<u>Type of Activity</u>	<u>Number of Respondents</u>	<u>Percent</u>
Agriculture	0	0.0
Canning/Bottling	39	4.6
Chemical	74	8.7
Food	41	4.8
Foundry	1	0.1
Laboratories	18	2.1
Mining	64	7.5
Petroleum	41	4.8
Power-Coal Feed	21	2.5
Pulp and Paper	138	16.2
State/Local Government	3	0.4
Steel	73	8.6
Waste Treatment	4	0.5
Other*	333	39.2

* The business activities in the "other" category are currently being tabulated.

EXHIBIT 4-4
PRINCIPAL BUSINESS ACTIVITIES SPECIFIED BY
SELF-POWERED LIGHT RESPONDENTS

<u>Type of Activity</u>	<u>Number of Respondents</u>	<u>Percent</u>
Aviation	8	0.3
Entertainment	6	1.8
Food and Lodging	21	6.4
Hospital	13	4.0
Laboratory	5	1.5
Manufacturing*	66	20.2
Office Building	24	7.3
School	38	11.6
State/Local Government	9	2.8
Other*	136	41.6

* The business activities in the "manufacturing" and "other" category are currently being tabulated.

4.2 Survey Findings: Awareness of Regulatory Requirements and Compliance with 10 CFR §31.5

This section discusses the survey results for each of the three categories of general licensees included in the survey. The questionnaires were designed to elicit information that paralleled the basic responsibilities and requirements of general licensees, and therefore included questions involving the following:

- Awareness of regulations and accountability. Under 10 CFR §31.5(a) a general license is issued to commercial, educational, and medical institutions, to individuals in the conduct of their business, and to federal, state or local governments to acquire, receive, possess, use or transfer byproduct material contained in the covered devices. The specific licensee who manufactures and distributes the device must supply a copy of the general license to each person to whom the device containing byproduct material is transferred. In practice, the pertinent sections of 10 CFR Part 31 are transferred by specific licensees and constitute notice to the general licensees of the general license terms and conditions. The survey questionnaire thus inquired whether the respondents had a copy of the general license, and whether the general licensee had specified a company representative who was responsible for knowing the general license conditions and ensuring compliance with them.
- Compliance with testing and maintenance requirements. Part 31 requires leak testing for certain categories of devices, depending on the amount and type of byproduct material that they contain. For those devices that must be leak tested, Part 31 also contains specific requirements concerning the recording of test results and retention of those records. In addition, Part 31 requires servicing of on-off mechanisms and retention of records of servicing. Finally, any events involving failure of shielding, on-off mechanisms or indicators, or the release of radioactive material must be reported within 30 days to NMSS.
- Compliance with transfer and disposal requirements. Under 10 CFR Part 32, firms with a specific distribution license are required to report all transfers quarterly within 30 days after the end of the quarter to NMSS, and must maintain the transfer records for five years. The general licensees to whom the transfer occurs are prohibited by 10 CFR §31.5(c)(6) from abandoning the device, by §31.5(c)(7) from exporting the device, except in accordance with specified procedures, by §31.5(c)(8) from transferring the device, except to a person holding a specific license and with notification to NMSS within 30 days, and by §31.5(c)(9) from transferring the device to another general licensee, except under certain specified conditions. Finally, Part 31 contains specific

requirements concerning receipt and transfer records and their retention, and 10 CFR Part 20 includes requirements for reporting of theft or loss of devices as well as reporting of exposures or incidents involving devices.

The survey results are reported in subsections 4.2.1, 4.2.2, and 4.2.3 for general licensees for analytic devices, gauges, and self-powered lights respectively. The responses of general licensees are provided in the following subsections on a question-by-question basis, unless otherwise noted. In general, questions allowed either a yes or no answer, and did not include an opportunity for the respondent to indicate "don't know" or a similar equivocal response. Responses are reported as yes or no, and an additional entry (DNR for "did not respond") is included to account for survey forms returned with the relevant question left blank. In addition, the tabulated results do not reflect inquiries made by respondents over the telephone hotline, or inquiries made to NRC directly or to specific licensees. In some cases, therefore, tabulated responses reflect the results of inquiries made by the respondents before completing the questionnaire, and may indicate a higher level of knowledge of regulatory requirements than the respondent actually possessed before receiving and responding to the survey. When hotline records indicate a high level of uncertainty on the part of respondents, those results are noted.

4.2.1 General Licensees for Analytic Devices

The results reported below are based on two categories of responses -- firm-specific responses and device-specific responses. Overall, 783 analytic device respondents had returned forms at the time the tabulations were calculated. Firm-specific responses may not total 783, however, because in some cases the form may have been returned with one or more questions unanswered. In addition, 783 separate firms are not represented by the responses. A smaller number of firms is represented, since in some cases different divisions or locations of the same firm were sent separate survey questionnaires.

The results pertain to more than 785 devices, since in a number of cases the same respondent possessed more than one device at the same site and was sent more than one device-specific questionnaire. The number of devices being reported upon varies from question to question, but is about 950 for most questions. Totals are provided for yes and no answers for each question, and the percentages are calculated from those answers rather than the total for the question including DNR.

Knowledge of General License Requirements

General licensees for analytic devices indicated a relatively high degree of uncertainty about whether they possessed a copy of the General License. This question was included to determine if general licensees were aware that the regulations themselves set the terms and conditions of the license. A copy of 10 CFR Part 31 was included in each questionnaire package.

A significant number of hotline calls were received asking about this question, and the hotline response explained the situation to callers (see Appendix C). Even so, about 30 percent of the respondents indicated they did not have a copy of the general license.

Question II.B Does your firm have a copy of the General License?

Yes	471	70%
No	<u>202</u>	30%
	673	
DNR	20	

Question II.C Is there a company representative who knows the General License Conditions?

Yes	601	88.1%
No	<u>81</u>	11.9%
	682	
DNR	11	

Question II.D Is that person responsible for ensuring compliance with the General License conditions?

Yes	593	90.0%
No	<u>66</u>	10.0%
	659	
DNR	34	

General licensees for analytic devices did not report difficulty in finding authorized recipients for storage or disposal of devices. Only 6 respondents (or less than one percent of respondents) reported difficulty in finding an authorized recipient to purchase devices; 11 (about 1.6 percent) reported difficulty in finding a recipient with whom to dispose the device; and only three reported difficulty in finding someone with whom to store unwanted devices.

Only a few respondents reporting holding either devices or sealed sources in secured storage (Questions F and G) as shown in Exhibit 4-6:



EXHIBIT 4-6
DEVICES AND SOURCES REPORTED AS IN STORAGE

Number Held in Secured Storage	Respondents With Devices in Storage		Respondents With Sources in Storage	
	Number	Percent	Number	Percent
0	644	92.9	648	93.5
1	26	3.8	24	3.5
2	14	2.0	12	1.7
3	3	0.4	1	0.1
5	3	0.4	1	0.1
8	--	--	2	0.3
9	1	0.1	--	--
11	--	--	1	0.1
12	1	0.1	1	0.1
13	1	0.1	--	--
19	--	--	2	0.3
40	--	--	1	0.1

Installation, Current Condition, Inspection, and Maintenance

A number of survey questions secured information about the installation of the device, the current condition of the device, inspection and maintenance practices, and recordkeeping.

About a third of the devices were reported as having been installed by the general licensees themselves (the number of responses is greater than the number of respondents because some respondents reported on more than one device).

Question V.D	Did your firm install this device?	
Yes	337	35.3%
No	<u>619</u>	64.7%
	956	
DNR	23	

A large number of respondents also reported that the Specific Licensee from whom they had obtained the device had installed the device at their site:

Question V.D1 Was the Specific Licensee identified in Part III the same organization that installed this device at your site?

Yes	546	94.5%
No	<u>32</u>	5.5%
	578	
DNR	401	

Finally, a small number of devices were apparently installed by neither the general licensee nor the specific licensee. Respondents indicated about seven other firms that had installed devices (Questions V.D.2 to D.10).

Under Part 31, devices containing krypton-85 and tritium (gas) sources are not required to be leak tested; devices containing gamma/beta sources are required to be tested only when the source is greater than 100 uci and devices containing alpha sources are required to be tested only when the source is greater than 10 uci. The most numerous types of sources are listed in Exhibit 4-7.

EXHIBIT 4-7
SOURCE TYPES MOST FREQUENTLY IDENTIFIED

Source type	Number	Percent
Ni-63	387	39.5
Cs-137	154	15.7
Po-210	93	9.5
Fe-55	52	5.3

Relatively few devices were reported containing krypton-85 (17 devices, 1.7%) or tritium (30 devices, 3.1%). The questions pertaining to initial radiation testing and leak testing outlined the categories of devices exempt from testing, and instructed respondents to skip the questions if they possessed such devices. About 180 responses were received in which these questions were not answered.

Exactly half of the respondents, however, reported having a copy of the initial radiation survey.

Question V.E Does your firm have a copy of the initial radiation survey performed at the time of installation?

Yes	395	50%
No	<u>395</u>	50%
	790	
DNR	189	

Slightly less than half reported that leak tests had been performed on the device following its receipt:

Question V.F Since your firm received this device, have leak tests been performed?

Yes	396	49.7%
No	<u>400</u>	50.3%
	796	
DNR	183	

Out of the 381 dates reported by respondents as the date of the latest leak test (Question V.F.1), about 16% (61 tests) fell before the beginning of 1990. Forty percent of the tests were performed after June 1, 1990, and 25% were performed after July 1.

A slightly smaller number of firms reported having a copy of the most recent leak test than reported having conducted such tests:

Question V.F.2 Does your firm have copies of the record of the latest test?

Yes	343	87.9%
No	<u>47</u>	12.1%
	390	
DNR	589	

Finally, 395 of the 396 firms reporting tests provided data on who had conducted the leak test. About two-thirds of such tests were not conducted by the general licensee.

Question V.F.3 Did your firm conduct the latest test?

Yes	142	35.9%
No	<u>253</u>	64.1%
	395	
DNR	584	

Section 31.5(c)(2) requires servicing of the on-off mechanism and indicator of a device with such mechanism every six months, or as otherwise specified. Records of the test results must be maintained for one year.

About one-third of the analytic devices covered by the survey had an on-off mechanism.

Question V.G Does the device have an on-off mechanism?

Yes	286	34.9%
No	<u>533</u>	65.1%
	819	
DNR	160	

Less than half of those analytic devices with on-off mechanisms, however, had a test of the mechanism performed at any time since the device was received.

Question V.G.1 Since your firm received this device, have on-off mechanism tests been performed?

Yes	109	39.8%
No	<u>165</u>	60.2%
	274	
DNR	705	

The dates of 97 out of the 109 tests were reported (Question V.G2). Of these dates, 18 fell between March 1988 and December 1989, outside the six month period preceding the mailing of the survey. In contrast, about 70 percent of the tests had occurred after February 1990, indicating that those general licensees who were conducting on-off mechanisms testing were doing so in a timely manner.

Relocation, Transfer, and Disposal

Respondents reported relatively few situations in which analytic devices were relocated after installation, although a number of hotline calls were received asking for clarification on the meaning of relocation.

Question V.H. Has this device been relocated (moved within the site and reinstalled) since it was initially installed?

Yes	143	15.5%
No	<u>804</u>	84.5%
	952	
DNR	27	

Almost all of those analytic devices that were relocated were moved and reinstalled by the general licensee in possession of the device.

Question V.H1 Did your firm relocate this device?

Yes	129	85.4%
No	<u>22</u>	14.6%
	151	
DNR	828	

Very few respondents (16) reported the date that the device was last relocated (Question V.H2). All but four reported relocating the device during 1990.

Sixteen of the 22 firms that reported that someone else had relocated a device for them provided information on the name of the company that had moved the device. Almost all of these firms appeared to be specific licensees; in two cases, however, commercial moving and storage firms were listed as having relocated analytic devices.

Even fewer general licensees reported having transferred devices to another company than reported relocations of devices.

Question V.J1 Has your company transferred this device to another firm?

Yes	78	8.1%
No	<u>884</u>	91.9%
	962	
DNR	17	

Out of the 78 devices that were reported transferred, general licensees provided information on the firm receiving the device in approximately 60 cases. Of these, slightly more than half did not appear on the list of specific licensees. Additional analysis is necessary before it can be determined if these transfers met the requirements of 10 CFR §31.5(c)(9), but in several cases the device does not appear to have remained at the same location. One respondent explained that until the survey had been received, he had been unaware of the transfer provisions in Part 31, and had immediately notified NRC of a transfer.

In at least one case, a device initially shipped to a location in the United States was transshipped overseas. The respondent, who replied by letter, did not supply enough information to determine if NRC procedures for such shipment were followed and appropriate reports submitted.

Finally, ten situations were reported in which devices could have been or were damaged.

Question V.I Has your firm experienced any events that could have caused damage or did cause damage to this device?

Yes	10	1.0%
No	<u>957</u>	99.0%
	967	
DNR	12	

A review of the supplementary information on these cases is underway to determine the circumstances of the damage.

4.2.2 General Licensees for Gauges

The results reported below are based on two categories of responses -- firm-specific responses and device-specific responses. Overall, 1068 respondents had returned forms at the time the tabulations were calculated. As with analytic device respondents, firm-specific responses may not total 1068 because in some cases the form may have been returned with one or more questions unanswered. In addition, 1068 separate firms are not represented by the responses. A smaller number of firms is represented, since in some cases different divisions or locations of the same firm were sent separate survey questionnaires.

The results pertain to more than 1068 devices, since in a number of cases the same respondent possessed more than one device at the same site and was sent more than one device-specific questionnaire. The number of devices being reported upon varies from question to question, but is about 1750 for most questions. Totals are provided for yes and no answers for each question, and the percentages are calculated from those answers rather than the total for the question including DNR.

Knowledge of General License Requirements

General licensees for gauges indicated a slightly higher degree of knowledge about whether they possessed a copy of the General License than did analytic device respondents. A copy of 10 CFR Part 31 was included in each questionnaire package. Potential gauge respondents placed a significant number of hotline calls asking about this question, and were told that the regulations constituted the terms and conditions of the general license. Even so, about 22 percent of the respondents indicated they did not have a copy of the general license.

Question II.B Does your firm have a copy of the General License?

Yes	652	78.2%
No	<u>182</u>	21.8%
	834	
DNR	22	

Question II.C Is there a company representative who knows the General License Conditions?

Yes	756	89.7%
No	<u>87</u>	10.3%
	843	
DNR	13	

Question II.D Is that person responsible for ensuring compliance with the General License conditions?

Yes	757	91.4%
No	<u>71</u>	8.6%
	828	
DNR	28	

General licensees for gauges also did not report difficulty in finding authorized recipients for storage or disposal of devices. Only 5 respondents (or about 0.6 percent of respondents) reported difficulty in finding an authorized recipient to purchase devices; 6 (about 0.7 percent) reported difficulty in finding a recipient with whom to dispose the device; and only three reported difficulty in finding someone with whom to store unwanted devices.

Only a few respondents reporting holding either devices or sealed sources in secured storage (Questions F and G) as shown in Exhibit 4-8:

EXHIBIT 4-8
GAUGES AND SOURCES REPORTED AS IN STORAGE

Number Held in Secured Storage	Respondents With Gauges in Storage		Respondents With Sources in Storage	
	Number	Percent	Number	Percent
0	796	93.0	801	93.6
1	28	3.3	21	2.5
2	13	1.5	15	1.8
3	6	0.7	7	0.8
4	2	0.2	3	0.4
5	1	0.1	1	0.1
9	1	0.1	--	--
11	2	0.2	1	0.1
14	2	0.2	--	--
15	1	0.1	1	0.1
16	--	--	1	0.1
20	2	0.2	1	0.1
37	--	--	2	0.2
50	--	--	1	0.1

Installation, Current Condition, Inspection, and Maintenance

A number of survey questions secured information about the installation of the device, the current condition of the device, inspection and maintenance practices, and recordkeeping.

About a third of the devices were reported as having been installed by the general licensees themselves (the number of responses is greater than the number of respondents because some respondents reported on more than one device).

Question V.D Did your firm install this device?

Yes	351	20.7%
No	<u>1342</u>	79.3%
	1693	
DNR	57	

A large number of respondents reported that the Specific Licensee from whom they had obtained the device had also installed the device at their site:

Question V.D1 Was the Specific Licensee identified in Part III the same organization that installed this device at your site?

Yes	8	91.2%
No	<u>1</u>	8.8%
	1279	
DNR	471	

Finally, a significant number of devices were apparently installed by neither the general licensee nor the specific licensee. Respondents indicated about 57 instances in which other firms that had installed devices, and identified at least 35 different firms that had performed installations. (Questions V.D.2 to D.10).

Under Part 31, devices containing krypton-85 and tritium (gas) sources are not required to be leak tested; devices containing gamma/beta sources are required to be tested only when the source is greater than 100 uci and devices containing alpha sources are required to be tested only when the source is greater than 10 nci. The most numerous types of sources are listed in Exhibit 4-9:



EXHIBIT 4-9
SOURCE TYPES MOST FREQUENTLY IDENTIFIED

Source type	Number	Percent
Kr-85	446	25.5
Cs-137	479	27.3
Am-241	451	25.7
Sr-90	146	8.6
Pm-147	89	5.1
H3	16	0.9

The questions pertaining to initial radiation testing and leak testing outlined the categories of devices exempt from testing, and instructed respondents to skip the questions if they possessed such devices. Over 250 responses were received in which these questions were not answered.

In contrast to the respondents for the analytical devices, a high proportion of the gauge respondents reported having a copy of the initial radiation survey.

Question V.E Does your firm have a copy of the initial radiation survey performed at the time of installation?

Yes	1229	83.7%
No	<u>239</u>	16.3%
	1468	
DNR	282	

A high proportion of gauge respondents also reported that leak tests had been performed on the device following its receipt:

Question V.F Since your firm received this device, have leak tests been performed?

Yes	1191	80.3%
No	<u>292</u>	19.7%
	1483	
DNR	267	

Out of the 1146 dates reported by respondents as the date of the latest leak test (Question V.F.1), over 300 (about 28%) fell before the beginning of 1990. About 33 percent of the tests were performed after June 1, 1990, and about 20 percent were performed after July 1.

A slightly smaller number of firms reported having a copy of the most recent leak test than reported having conducted such tests:

Question V.F.2 Does your firm have copies of the record of the latest test?

Yes	1113	95.5%
No	<u>52</u>	4.5%
	1165	
DNR	585	

Finally, 1750 names of testing companies were provided as data on who had conducted the leak test if the general licensee had not done so. About 80 percent of such tests were not conducted by the general licensees.

[REDACTED]

Question V.F.3 Did your firm conduct the latest test?

Yes	232	19.8%
No	<u>942</u>	80.2%
	1174	
DNR	576	

Section 31.5(c)(2) requires servicing of the on-off mechanism and indicator of a device with such mechanism every six months, or as otherwise specified. Records of the test results must be maintained for one year.

Almost 90 percent of the gauges covered by the survey had an on/off mechanism.

Question V.G Does the device have an on-off mechanism?

Yes	1465	87.3%
No	<u>213</u>	12.7%
	1678	
DNR	72	

A high proportion of the gauges with on-off mechanisms had a test of the mechanism performed since the device was received. In one case, however, the respondent added a note explaining that since the device was never turned off no tests were performed on the on-off mechanism (G320/0666).

Question V.G.1 Since your firm received this device, have on-off mechanism tests been performed?

Yes	1252	86.1%
No	<u>202</u>	13.9%
	1454	
DNR	296	

The dates of 1199 out of the 1252 tests were reported (Question V.G2). Of these dates, about 240 fell prior to December 1989, outside the six month period preceding the mailing of the survey. In contrast, about 75 percent of the tests had occurred after February 1990, indicating that those general licensees who were conducting on-off mechanisms testing were doing so in a timely manner.

Relocation, Transfer, and Disposal

Respondents reported relatively few situations in which gauges were relocated after installation, although a number of hotline calls were received asking for clarification on the meaning of relocation.



Question V.H. Has this device been relocated (moved within the site and reinstalled) since it was initially installed?

Yes	128	7.6%
No	<u>1554</u>	92.4%
	1682	
DNR	68	

A high proportion of the gauges that were relocated were moved and reinstalled by the general licensee in possession of the gauge, according to the survey responses. Respondents reported, for example, that the production line where the gauge had been used had been closed and the gauge removed from the line and stored (G305/643), or that a truck-mounted gauge had been removed and transferred to another section of the same company located several miles away (hotline contact).

Question V.H1 Did your firm relocate this device?

Yes	95	75.4%
No	<u>31</u>	24.6%
	126	
DNR	1624	

Few respondents (29) reported the date that the device was last relocated (Question V.H2). Less than half reported relocating the device during 1990.

Information was presented for 29 situations in which a gauge was relocated by a firm other than the general licensee on the identity of the firm that performed the relocation. At least 20 of the firms were specific licensees whose quarterly reports had been used to develop the survey frame; at least two other firms were included on NRC device vendor lists.

A greater number of general licensees reported having transferred gauges to another company than reported relocations of gauges.

Question V.J1 Has your company transferred this device to another firm?

Yes	150	8.7%
No	<u>1578</u>	91.3%
	1728	
DNR	22	

Out of the 150 devices that were reported transferred, general licensees provided information on the firm receiving the device in almost every case. A substantial number of the approximately 70 firms listed did not appear on the list of specific licensees who had been the initial vendors of the gauges. Additional analysis is necessary before it can be determined if these transfers met the requirements of 10 CFR §31.5(c)(9), but in several cases the

gauge does not appear to have remained at the same location. Respondents reported, for example, that after a particular plant had closed the device had been transferred to another company but was currently in storage at the specific licensee's facility (G417/864). In some cases, respondents enclosed copies of reports made to NRC concerning transfers (G461/970).

In at least one situation, the respondent reported that the gauge had been shipped overseas without having been used or even unpackaged in the United States (G049/107).

Finally, 13 situations were reported in which devices could have been or were damaged.

Question V.I Has your firm experienced any events that could have caused damage or did cause damage to this device?

Yes	13	0.8%
No	<u>1701</u>	99.2%
	1714	
DNR	36	

A review of the supplementary information of these cases revealed the following:

- The shutter on a void detection device in a silo was found to be broken in the closed position. The unit was locked in the closed position and the area secured. A new housing was installed, following modification by the manufacturer. About 16 months later the shutter failed again. Following consultation with the manufacturer, the gauge was moved to a position where it experienced less vibration. (G698/1422)
- Electronic components failed due to power surges in the 110 v circuit. Electrical repairs were conducted by the manufacturer. (G613/1268)
- Following a plant fire the manufacturer's service personnel checked the gauge immediately. No source leakage was detected. (G601/1252)

4.2.3 General Licensees for Tritium-Powered Exit Signs

The results reported below are based on two categories of responses -- firm-specific responses and device specific responses. Overall, 377 respondents had returned forms at the time the tabulations were calculated. Firm-specific responses may not total 377, however, because in some cases the form may have been returned with one or more questions unanswered. In addition, 377 separate firms are not represented by the responses. A smaller

number of firms is represented, since in some cases different divisions or locations of the same firm were sent separate survey questionnaires.

The results pertain to more than 377 signs, since in a number of cases the same respondent possessed more than one sign at the same site. In contrast to the questionnaires for analytic devices and gauges, however, a duplicate device-specific questionnaire was not sent for each tritium-powered exit sign held by a respondent. Instead, the respondent was provided information about the number of signs covered by the survey and requested to respond to each question as it related to the sign or signs held. In some cases, when respondents had received shipments of signs in addition to those covered by the survey, respondents indicated that they held additional signs and a number of hotline telephone calls were received from general licensees inquiring whether they should respond for all signs held by them. Other potential respondents did not understand that the questionnaire pertained to more than one sign. The number of signs being reported upon varies from question to question. Totals are provided for yes and no answers for each question, and the percentages are calculated from those answers rather than the total for the question including DNR.

Knowledge of General License Requirements

General licensees for tritium-powered exit signs indicated a relatively high degree of uncertainty about whether they possessed a copy of the General License. This question was included to determine if general licensees were aware that the regulations themselves set the terms and conditions of the license. A copy of 10 CFR Part 31 was included in each questionnaire package. A significant number of hotline calls were received asking about this question, and the hotline response explained the situation to callers (see Appendix C). Even so, almost half of the respondents indicated they did not have a copy of the general license.

Question II.B Does your firm have a copy of the General License?

Yes	163	51.1%
No	<u>156</u>	48.9%
	319	
DNR	9	

Question II.C Is there a company representative who knows the General License Conditions?

Yes	189	59.4%
No	<u>129</u>	40.6%
	318	
DNR	10	

Some survey respondents reported that a company representative was appointed after receipt of the survey (223), or that the receipt of the survey was their first knowledge that the signs were "anything special." (237)(485)

Question II.D Is that person responsible for ensuring compliance with the General License conditions?

Yes	188	68.9%
No	<u>85</u>	31.1%
	273	
DNR	55	

One important result of the survey was the development of information that certain respondents and/or specific licensees had been reporting emergency signs for aircraft under §31.5 rather than §31.7. Steps were taken to ensure that results contained in this report pertain only to §31.5 general licensees.

General licensees for self-powered lights reported that they had no difficulty in finding authorized recipients for storage or disposal of the lights. Only 2 respondents (or less than one percent of respondents) reported difficulty in finding an authorized recipient to purchase lights; similarly, only 2 reported difficulty in finding a recipient with whom to dispose of the lights; and only 1 reported difficulty in finding someone with whom to store unwanted lights. One respondent suggested, however, that it had tried and failed to dispose of a sign in the local landfill, which would not accept it, and had not yet contacted the manufacturer about disposal. (218)

Only a few respondents reporting holding either exit lights in secured storage (Questions F and G) as shown in Exhibit 4-9:

[REDACTED]

EXHIBIT 4-9
EXIT SIGNS REPORTED AS IN STORAGE

Number Held in Secured Storage	Respondents With Signs in Storage	
	Number	Percent
0	309	94.2
1	11	3.4
4	2	0.6
7	1	0.3
12	2	0.6
14	1	0.3
30	1	0.3
46	1	0.3

Installation, Current Condition, Inspection, and Maintenance

A number of survey questions secured information about the installation of the exit signs, the current condition of the signs, inspection and maintenance practices, and recordkeeping.

Over half of the signs were reported as having been installed by the general licensees themselves (the number of responses is greater than the number of respondents because several respondents reported on more than one sign).

Question V.B Did your firm install the exit sign(s)?

Yes	188	60.1%
No	116	37.1%
Some	<u>9</u>	2.9%
	313	
DNR	15	

Very few respondents reported that the Specific Licensee from whom they had obtained the signs had also installed the signs at their site:

Question V.B1 Was the Specific Licensee identified in Part III the same organization that installed the exit sign(s) at your site?

Yes	14	14.7%
No	<u>81</u>	85.3%
	95	
DNR	233	

A significant number of respondents identified themselves as electrical contractors, and explained that they had purchased the signs either for a specific client or had purchased them for resale. In most cases, such respondents could not identify the ultimate recipient of the signs. Respondents also identified non-standard uses of signs that complicated their tracking (e.g., installation on movable partitions (119), use for aircraft escape hatch marker (020), use for commercial display sign, after modification (site visit information).

Under Part 31, devices containing tritium (gas) sources are not required to be leak tested; therefore, the questionnaire did not contain any questions about leak testing. Similarly, testing of on-off mechanisms and indicators is not required for self-powered exit signs, and no questions were asked on that subject. Such signs are required, however, to be clearly labeled and to be installed in such a manner that the label is visible. Almost all respondents reported that the labels were durable, legible, and clearly visible.

Question V.E Do the exit sign(s) have:

E1. A durable label?

Yes	285	99.0%
No	<u>3</u>	1.0%
	288	
DNR	40	

E2. A legible (easily readable) label?

Yes	281	98.3%
No	<u>5</u>	1.7%
	286	
DNR	42	

E3. A clearly visible (easily seen) label?

Yes	262	92.3%
No	<u>22</u>	7.7%
	284	
DNR	44	

These results should be considered in light of a number of calls that were received on the telephone hotline from respondents who noted that the labels were quite small and could be located over doors that placed them ten or more feet over the head of someone trying to read the label. One respondent included a copy of a label indicating that the recommended maximum distance for legibility is 100 feet.(283) Another stated that the signs were not legible upon receipt, but that they had been relabeled by the general licensee (299)

Respondents also reported that a substantial proportion of the signs were checked for defects after receipt by the general licensee.

Question V.A Were the exit sign(s) ever checked for defects after receipt?

Yes	223	70.8%
No	90	28.6%
Some	<u>2</u>	0.6%
	315	
DNR	13	

Relocation, Transfer, and Disposal

Respondents reported several situations in which exit signs were relocated after installation, although a number of hotline calls were received

asking for clarification on the meaning of relocation.

Question V.C Have the exit sign(s) been relocated (moved within the site and reinstalled) since initial installation?

Yes	8	2.6%
No	300	96.8%
Some	<u>2</u>	0.6%
	310	
DNR	18	

One respondent, however, reported having relocated 34 such signs.

Almost all of the general licensees who reported signs that had been relocated also reported relocating the signs themselves.

Question V.H1 Did your firm relocate this device?

Yes	6	54.5%
No	<u>5</u>	45.5%
	11	
DNR	317	

Few general licensees for tritium exit signs reported having transferred signs to another company.

Question V.F Has your company transferred any exit sign(s) to another firm?

Yes	26	8.2%
No	<u>292</u>	91.8%
	318	
DNR	10	

As noted above, a number of survey responses were received from electrical contractors, vendors, and building contractors stating that they had received lights from specific licensees but had passed them on to ultimate users or installed them in structures. In such cases, the number of lights that were transferred was usually not specified; however, two respondents did state that they had transferred 100 and 150 lights respectively. In at least one case, a transfer of signs back to the manufacturer was initiated by the survey. This respondent reported that the survey had requested information on 11 signs; inquiries revealed that 24 had been purchased by plant personnel. Two could not be located, 2 were removed from use and returned, and 20 that had not been used were also returned. (185)

Finally, two situations were reported in which lights could have been or were damaged.

[REDACTED]

Question V.D Has your firm experienced events that could have caused damage or did cause damage to any exit sign(s)?

Yes	2	0.6%
No	<u>314</u>	99.4%
	316	
DNR	12	

A review of other information revealed the following additional situations:

- Signs installed by a previous owner were removed prior to receipt of the survey by a licensed demolition contractor and disposed. (123)
- A forklift struck a sign and damaged the mounting device. The mounting device was replaced and the sign rehung. Geiger counter monitoring was performed on the sign. (076)
- A motel suffered the theft of 18 signs and damage to the casing of 4 others. The incident was reported to the NRC and the damaged signs were returned to the supplier. (390)
- A motel suffered the theft of 2 signs. The incident was reported to the police. (391)

Finally, one respondent reported that all equipment delivered to the location in question had been shipped overseas. (33)

APPENDIX A
SURVEY QUESTIONNAIRES

U.S. Nuclear Regulatory Commission

Survey of General Licensees: Facility Information

Part I contains preprinted information about your firm. Please review the preprinted information in Part I and correct any incorrect information by crossing through the incorrect information and writing in the correction above or to the right of the original information. Please complete any missing information by writing the information in the space provided for that information.

Part II of this survey asks additional information about your firm and your experience dealing with the devices identified in Parts III and IV of this survey. Please answer all questions in Part II as they relate to your firm as a whole. Please fill in or circle the answer as appropriate.

Part I: Respondent Identification *(Please correct any incorrect information)*

A. Company Name _____

B. Company Contact (First/Last Name) _____

C. Contact Title _____

D. Address _____

E. City _____

F. State _____

G. Zip Code _____

H. Phone Number _____

Part II: Additional Respondent Information

Please fill in or circle the answer as appropriate

A. What is the principal business conducted at your facility?

☐ 1 - Agriculture

☐ 6 - Laboratories

☐ 11 - State/Local Gov't

☐ 2 - Can/Bottling

☐ 7 - Mining

☐ 12 - Steel

☐ 3 - Chemical

☐ 8 - Petroleum

☐ 13 - Waste Treatment

☐ 4 - Food

☐ 9 - Power-Coal Feed

☐ 14 - Other *(please specify)* _____

☐ 5 - Foundry

☐ 10 - Pulp and Paper

B. Does your firm have a copy of the General License?

☐ Yes

☐ No

U.S. Nuclear Regulatory Commission

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OMB No. 3150-0152
Expires 10/31/92
ID # _____

Survey of General Licensees: Facility Information (continued)

Part II: Additional Respondent Information (continued)

C. Is there a company representative who knows the General License conditions?	<input type="checkbox"/> Yes	<input type="checkbox"/> No
D. Is that person responsible for ensuring compliance with the General License conditions?	<input type="checkbox"/> Yes	<input type="checkbox"/> No
E. Is your firm having any difficulty finding an authorized recipient to:		
E1. Purchase any devices held under General License you no longer want?	<input type="checkbox"/> Yes	<input type="checkbox"/> No
E2. Dispose of any devices you no longer want?	<input type="checkbox"/> Yes	<input type="checkbox"/> No
E3. Store any devices you no longer want?	<input type="checkbox"/> Yes	<input type="checkbox"/> No

If you answered No to question E1, E2, and E3, skip to question F.

E4. If your firm is having any difficulty finding an authorized recipient, please elaborate by describing who you contacted and describe the nature of the difficulty. Please continue your answer on additional blank pages as needed. Label each blank page with your company name and the ID number at the top of this page.

F. How many devices, held under a General License, are now in secured storage and are not intended for use?	_____
G. How many sealed sources, held under a General License, are now in secured storage and are not intended for use?	_____

U.S. Nuclear Regulatory Commission

Survey of General Licensees: Device Information

Parts III and IV contain preprinted information about a device containing byproduct material that NRC records indicate was delivered to your firm under a General License. Please review the preprinted information in Parts III and IV and correct any incorrect information by crossing through the incorrect information and writing in the correction above or to the right of the original information. Please complete any missing information by writing the information in the space provided for that information.

Part V of this survey refers to the particular device identified in Part IV. Please answer all questions in Part V as they relate to the device identified in Part IV. Please fill in or circle the answer as appropriate. (In all cases, MM means month, DD means day, and YY means the last two digits of the year.)

Parts III, IV, and V are repeated for each device the supplier identified as delivering to your facility. You have been supplied with a complete Part III, IV, and V for each device. Each answer in Parts III, IV, and V should refer to the specific device identified in Part IV of that stapled set of Parts III, IV, and V.

Part III: Supplier (Manufacturer or Distributor) of Device in Part IV

(Please correct any incorrect information)

A. Supplier's Specific License Number	_____
B. Supplier's Company Name	_____
C. Supplier's Address	_____ _____ _____
D. City	_____
E. State	_____
F. Zip Code	_____
G. Phone Number	_____ _____

Part IV: Device Identification (Please correct any incorrect information)

A. Device Type	_____
B. Supplier's Model Number	_____
C. Radionuclide Source	_____
D. Quantity of radioactivity (Curies)	_____
E. Determination date of quantity (MM/DD/YY)	_____
F. Device Serial Number	_____
G. Date device shipped to you (MM/DD/YY)	_____

ID # _____

Device ID # _____

U.S. Nuclear Regulatory Commission

Part V: Additional Information About Device Described in Part IV (Please fill in or circle the answer as appropriate)

Please answer all questions in Part V as they relate to the device identified in Part IV.

A. Was the radionuclide ever replaced on this device?

☐ Yes ☐ No

*If you answered No to
question A, skip to
question E*

A1. If yes, what was the date of this replacement? (MM/DD/YY) _____

B. Does this device have:

B1. A durable label?

☐ Yes ☐ No

B2. A legible (easily readable) label?

☐ Yes ☐ No

B3. A clearly visible (easily seen) label?

☐ Yes ☐ No

☐ 1 - On the Source Housing

☐ 2 - On the Detector

☐ 3 - On the Supporting Frame

C. Was this device ever checked for defects after receipt?

☐ Yes ☐ No

*If you answered No to
question C, skip to
question D*

C1. If yes, when was the device last checked for defects? (MM/DD/YY) _____

D. Did your firm install this device?

☐ Yes ☐ No

D1. Was the Specific Licensee identified in Part III the same organization that installed this device at your site? A Specific License is issued to named persons upon applications filed pursuant to the regulations in 10 CFR 30.31 through 30.35.

Specific Licensees are authorized to receive, store, demonstrate, and redistribute devices containing sealed sources listed on their License (NRC Form 374).

*If you answered Yes to
question D, skip to
question E*

☐ Yes ☐ No

*If you answered Yes to
question D1, skip to
question E*

U.S. Nuclear Regulatory Commission

Part V: Additional Information About Device Described In Part IV (continued)

If another organization installed this device, please complete the following questions:

D2. Installer's Specific License Number	_____
D3. Company Name	_____
D4. Contact (First Name/Last Name)	_____
D5. Contact Title	_____
D6. Address	_____ _____ _____
D7. City	_____
D8. State	_____
D9. Zip Code	_____
D10. Phone Number	_____

If the device identified in Part IV contains only krypton, skip to question G. If the device identified in Part IV contains only tritium, not more than 100 microcuries of other beta and/or gamma emitting material, not more than 10 microcuries of alpha emitting material, or is held in storage in the original shipping container prior to initial installation, skip to question H.

E. Does your firm have a copy of the initial radiation survey performed at the time of installation?	<input type="checkbox"/> Yes <input type="checkbox"/> No
F. Since your firm received this device, have leak tests been performed?	<input type="checkbox"/> Yes <input type="checkbox"/> No
If you answered No to question F, skip to question G.	

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Part V: Additional Information About Device Described in Part IV (continued)

F1. What was the date of the latest test? (MM/DD/YY) _____

F2. Does your firm have copies of the record of the latest test?

☐ Yes ☐ No

F3. Did your firm conduct the latest test?

☐ Yes ☐ No

If you answered Yes to question F3, skip to question G

F4. Testing Company Name _____

F5. Contact (First Name/Last Name) _____

F6. Contact Title _____

F7. Address _____

F8. City _____

F9. State _____

F10. Zip Code _____

F10. Phone Number _____

G. Does the device have an on-off mechanism?

☐ Yes ☐ No

If you answered No to question G, skip to question H

G1. Since your firm received this device, have on-off mechanism tests been performed?

☐ Yes ☐ No

If you answered No to question G1, skip to question H

G2. What was the date of the latest test? (MM/DD/YY) _____

U.S. Nuclear Regulatory Commission

Part V: Additional Information About Device Described in Part IV (continued)

G3. Testing Company Name	_____
G4. Contact (First Name/Last Name)	_____
G5. Contact Title	_____
G6. Address	_____ _____ _____
G7. City	_____
G8. State	_____
G9. Zip Code	_____
G10. Phone Number	_____

H. Has this device been relocated (moved within the site and reinstalled) since it was initially installed?	<input type="checkbox"/> Yes <input type="checkbox"/> No
<div style="border: 1px solid black; padding: 5px; text-align: center;"><i>If you answered No to question H, skip to question I</i></div>	
H1. Did your firm relocate this device?	<input type="checkbox"/> Yes <input type="checkbox"/> No
<div style="border: 1px solid black; padding: 5px; text-align: center;"><i>If you answered Yes to question H1, skip to question I</i></div>	
H2. What was the date that this device was last moved? (MM/DD/YY) _____	

U.S. Nuclear Regulatory Commission

Part V: Additional Information About Device Described in Part IV (continued)

- H3. Name of company that moved the device? _____
- H4. Company Contact (First Name/Last Name) _____
- H5. Company Contact Title _____
- H6. Address _____
- H7. City _____
- H8. State _____
- H9. Zip Code _____
- H10. Phone Number _____

- I. Has your firm experienced any events that could have caused damage or did cause damage to this device?

☐ Yes ☐ No

If you answered No to question I, skip to question J

- II. If your firm experienced any event that could have damaged or did damage this device, please explain what occurred and how you responded to the event. Include the date of the event, a short description of the event, a description of any actual or suspected damage to the device, actions taken to return the device to service, any equipment or tests that were performed, and any organizations that assisted you. Please report all events that have occurred to this device. Please continue your answer on additional blank pages as necessary. Label each blank page with your company name and the ID number and device ID at the top of this page.

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Part V: Additional Information About Device Described in Part IV
(continued)

J1.	Has your company transferred this device to another firm?	<input type="checkbox"/> Yes <input type="checkbox"/> No
<i>If you answered No to question J1, you have completed the questions on this device. Please continue to the next device.</i>		
J2.	What was the date of the transfer? (MM/DD/YY)	_____
J3.	Name of Company to Whom Device Was Transferred	_____
J4.	Contact (First Name/Last Name)	_____
J5.	Contact Title	_____
J6.	Address	_____ _____ _____
J7.	City	_____
J8.	State	_____
J9.	Zip Code	_____
J10.	Phone Number	_____

Thank you for your assistance.

U.S. Nuclear Regulatory Commission

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OMB No. 3150-0152
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ID # _____

Survey of General Licensees: Facility Information

Part I contains preprinted information about your firm. Please review the preprinted information in Part I and correct any incorrect information by crossing through the incorrect information and writing in the correction above or to the right of the original information. Please complete any missing information by writing in the information in the space provided for that information.

Part II of this survey asks for additional information about your firm and your experience with the tritium-powered exit sign(s) identified in Parts III and IV of this survey. Please answer all questions in Part II as they relate to your firm as a whole. Please fill in or circle the answer as appropriate.

Part I: Respondent Identification *(Please correct any incorrect information)*

A. Company Name	_____
B. Company Contact (First/Last Name)	_____
C. Contact Title	_____
D. Address	_____ _____ _____
E. City	_____
F. State	_____
G. Zip Code	_____
H. Phone Number	_____

Part II: Additional Respondent Information

Please fill in or circle the answer as appropriate

A. What is the principal business conducted at your facility?		
<input type="checkbox"/> 1 - Aviation	<input type="checkbox"/> 5 - Laboratory	<input type="checkbox"/> 8 - School
<input type="checkbox"/> 2 - Entertainment	<input type="checkbox"/> 6 - Manufacturing (please specify) _____	<input type="checkbox"/> 9 - State/Local Gov't
<input type="checkbox"/> 3 - Food and Lodging	<input type="checkbox"/> 7 - Office Building	<input type="checkbox"/> 10 - Other (please specify) _____
<input type="checkbox"/> 4 - Hospital		

B. Does your firm have a copy of the General License?	<input type="checkbox"/> Yes	<input type="checkbox"/> No
---	------------------------------	-----------------------------

U.S. Nuclear Regulatory Commission

Survey of General Licensees: Facility Information (continued)

Part II: Additional Respondent Information (continued)

- C. Is there a company representative who knows the General License conditions? ☐ Yes ☐ No
- D. Is that person responsible for ensuring compliance with the General License conditions? ☐ Yes ☐ No
- E. Is your firm having any difficulty finding an authorized recipient to:
- E1. Purchase any tritium-powered exit signs you no longer want? ☐ Yes ☐ No
- E2. Dispose of any tritium-powered exit signs you no longer want? ☐ Yes ☐ No
- E3. Store any tritium-powered exit signs you no longer want? ☐ Yes ☐ No

*If you answered No to
question E1, E2, and E3,
skip to question F.*

- E4. If your firm is having any difficulty finding an authorized recipient, please elaborate by describing who you contacted and describe the nature of the difficulty. Please continue your answer on additional blank pages as needed. Label each blank page with your company name and the ID number at the top of this page.

- F. How many tritium-powered exit signs, held under a General License, are now in secured storage that are not intended for use? _____

U.S. Nuclear Regulatory Commission

Survey of General Licensees: Exit Sign Information

Parts III and IV contain preprinted information about tritium-powered exit sign(s) that NRC records indicate were delivered to your firm under a General License. Please review the preprinted information in Parts III and IV and correct any incorrect information by crossing through the incorrect information and writing in the correction above or to the right of the original information. Please complete any missing information by writing in the information in the space provided for that information.

Part V of this survey refers to the exit sign(s) identified in Part IV. Please answer all questions in Part V as they relate to the exit sign(s) identified in Part IV. Please fill in or circle the answer as appropriate. (In all cases, Curies are a measure of radioactivity, MM means month, DD means day, and YY means the last two digits of the year.)

Parts III, IV, and V are repeated for each type of exit sign that the supplier reported it had supplied to your facility. You have been provided with a complete Part III, IV, and V for each type of exit sign. Each answer in Parts III, IV, and V should refer to the specific type of exit sign identified in Part IV of that stapled set of Parts III, IV, and V.

Part III: Supplier (Manufacturer or Distributor) of Exit Sign(s) in Part IV (Please correct any incorrect information)

A. Supplier's Specific License Number _____

B. Supplier's Company Name _____

C. Supplier's Address _____

D. City _____

E. State _____

F. Zip Code _____

G. Phone Number _____

Part IV: Exit Sign(s) Identification (Please correct any incorrect information)

A. Exit Sign(s) Type _____

B. Supplier's Model Number _____

C. Radionuclide Source _____

D. Quantity of radioactivity (Curies) _____

E. Number of exit sign(s) held _____

G. Date exit sign(s) shipped to you (MM/DD/YY) _____

U.S. Nuclear Regulatory Commission

Part V: Additional Information About Exit Sign(s) Described in Part IV (Please fill in or circle the answer as appropriate)

Please answer all questions in Part V as they relate to the exit sign(s) identified in Part IV.

A. Were the exit sign(s) ever checked for defects after receipt?

☐ Yes ☐ No

☐ Some but not all exit sign(s) checked

If you answered No to question A, skip to question B

A1. How many exit signs were checked for defects after receipt? _____

B. Did your firm install the exit sign(s)?

☐ Yes ☐ No

☐ We installed some but not all exit sign(s)

B1. Was the Specific Licensee identified in Part III the same organization that installed this exit sign(s) at your site?
A Specific License is issued to named persons upon applications filed pursuant to the regulations in 10 CFR 30.31 through 30.35. Specific Licensees are authorized to receive, store, demonstrate, and redistribute devices containing sealed sources listed on their license (NRC Form 374).

If you answered Yes to question B, skip to question C

☐ Yes ☐ No

C. Have the exit sign(s) been relocated (moved within the site and reinstalled) since initial installation?

☐ Yes ☐ No

☐ Some but not all exit signs relocated

If you answered No to question C, skip to question D

C1. How many exit sign(s) were relocated? _____

C2. Did your firm relocate these exit sign(s)?

☐ Yes ☐ No

☐ We relocated some but not all exit signs

U.S. Nuclear Regulatory Commission

Part V: Additional Information About Exit Sign(s) Described In Part IV (continued)

- D. Has your firm experienced events that could have caused damage or did cause damage to any exit sign(s)?

☐ Yes ☐ No

If you answered No to question D, skip to question E

- D1. If your firm experienced any event that could have damaged or did damage any exit sign(s), please explain what occurred and how you responded to the event. Include the date of the event, a short description of the event, a description of any actual or suspected damage to the exit sign(s), actions taken to return the exit sign(s) to service, any equipment or tests that were performed, and any other organizations that assisted you. Please report all events that have occurred to the exit sign(s). Please continue your answer on additional blank pages as needed. Label each blank with your company name and the ID number at the top of this page.

- E. Do the exit sign(s) have:

E1. A durable label?

☐ Yes ☐ No

E2. A legible (easily readable) label?

☐ Yes ☐ No

E3. A clearly visible (easily seen) label?

☐ Yes ☐ No

☐ 1 - On the sign housing

☐ 2 - On the supporting frame

- F. Has your company transferred any exit sign(s) to another firm?

☐ Yes ☐ No

If you answered No to question F, you have completed the questions on these exit sign(s).

- F1. How many exit sign(s) were transferred? _____

Thank you for your assistance.

APPENDIX B

SURVEY COVER LETTER



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

July 1, 1990

Dear Sir/Madame:

This letter is to notify you that your firm has been chosen by the Nuclear Regulatory Commission (NRC) to participate in a survey of general licensees for gauging devices and laboratory equipment containing radioactive material. The survey is intended to confirm the placement, use, condition, and/or disposition of these devices for the purpose of evaluating the effectiveness of the current NRC regulations (10 CFR Part 31, Section 31.5) pertaining to general licensees.

Each attached questionnaire describes a particular device (including manufacturer or initial transferor, model number, type and quantity of radionuclide, and date that you received the device) that Nuclear Regulatory Commission records indicate was shipped to you during the 1980's and is currently in your possession. (If you were shipped and currently possess more than one such device, you should receive more than one questionnaire form.) Please confirm the preprinted information on each questionnaire you receive or make any necessary changes if it is not correct. Then answer for the particular device described in each questionnaire all of the other questions on that questionnaire. NRC records indicate that the person named in the address above is the contact person for this device or devices. If the listed contact person is not available, or if some other person might be better able to provide the requested information, please review the detailed instructions included in the questionnaire(s) and forward the questionnaire(s) to the appropriate respondent. Your failure to return the completed questionnaire(s) could result in your being selected for a followup onsite visit to inspect the device(s).

This survey has been approved by the Office of Management and Budget. It is being conducted for the Nuclear Regulatory Commission by ICF Incorporated of Fairfax, Virginia. If you have any questions concerning the survey, call the following toll-free number between the hours of 9:00 a.m. and 5:00 p.m., Eastern Standard Time: (800) 331-9212. The information that you provide in this survey will be treated according to confidential procedures under the Privacy Act (5 U.S. Code §552a).

If you wish to dispose of the device(s) in this survey, or of any other device containing radionuclides that you hold under a general license and is currently under your control, you should contact the manufacturer of the device(s) directly, or the nearest Regional Office of the Nuclear Regulatory Commission for assistance. For your information, a copy of those sections of the NRC's rules and regulations pertaining to you as a general licensee (Title 10 Code of Federal Regulations Parts 20, 30, and 31) is also enclosed.

Please return the survey within two weeks of receipt. Thank you for your assistance.



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Dear Sir/Madame:

Over three weeks ago your firm received one or more survey questionnaires from the Nuclear Regulatory Commission requesting information about exit signs, gauging devices, or laboratory equipment containing radioactive byproduct material. The survey is intended to confirm the placement, use, condition, and/or disposition of these devices for the purpose of evaluating the effectiveness of the current NRC regulations (10 CFR Part 31, Section 31.5) pertaining to general licensees. According to our records, however, you have not returned a completed survey.

Each of the questionnaires that you received describes signs, devices, or instruments that Nuclear Regulatory Commission records indicate were shipped to you and are currently in your possession. It is extremely important that your firm return each questionnaire, so that the treatment and disposition of these signs, devices, or instruments can be verified. Your failure to return the completed questionnaire(s) could result in your being selected for an onsite visit.

We have enclosed another copy of the questionnaire(s). If you have any questions concerning the survey, call the following toll-free number between the hours of 9:00 a.m. and 5:00 p.m., Eastern Standard Time: (800) 331-9212. Please indicate your name, address, and telephone number, the identification number on the mailing label attached to the envelope for this letter, and leave a short description of your question. If you have recently returned your survey, it may have been in the mail when this letter was mailed. If so, thank you for your assistance.

The information that you provide in this survey will be treated according to confidential procedures under the Privacy Act (5 U.S. Code §552a). This survey has been approved by the Office of Management and Budget.

Thank you for your assistance.

APPENDIX C

RESPONSES TO COMMONLY-ASKED HOTLINE QUESTIONS

RECOMMENDED RESPONSES TO COMMON QUESTIONS ON NRC SURVEY
OF GENERAL LICENSEES FOR TRITIUM EXIT SIGNS

1. The proper person to answer the questionnaire is on vacation for two weeks?

Response: Please make certain that the questionnaire is filled out and returned as soon as the proper person returns. You will be contacted by phone, and you may receive a follow-up questionnaire if substantially more than two weeks elapses.

2. I don't have possession of the light; I only installed it for someone else.

Response: Please answer Question F and Question F1. On a separate piece of paper, labeled with your firm name and the ID# at the top of the questionnaire, list the names and addresses of the persons for whom you installed the lights.

3. We don't have a General License.

Response: NRC doesn't issue written licenses for possession of exit lights [the devices that NRC is asking about in this survey]. Instead, NRC issues a license (called a specific license) to the firm that manufactures the light [device], and that firm is required to inform the people to whom it ships lights [devices] of their responsibilities. Those responsibilities are described in 10 CFR §31.5, and the terms of the regulation constitute the "General License." You should have received a copy of the regulations in the same envelop as the survey questionnaire, and you may want to review what they say, especially Section 31.5.

4. What if we never received the device?

Response: The exit signs that we are interested in should have a label. If you have checked for such a label, and if you are certain that you don't have any exit lights that aren't connected to a source of electricity, then please write on the lines in Part IV on page 3 next to the sign information "Did not receive this sign/these signs."

5. What if we received the device, but are no longer in possession of it; should we still complete the survey in its entirety or simply fill out certain parts of it?

Response: You should fill out as much of the survey as you can.

6. If we are no longer in possession of the device, how much of the survey needs to be filled out?

Response: You should fill out as much of the survey as you can.

7. We received the device, but never installed it; how much of the survey must we fill out?

Response: You should be able to fill out all of Parts I, II, III, IV, and V A., B., D., E., and F.

8. We don't know the answers to certain questions (i.e. the Installer's specific license number); should we just leave that space blank?

Response: Yes.

9. What is the Privacy Act and what are my rights in regard to confidentiality?

Response: The Privacy Act governs who can see records maintained by the federal government. Under the Privacy Act, the NRC will be able to see the results of this survey, including your responses, but the responses won't be given to anyone else in a form that allows them to identify your answers individually. A final report on the results of the Survey may be prepared, but if it is it will not name individual firms.

10. Will any actions be taken against me if the answers to the survey reveal that my company is in violation of the regulations?

Response: NRC is conducting this survey in part to see if it needs to change the regulations. If your responses show that a potentially dangerous situation may exist, NRC will contact you in order to assist you in fixing the situation.

11. What is tritium (or any of the other radionuclides)?

Response: Tritium is nuclear byproduct material. Its possession and use is licensed by NRC. You should contact the specific licensee from whom you obtained the exit light [device] for an explanation of its characteristics and detailed instructions on the care of the tritium exit light.

12. How can I determine if the label on the device is clearly visible?

Response: The label should be in a place where you can see it easily, without removing the light from where it is installed. The purpose of the label is to inform people that the light contains byproduct material, so it should be situated so that it can do that.

13. Does Hurricane Hugo count as an event which could have caused damage to the device (or other situations or events)?

Response: NRC is interested in making certain that proper precautions are taken, and that the specific licensee is notified in situations when the light might have been damaged. Please

provide details of events that did raise a question in your mind about whether the light was damaged.

14. I lost the first two sections of the survey; can you send me a new copy?

Response: Yes. Please allow a few days for delivery.

15. Is the survey optional?

Response: One of the conditions for having byproduct material is that you provide reasonable information about how you are treating the material in your possession. The Office of Management and Budget, which has reviewed NRC's request to conduct this survey, has given its approval for the survey. You are not subject to any penalty for not returning the survey questionnaire, but the NRC may send out an inspector to obtain the same information that is asked for on the survey form.

16. Where can I find the serial number of my device?

Response: Exit lights may not have an obvious serial number. You should contact the supplier if you want to know the number. However, you are not required to provide the serial number as an answer on the questionnaire.

For gauges and analytic devices, the serial number should be plainly marked on the housing or frame of the device.

17. Does a given situation count as "difficulty" in finding an authorized recipient; if so, will anything be done to offer us assistance in resolving the difficulties?

Response: By difficulty, we mean a situation in which you have more than usual trouble in finding someone to do what you want. If you have such difficulty, your NRC Regional Office will be able to assist you. The location and phone number of NRC Regional Offices are:

[Insert]

18. We were sent the wrong survey (i.e. name and address on envelope do not match those on the cover letter); what should we do?

Response: If the survey questionnaire information is correct, please disregard the cover letter. If the questionnaire is not correct, please indicate that in the space provided in Part I by writing in the words "Sent to wrong firm." Please include your own name and address.

19. How and/or why were we chosen by the NRC to participate in this survey?

Response: Suppliers of exit signs are required by law to report to NRC

every three months the names and addresses of the firms/persons to whom they have shipped exit signs containing byproduct materials. NRC chose survey respondents from those quarterly reports.

What is your name and phone number?

Response: Give your name. The phone number is 800-331-9212.

21. What Office are you with at NRC?

Response: I am employed by ICF Incorporated, who has been hired by NRC to conduct the survey for NRC.

TECHNICAL LETTER REPORT:
TASK 7, FINAL REVIEW OF THE 1987 REPORT BY
OAK RIDGE ASSOCIATED UNIVERSITIES,
"IMPROPER TRANSFER/DISPOSAL SCENARIOS FOR
GENERALLY LICENSED DEVICES"

NRC JOB CODE L2536
PNL No. 20278

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1.0 EXECUTIVE SUMMARY

Task 7 of the project "Review of Improper Transfer/Disposal Scenarios for Generally Licensed Devices Study" requires that, "after Tasks 5 and 6 are completed, PNL will prepare a draft report based on the results of the individual reviews that documents its findings, conclusions and recommendations. The report will present a critical evaluation of the methods, data and assumptions used in the ORAU [Oak Ridge Associated Universities; now Oak Ridge Institute for Science and Engineering] study (Stabin et al. 1987). Any limitations or inadequacies will be described, as well valid insight and conclusions reached by the ORAU team. The report will specifically address the suitability of the 1987 ORAU report as a basis for revising regulatory requirements or guidance in 1993. It will provide the Staff with recommendations on how the contents of the ORAU report can be used in formulation of policy or regulation, as well as additional facts and information to support their decisions." "Within 2 months of receiving NRC staff comments on the draft report, PNL will respond to the comments and provide a final report to the Project Manager..."

As part of Task 5 of this project, Roger Cloutier, Kermit Paulson, Mike Stabin, and Evelyn Watson (the "ORAU Team") attended a meeting with Robin Hill, George Cicotte, Jim Dukelow, and Dan Strom of the Pacific Northwest Laboratory (PNL) on October 5, 1993, in Oak Ridge, Tennessee. The meeting was a very productive discussion with the original authors and team of the 1987 ORAU report. All PNL questions were satisfactorily resolved, as is documented in the Appendices to this final report.

The PNL reviewers conclude that the 1987 ORAU Report provides a good start on assessing worst-case consequences of improper transfer and disposal scenarios for generally licensed devices. For use as a basis for regulatory decision making, the principal shortcomings of the ORAU Report identified by the PNL review are:

- The 1987 ORAU Report does not include probabilities of the scenarios occurring on a per source, per year basis.
- The 1987 ORAU Report does not include a complete enough enumeration of the numbers of sources, and the distributions of source activities, in each category.
- The 1987 ORAU Report does not include the probabilistic distributions of outcomes (as opposed to the worst case outcomes) needed to realistically assess the probable human health consequences of such scenarios.

The original ORAU Team was constrained from considering many of the above issues by the limited scope of work of their project. The PNL reviewers repeat that the ORAU Report is a good start on the collection of data to support regulatory decision making.

As in the preliminary PNL review, the PNL reviewers conclude that the 1987 ORAU report is not an adequate basis for 1993 regulatory decision making, and that members of the NRC

Staff may need to use additional facts and information.

2.0 SUMMARY OF CHANGES SINCE 1987

Since 1987, there have been changes in several areas that impact the current relevance of the 1987 ORAU Report. These include changes in recommended limits on dose to the public, significant changes and developments in probabilistic dose assessment methodologies, and changes in dosimetric quantities and models that affect risk assessments.

2.1 CHANGES IN, AND CREATION OF, PUBLIC DOSE LIMITS

The original work under review in this project is referred to as the "ORAU Report" (Stabin et al. 1987). Since that report was prepared, NRC has instituted a dose limit for the public that is in concert with recommendations of radiation protection advisory groups.

For doses to the public (when such doses arise from licensee activities), the NRC has implemented a limit of 0.1 rem (0.001 Sv) total effective dose equivalent (TEDE) per year (10 CFR Part 20.1301(a)(1)). In addition, the NRC now specifies that the provisions of the U.S. Environmental Protection Agency's 40 CFR Part 190 apply to licensee activities.

Parenthetically, it is noted that the U.S. Department of Energy has lowered its limits for exposure to the public (10 CFR 835) to 0.1 rem (0.001 Sv) TEDE.

Since 1987, both the NCRP (1993) and the ICRP (1991) have lowered their recommended limits for members of the general public, to the same value of 0.1 rem (0.001 Sv) for the quantity *effective dose*, which is similar to TEDE although not identical. The NCRP and ICRP recommendations are based on new risk findings of the National Academy of Sciences (NAS 1988, 1990) and the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR 1988).

With the establishment of a limit on TEDE to members of the public of 100 mrem (1 mSv; 10 CFR 20.1301(a)), or 2% of the occupational dose limit, there is less margin for error in dose assessments for improper transfer and disposal of generally licensed devices. The new, lower 10 CFR 20 limits for the public should be used in assessing impacts of improper transfer or disposal of generally licensed devices.

2.2 CHANGES IN PROBABILISTIC RISK METHODOLOGY

Since 1987, many changes have occurred in probabilistic risk methodologies. Recent summaries of these techniques are provided by IAEA (1989), Finkel (1990), and Morgan and Henrion (1990). In addition, the advent of user-friendly Monte Carlo simulation software for probabilistic health risk analysis, such as Crystal Ball (™Decisioneering, Inc., Denver, CO), makes it feasible to perform probabilistic risk assessments for this kind of work.

Distributions of source activities, and distributions of consequence severities, and probabilities of incidents occurring should be used to predict likely outcomes, with worst case outcomes being found in the extreme values of the resultant distributions.

2.3 CHANGES IN DOSIMETRIC QUANTITIES AND MODELS

Since 1987, quantities such as effective dose or total effective dose equivalent has replaced "dose equivalent" in most radiation protection recommendations and standards. Many improvements in internal dosimetry have occurred, as detailed in Appendix B.

3.0 THE ADEQUACY OF THE 1987 ORAU REPORT AS A BASIS FOR 1993 DECISION MAKING

The 1987 ORAU Report, in the judgment of the PNL reviewers, is no longer adequate for 1993 decision making. For use as a basis for regulatory decision making, the principal shortcomings of the ORAU Report identified by the PNL review are:

- The 1987 ORAU Report does not include probabilities of the scenarios occurring on a per source, per year basis.
- The 1987 ORAU Report does not include a complete enough enumeration of the numbers of sources, and the distributions of source activities, in each category.
- The 1987 ORAU Report does not include the probabilistic distributions of outcomes (as opposed to the worst case outcomes) needed to realistically assess the probable human health consequences of such scenarios.

The original ORAU Team was constrained from considering many of the above issues by the limited scope of work of their project. The PNL reviewers repeat that the ORAU Report is a good start on the collection of data to support regulatory decision making.

Additional shortcomings include

- changes in dose quantities (e.g., the introduction of effective dose equivalent from external irradiation) and in regulations (e.g., public limits on total effective dose equivalent) have taken place; and
- several potentially significant scenarios, such as an intact source out of a shield, and potentially significant consequences, such as doses to workers (rather than the public), have been omitted.

These topics are supported in individual reviewer comments in the Appendices to this report.

4.0 ADDITIONAL FACTS AND INFORMATION THAT MIGHT BE USED BY THE NRC STAFF TO DECIDE ON THE NEED FOR REGULATORY ACTION

As in the preliminary PNL review, the PNL reviewers conclude that the 1987 ORAU report is not an adequate basis for 1993 regulatory decision making, and that members of the NRC Staff may need to use additional facts and information.

4.1 AREAS IN WHICH ADDITIONAL FACTS AND INFORMATION ARE NEEDED

NRC Staff should consider

- the annual *rates* of incidents involving improper transfer or disposal by source category. Rate assessments require improved data (from both NRC-regulated and Agreement States) on numbers of sources, numbers of incidents, and dosimetric consequences (both individual and collective) of incidents;
- probability distributions of severity and of occurrence for various accident scenarios. There should be assessments of the magnitudes of doses and the sizes of the exposed populations that are likely to result from each instance of improper transfer or disposal of these devices. Such assessments can be based on historical incidents such as the Juarez, Mexico; Goiânia, Brazil; Korea-to-USA; and Indiana, Pennsylvania incidents. These distributions for various accident scenarios are needed to develop both the individual and collective dose estimates;
- a state-of-the-art probabilistic risk assessment with predictions of the impact of improper transfer and disposal scenarios on individual and collective TEDE. Such a risk assessment should include "worst case" scenarios only as limits of distributions on a probabilistic basis, not as simple point estimates;
- the impact of proposed changes in regulations on the benefits and economics of use of generally licensed devices, and on the reduction in risk to users and the public. Such assessments are optimization studies or regulatory impact analyses; and
- the need for much better data on numbers of devices. For a complete risk analysis, no *assumptions* should have to be made about the numbers of devices, their isotopes, their activities, Device Codes, design, and date placed into service: these numbers should be used directly or in categories with sufficient detail (e.g., 1600 ¹³⁷Cs gamma gauges of design XYZ placed in service at a rate of 100 per year beginning in 1978...) to perform the risk analysis.

4.2 NEW WORK BY PNL REVIEWERS FOR DISTRIBUTIONS OF ACCIDENT CONSEQUENCES

Pacific Northwest Laboratory reviewers have provided additional information in three areas.

4.2.1 Incident Rates on a Per Source, Per Year Basis

Appendix E contains the Revised Task 3 Technical Letter Report: Evaluation Of Historical Sealed Source Device Experience. This report incorporates additional data provided by NRC/NMSS on sealed source registrations. Incident rates, on a per source, per year basis, can be calculated as detailed in Appendix G of this report, using data from Appendix E of this report.

4.2.2 Proposed Probabilistic Framework for Risk Analysis

Appendix F contains the Revised Task 6 Technical Letter Report on the Development of Additional Probability and Risk Information. A proposed framework for risk analysis is included in that report. This framework could be used to support regulatory decision making. The decision makers would have to make judgements regarding acceptable levels, expressed in probabilistic terms, of collective effective dose equivalent, individual total effective dose equivalent, and individual local (or skin) dose equivalent for each category of sources analyzed.

For example, one category of source may have a once-in-20-years probability that a member of the public may receive a TEDE in excess of 0.1 rem from the practice of generally licensing such sources. Another category may have a higher probability, or even a virtual certainty, of one or more members of the public exceeding this 0.1 rem TEDE every year, but with a very large benefit to other members of the public. These are admittedly tough decisions, but, in the view of the PNL review team, the probabilistic information on which to base them should be made available to the decision makers.

4.2.3 Exposure Probabilities in Accidents

The revised Task 6 Report in Appendix F contains analyses of 42 historical accidents involving external doses to workers or the public, and gives distributions of Time-and-Proximity Factors for individual whole body, individual local (or skin), and collective doses.

Distributions of Time-and-Proximity Factors from historical accidents can be used in probabilistic risk analyses for both whole-body and local irradiation from external sources. An analysis of 42 accidents for which source identity and strength are available show that the average accident victim gets a whole body dose equal to that from being at 1 meter from the accident's unshielded source for an hour. The average accident is characterized by a value of 46 hours at a meter. In other words, the population-weighted average is about 1 hour at a meter, while the accident-weighted average is 46 hours at a meter. Clearly, the accidents with large numbers of victims (e.g., Goiânia and Juarez) dominate the former average. The maximum value seen for whole-body doses is about 700 hours at a meter (from the 1972 Texas child-abuse case). The average, geometric mean, and maximum values for local irradiation are 3100, 60 and 24,000 hours at a meter, respectively.

Such distributions should be used in probabilistic risk analyses to determine likely distributions of risks or doses from improper transfer and disposal scenarios for generally licensed devices.

Accidents that were terminated due to the appearance of clinical symptoms of acute irradiation have less value for risk analyses than accidents that were terminated by other means, or never terminated.

The current NRC Office of Analysis and Evaluation of Operational Data (AEOD) Nuclear Regulatory Event Report (NRER) incident database does not contain the kinds of information needed to perform analysis of accidents for Time-and-Proximity Factors. It is recommended that the database either be modified to include this information, or a separate database be created. There is a great deal of work to be done to refine these preliminary analyses, extend them to additional accidents, and develop the logical framework for extrapolating to other kinds of sources and scenarios.

For intakes of radioactive materials, 60 historical accidents have been characterized by distributions of individual Fractions-Taken-In, that is, the fraction of the activity in the original source that was taken in by each individual involved in the accident. For most accidents involving radioactive sources, the fraction taken in is zero (0). For one accident, that at Goiânia, Brazil, in 1987, hundreds of persons had intakes, including 194 cleanup workers and at least 77 members of the public. The former were characterized by Fractions-Taken-In on the order of 10^{-12} , and the latter by Fractions-Taken-In averaging 5×10^{-6} . These values are far below the maximum possible value of 1, or of the value of 0.3 adopted as a worst case in the ORAU Report. In no accident has a value greater than 0.01 been seen.

5.0 PRELIMINARY RISK ANALYSIS

A full implementation of the risk analysis described in Section 2 is beyond the scope of work of the current project. However, a sample risk analysis for ^{137}Cs gamma gauges is given in Appendix G.

6.0 CONCLUSIONS

The PNL reviewers conclude that the 1987 ORAU Report provides a good start on assessing worst-case consequences of improper transfer and disposal scenarios for generally licensed devices. For use as a basis for regulatory decision making, the principal shortcomings of the ORAU Report identified by the PNL review are:

- The 1987 ORAU Report does not include probabilities of the scenarios occurring on a per source, per year basis.

The 1987 ORAU Report does not include a complete enough enumeration of the numbers of sources, and the distributions of source activities, in each category.

The 1987 ORAU Report does not include the probabilistic distributions of outcomes (as opposed to the worst case outcomes) needed to realistically assess the probable human health consequences of such scenarios.

The PNL reviewers conclude that the 1987 ORAU report is not an adequate basis for 1993 regulatory decision making, and that the NRC Staff should consider additional facts and information in the areas described above. In particular, the use of worst case scenarios with unrealistically high exposure factors tends to make the consequences of improper transfer and disposal seem worse than they would probably be.

An additional conclusion is that existing databases are not adequate for performing modern, probabilistic risk analyses. It would be desirable to collect and store in database format dosimetric information (including quantitative measurements or estimates of intakes) for all individuals involved in accidents. Such a data base would have to have an entry for each exposed individual, rather than simply one line of data per accident.

7.0 ACKNOWLEDGEMENTS

The PNL reviewers are pleased to acknowledge the help and support of numerous colleagues in the preparation of this report. At NRC headquarters, significant technical help and feedback has been given by Steven L. Baggett and Sterling Bell. The ORAU team of authors of the 1987 report, Roger Cloutier, Kermit Paulson, Mike Stabin, and Evelyn Watson, were very helpful in discussions held concerning that report. At PNL, original project manager James R. Jamison and technical contributor Peter C. Olsen have helped, as have technical reviewers Paul S. Stansbury and Eva Eckert Hickey. We are grateful to Betty Anderson and Rebecca Webster for clerical support.

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APPENDIX A: FINAL REVIEW OF 1987 ORAU REPORT BY D.J. STROM,
INCORPORATING COMMENTS FROM VISIT TO ORAU TEAM, OCTOBER 4, 1993.

For this appendix, the following conventions are used:

- The original reviewer's comments made in the Task 4 draft letter report are shown in italics.
- The comments and results received during the meeting with the authors of the 1987 ORAU report are given in standard text format following specific Task 4 comments.

A.1. GENERAL COMMENTS

1. *Effective dose equivalent (EDE), committed effective dose equivalent (CEDE, which is used for internal doses), and total effective dose equivalent (TEDE) should be used, not simply dose equivalent.* The ORAU team was in agreement on this point.
2. *Which sources are normal form, and which are special form (IAEA/DOT classification)? This affects transport and fate of radionuclides in many accident scenarios. This should be determined for sources in future analyses. ORAU team did not consider this. However, Dodd et al. 1989 have considered this as important.*
3. *Probabilistic risk assessment methodologies have changed dramatically since 1987. This reviewer would suggest using making better use of historical accidents to evaluate probabilities of dose relationships for accidents. If risk is defined as (probability) \times (severity), then probability distributions for severity and for probability of occurrence should be given for various accident scenarios. There should be assessments of how likely a given scenario is to happen per source or device, and what kinds of doses to how many people are likely to result. Distributions of doses to individuals and the numbers of individuals receiving doses for various accident scenarios are needed to develop individual and collective dose estimates. The ORAU team made it explicitly clear that they had been requested not to consider the probabilities of accidents happening, but rather what the scenarios and consequences would be if an incident of improper transfer or disposal did occur. In our discussions with the ORAU team, they made it clear that they understood a risk analysis to consist of three parts: 1) the probability that an event will occur per source; 2) the number of sources; and 3) what the consequences are if an event does occur. They said that their statement of work limited them to the third part of this, even though they had addressed, to some extent, the second part.*
4. *Specific equations for the various models used are not present. This reviewer would prefer to see an equation for each dose that is arrived at. The ORAU Team thought this was a good idea.*

5. *Better assessments of the rates of incidents per source or device, along with the numbers of sources and devices in use, are needed. The ORAU Team concurred wholeheartedly, but made it clear that this was outside their scope of work.*
6. *10 CFR 32.51(a)(2)(iii) provides the justification for the probabilistic approach in its use of the word "unlikely." The intent of this regulation seems not to be focused on the worst case scenario, but rather on the likelihood of a given scenario. This significantly affects the likelihood of source dispersal in some accident settings. There was considerable discussion with the ORAU Team concerning worst case versus average case incidents. They insisted that the worst case analysis had been requested. They felt that average cases and distributions should also be considered. They considered, for example, a "worst case" intake fraction of 0.3, with a "realistic" intake fraction in the range of 10^{-4} to 10^{-6} .*

Specific Comments

1. *p2 Generic scenarios: Why no "release" from incinerator, scrap dealer, and metal recovery on the generic drawing? Certainly such releases have been seen, e.g., Juarez, Auburn Steel. This should be considered.*
2. *p3 ¶1 Relative radiotoxicity per unit activity given in Safe Handling of Radionuclides (IAEA 1973): obsolete and ordinal; interval ALIs should be used. Also, something along the lines of the derivation of the values the A₁A₂ system for transportation could be used for relative radiotoxicity. Essentially, the A1 (Special Form, not expected to disperse in an accident) value was 9/T curies, where T is the external exposure rate in $R\ h^{-1}\ m^2\ Ci^{-1}$. While this comment is true, the risk analyses do not really depend on the old IAEA classification; PNL reviewers concur.*
3. *p3 ¶3 The "estimated probabilities are educated guesses ..." Was a Delphi process used? How was consensus reached? The ORAU Team explained to us that for many needed data items, there was no way to scientifically get the numbers. They simply discussed some values, and came to a group consensus on their values. Examples of educated guesses are ¶2, p.3, §2.2; §2.3.2.2; §2.3.2.3. Many other unsupported data items were taken directly from Buckley et al. (NUREG/CR 1775; 1980).*
4. *p3 ¶5 Table 4: the known number of cases of mishandled devices. This table doesn't list rates of accidents, i.e., per source and per year. The incidents are from NRC and agreement state data. The numbers of sources are from NRC data. What is needed is the accident rate per source-year, by type of source. This would be the number of incidents reported in a category divided by likelihood of an accident being reported in that category per year. Let*

$R(C,y)$ denote the incident rate, per source and per year ($source^{-1}\ year^{-1}$);
 C denote the source category, e.g., A-1;

y denote the year of incident reporting and source use;
 $n(C, y)$ denote the number of incidents reported for that source category for that year;
 $N(C, y)$ denote the number of sources of category C in use in year y ; and
 $p(C)$ denote the fraction of incidents involving sources in a given category that are reported (presumed to be independent of year).

Then

$$R(C, y_i) = \frac{n(C, y_i)}{N(C, y_i) p(C)}$$

and, letting \bar{R} denote the average rate,

$$\bar{R}(C) = \frac{1}{I} \sum_{i=1}^I R(C, y_i)$$

To fully address the rates at which accidents occur in each category, it would be necessary to have values of the above variables by year for a period of time long enough to give confidence in the rates and their fluctuations. In particular, the variable $p(C)$ is problematic, since estimation of under-reporting is always difficult to ascertain.

In the 1987 ORAU report, it seems that the incidents reported represent an unknown fraction of incidents. Furthermore, it is not clear whether those incidents are for all sources, agreement state sources, what fraction of agreement states, since there may be some reporting bias. For example, data may not be available from all Agreement States or may not be available in the same level of detail. The degree of extrapolation needed to get rates should be specified. Reports probably underestimate incidents, and the underestimation is probably worst for smaller sources. Again, the ORAU team agreed that this was an important point, but that it was outside the scope of their work.

5. p4 ¶2 §2.3 line 2: Using the "largest source:" will result in a high-biased estimate of risk. Worst-case analysis was requested by NRC in 1987, according to the ORAU team.
6. p4 ¶2 §2.3: Between "intact, shielded" (Case 1) and "wide dispersal" (Case 2) there should also be "source intact but unshielded." The latter is a likely outcome, e.g., in the Indiana, PA ^{192}Ir incident and the bulk of the radiography incidents.

Discussions with the ORAU Team led us to create Table A-1 of exposure potentials for various scenarios. The question of whether the ingestion of an intact source is "internal" exposure is moot; ingestion may or may not result in uptake of radioactive material from the gastrointestinal tract.

Table A-1. Exposure potential for external, contact, and internal irradiation for seven general device and source scenarios.

Scenario	Exposure Potential		
	External	Contact	Internal
1. Source inside device	x		
2. Source out of device or shielding compromised (e.g., shutter open)	xx		
3. Source in device and source leaking	x		x
4. Source removed from device and source not leaking	xx	xx	
5. Source removed from device and source leaking	xx	xx	x
6. Source removed from device and source dispersed	xx	xx	xx
7. Source removed from device and intact source ingested	x	xx	x

The accidents involving the most people, such as the Juarez, Mexico accident in 1983-4 and the Goiânia, Brazil accident in 1987, have been in scenario 6. Many fatalities and acute radiation injuries have resulted from scenario 2 and 4 accidents.

7. p4 ¶2 §2.3: Average dose equal to 1/2 maximum (NUREG 1980) does not agree with history. Doses are likely to be lognormally distributed (e.g., uranium in urine, occupational doses (UNSCEAR 1977), etc.) The 1/2 value was taken from Buckley et al. (1980).
8. p4 ¶5: "...population dose equivalents were not calculated." Collective dose equivalent can and should be calculated, using a probabilistic basis. While there is concern for the maximally-exposed individual, the overall detriment under a linear, no-threshold dose-response hypothesis is proportional to the collective dose equivalent. The size of the population to be used in collective dose calculations should be addressed. Collective dose was considered for incineration scenarios.
9. p4 ¶6: "Gamma ray dose constants were taken from Unger and Trubey..." All work should be in EDE rate constants; see attached. For doses derived from values in roentgens, Γ values in roentgens should be used for calculating. If new risk factors (e.g., ICRP 1991, UNSCEAR 1988, NAS 1990) are used, then appropriate quantities and units should be used. This may make a significant difference for external

exposures to some low energy photon emitters.

10. pp4-5: Mexican ^{60}Co Accident (Andrews 1963): The most highly exposed boy's exposure corresponded to 687 hours at 1 meter, a high very time-and-proximity factor, with average being 446, geometric mean 408, GSD 1.7. However, the ORAU Report assumption (Exec. Summary) of 20 weeks at 100 cm (1 m) corresponds to 3360 h at a meter. (Note: Generalization of high-dose accidents is limited because of the censorship that occurs due to fatality. Presumably, with no fatalities, the source would be there for a very long time.) See TASF 6, development of additional probability and risk information. Both whole body exposures and localized (often extremity) exposures should be considered in a risk analysis, since the latter may result in deterministic (formerly non-stochastic) effects even though the former may not.
11. p6 ¶2: "These values are given with the qualification that the listed values must be increased by 25-45 percent to account for electron production in the stainless steel walls assumed to encapsulate the source." This "buildup" is not founded in either experiment or theory. To verify that there is no need for this correction, one could perform Monte Carlo calculations using the MCNP code for photon emissions from stainless steel encapsulated sources of ^{137}Cs , ^{60}Co , and ^{192}Ir . Indeed, the correction that is needed is one to effective dose equivalent rather than air kerma ("exposure is the ionization equivalent of collision kerma in air" - Attix, 1980) or "free-air dose." Table A-2 shows the percent (effective dose equivalent)/(ambient dose equivalent) [i.e., $\text{EDE}/\text{H}^*(10)$] for various nuclides, expressed as a %, by irradiation geometry.

Table A-2. EDE/H*(10) by nuclide and irradiation geometry using ICRP 51 conversions of photon fluence vs. energy to EDE and H*(10) (Strom, 1993).

Nuclide	(AP)	(PA)	(ROT)	(ISO)	(LAT)
Cd-109	33.2%	14.6%	16.4%	13.6%	8.4%
I-125	42.6%	22.0%	21.9%	18.0%	12.4%
La-133	61.2%	40.2%	36.0%	29.3%	23.9%
La-201	72.6%	52.5%	46.1%	37.7%	33.2%
Co-57	78.9%	61.6%	53.2%	42.9%	38.8%
Tc-99m	81.4%	64.1%	55.3%	44.9%	40.5%
I-131	86.8%	73.7%	64.9%	54.4%	51.7%
Ir-192	87.3%	74.1%	65.3%	54.6%	51.9%
Ra-226	87.5%	78.6%	70.7%	61.7%	60.0%
Cs-137	87.8%	77.3%	69.0%	58.7%	57.2%
Al-26	89.7%	81.9%	73.7%	64.8%	64.5%
Co-60	90.0%	82.8%	74.7%	65.8%	65.7%
Na-24	91.6%	86.3%	78.5%	71.0%	72.1%

12. p5 ¶3: Maximum contact time of 3 hours: where did this "hypothesis" come from? ORAU Team consensus arrived at this value.
13. p5 ¶4: The use of internal dose assessments based on committed effective dose equivalent methods of ICRP 30 is good. No comment.
14. pp5-6: Brodsky was not cited (Brodsky, A. 1980. Resuspension Factors and Probabilities of Intake of Material in Process (or "Is 10^{-6} a Magic Number in Health Physics?"). Health Phys. 39(6):992-1000.). Intake should depend on mass involved. Intakes of significant masses are not plausible (e.g., for ^{238}U and ^{232}Th). This is addressed above under worst case scenarios.
15. p6 §2.3.2: An important scenario is missing, namely, the scenario of an intact, but unshielded, source. This has happened repeatedly in industrial radiography settings. See Table A-2, above.
16. p6 last ¶: The value of "5 devices per year" needs clarification. Is 5 devices per year of each kind or 5 devices per year of all kinds? An ORAU Team consensus.
17. p7 11: There is no justification for the value of a nearby population of 73,000.

Buckley et al., p. D-31.

18. *p9 ¶2: average dose half as large as maximum dose? Unjustified based on accident histories. Review NUREG 1980. Buckley et al.*
19. *p14 Incineration studies: this seems incomplete, but may be adequate. Some incineration references are missing, e.g., Hamrick'a & Watson's work. There are doubtless other NTIS reports or DOE reports, in addition to the refereed literature. Jim Tripodes, who has hosted the recent incineration conferences, should be contacted. The question to be addressed is releases from low-tech incinerators, not incinerators with high-tech, scrubbed effluents. The ORAU team did a fairly thorough search.*

In conclusion, many of the major shortcomings that this reviewer found in the ORAU Report stemmed from the assumption that it was to be a complete risk analysis, when in fact it was a worst-case analysis starting from the assumption that a device had already gotten out of control. Other issues, as discussed above, remain unresolved.

APPENDIX B: R.L. HILL'S REVIEW OF THE 1987 ORAU REPORT,
"IMPROPER TRANSFER/DISPOSAL SCENARIOS
FOR GENERALLY LICENSED DEVICES"

For this appendix, the following conventions are used:

- The original reviewer's comments made in the Task 4 draft letter report are shown in italics.
- The comments and results received during the meeting with the authors of the 1987 ORAU report are given in standard text format following specific Task 4 comments. Major conclusions from this review are given in the subsection following the specific comments.
- New information pertaining to aspects related to internal dosimetry are given in a separate section at the end of this appendix.

B.1 GENERAL COMMENTS

- G1. *This document should be a complete, stand-alone document where the methods are fully described in the document. As it stands now, NUREG/CR-1775 is referred to in order to obtain descriptions of many of the methods used in the study. The assumptions used from NUREG/CR-1775 for determination of external dose should be reviewed for appropriateness and consistency. For instance, on Page 4 of the NUREG/CR report, it is stated that a point source is assumed, while in Appendix A, a line source is assumed. Also, in Appendix A of the NUREG/CR report, the internal doses are calculated using dose conversion factors based on ICRP 2 methodology, which is not the case for the GLD study.*

The authors of the 1987 ORAU report indicated that, while NUREG/CR-1775 was heavily relied upon, the internal dose calculations were done using ICRP 30 methodology.

- G2. *It appears that a lot of "short-cuts" were taken in this study, i.e., referring to another document for methods and not including worker dose calculations, food pathways, or intruder-type scenarios. A more in depth analysis is needed in order to encompass all probable scenarios that may lead to a public dose.*

The authors of the 1987 ORAU report indicated that they include the worker doses as part of the doses to members of the public.

- G3. *Better presentation and summary of the resulting dose estimates are needed. It would be much easier to reach conclusions on the study if all results were located in one location, such as a collection of several summary tables.*

The ORAU authors agreed that a better summary of the data would greatly enhance the readability of the report.

- G4. *The difference between MEI and average individual, and maximum individual and realistic individual needs to be better defined.*

The ORAU authors agreed that more realistic values for the MEI and average individual doses need to be used.

- G5. *Two important pathways/scenarios missing from this study that may have potentially large impacts on the reported dose estimates are food pathways and worker scenarios.*

The authors of the 1987 ORAU report indicated that the worker doses were lumped in as part of the public doses; and that, since they were not experts in using the dispersion and groundwater models, they relied heavily with methods used in NUREG/CR-1775, which did not include food pathways.

- G6. *Better estimates for parameters used to model incineration are needed.*

The authors of the 1987 ORAU report indicated that at the time the report was prepared, most, if not all, of the available literature on the topic was based on incineration of medical wastes. They used what information could be obtained from the available literature.

- G7. *More up-to-date computer codes are available for modeling the doses from plumes, incineration and landfills (i.e., GENII, GENII-S, CAP88-PC, etc.).*

The authors of the 1987 ORAU report they realize that they were not experts in using the dispersion and groundwater models, and that they probably did not use the most up-to-date models for the 1987 time frame. They stated that now they realize the project would have benefitted from having specialists in these areas involved on the project.

- G8. *An uncertainty and sensitivity analysis is needed since the values for most parameters used in this modeling exercise have wide or largely unknown ranges.*

The authors of the 1987 ORAU report stated that they were not asked to perform an uncertainty and sensitivity analysis for that report, and they probably could not have done such an analysis with the data and analytical tools available when the report was being prepared.

B.2. SPECIFIC COMMENTS

- S1. Page i, last ¶: *It appears from the information presented here, that the four cases considered in this report for internal doses are 30% intake for inhalation, 30% intake for ingestion, inhalation after incineration, and ingestion after leached from a landfill. The fractional intake values need to be substantiated.*

The authors of the 1987 ORAU report stated that the 30% values was taken as an educated guess or a benchmark value. However, they could not provide data to support the selected value.

- S2. Page ii, last ¶: *A metal recycling scenario was only considered for ^{60}Co . For a more realistic assessment, any of the GLD materials considered in this study could possibly find their way to a melter. Thus, all nuclides involved for GLDs should be considered in the dose assessments.*

The authors of the 1987 ORAU report agreed that the portion of the report dealing with recycling should be updated.

- S3. Page ii, last ¶: *A justification (i.e., calculated dose estimates) is needed for the statement that for metal recycle, "...dose equivalents received by members of the general public who purchase contaminated products would most likely not exceed 500 mrem/yr (0.005 Sv/yr) in most cases". This might not be the case since in IAEA Safety Series 111-P-1.1 (IAEA 1992), where generic exposure and pathway analyses were used for recycle of steel, a limiting dose of $8.8\text{E-}5$ Sv per Bq/g in the scrap (0.33 mrem per pCi/g) was estimated for ^{60}Co .*

No response was given for this comment.

- S4. Page I, 4th ¶: *It is stated that "potential" scenarios are developed and assessments provided for "realistic" and "maximum dose equivalent to individuals" (MEI). However, in the current regulatory environment (for the NRC), rulemaking is not necessarily based on "worst-case" but rather on "prudently conservative" assumptions for the dose assessments. To this end, both deterministic assessments (based on the most realistic data available from literature, etc.) and stochastic uncertainty and sensitivity analyses are needed. The Latin Hypercube Monte Carlo method for uncertainty and sensitivity (U&S) analyses was published in 1964 by staff at the Sandia National Laboratory and is a possible approach to performing U&S analyses.*

The authors of the 1987 ORAU report stated that they were not asked to perform an uncertainty and sensitivity analysis for that report, and they probably could not have done such an analysis with the data and analytical tools available when the report was being prepared.

- S5. Page 3, 1st ¶: *The radiotoxicity classification from the 1973 IAEA document cited is perhaps out of date. More recently, the IAEA has used the following classification which may be substituted: alpha emitters, photon emitters with large external dose conversion factors (DCF), no-photon emission with moderate internal DCFs, and other low dose radionuclides (IAEA Safety Series 111-P-1.1, 1992).*

The authors of the 1987 ORAU report stated that they used this radiotoxicity classification to narrow down from 600 categories to 16. It was used because the authors were familiar with it at the time the report was being prepared.

- S6. Page 3, 3rd ¶: *The probabilities assigned were "educated guesses". This brings up another reason to perform an uncertainty and sensitivity analysis (U&S) for the assessment. The upper and lower bounds for parameter values can be researched and used in the U&S analysis. This way, a range of possible dose estimates can be reported instead of point estimates.*

See response to comment S4.

- S7. Page 4, Sect. 2.3: *For the dose assessments, only 'intact device' and 'dispersed device over wide areas' were considered. Shouldn't damaged device with limited dispersion be considered since the concentrations available for intake or exposure would be higher for limited dispersion (i.e., less dilution)?*

The authors of the 1987 ORAU report agreed with this assessment.

- S8. Page 4, 4th ¶: *Since only several people are assumed to be exposed to intact devices, no population doses are calculated. Is this valid, or should population dose be considered?*

The authors of the 1987 ORAU report contend that contact with an 'intact' source would be limited to a very small number of people. However, as shown in the appendix to the report that deals with additional risk information, data from actual accidents of mishandled sources indicate that a great many people can be exposed from an 'intact' source. (See Tables 3.3.1 and 3.3.2. of that appendix.)

S9. Page 5, last ¶: *It is very hard to justify the intake fractions assumed in this report. Using DOT-based scenarios as a source of these values is not entirely appropriate. More recent incidents or accidents involving improperly handling or disposal of specifically licensed sealed sources can be used to update the "realistic" parameter values used.*

See response to S1, and the internal dosimetry section of the additional risk information in Appendix ■.

S10. Page 6, 2.3.2: *For extensively damaged devices with wide dispersion and many people involved, the following scenarios were described: Incinerator with release to the environment, incinerator with subsequent burial in landfill, metal recycling into consumer products, and metal recycling into construction materials. One very significant set of dose scenarios that were left out of this study were the dose to workers at the incinerators or melters. Generic exposure and pathway analyses performed by PNL for US DOE for incineration of hazardous materials and for smelting associated with recycling and reuse of contaminated materials have shown that, in many cases, the limiting scenarios and doses were found to be those specified for workers.*

The ORAU authors stated that worker doses were considered as and lumped in with the doses to members of the public.

S11. Page 6, 2.3.2, ¶ *Information on incineration from NUREG/CR-1775 was used as last basis for this study. Assumptions that are needed are stated to be (1) the number of devices incinerated per year, (2) the fraction of activity released by the incineration process, and (3) the fraction of activity released that escapes into stack emissions. Five devices incinerated per year was chosen as the value used in this study. All other dose calculations in the report are based on one device. The assumed values taken for the three major parameters need to be justified, especially the "arbitrarily chosen" value of 5 devices incinerated per year. Also, several parameter values that may have a major impact on the resulting dose estimates that were not discussed are the throughput of the incinerator, the feed composition, the incineration temperature, and the final population size considered.*

The arbitrarily chosen value of 5 devices/year was chosen only as a benchmark value for this study. No data to support this selection was given.

S12. Page 6, 2.3.2, ¶ Dose calculation methods similar to those from NUREG/CR-1775 last were used in this study. The methods stated in NUREG/CR-1775 do not state that food pathways were considered in the analysis of landfills, thus it is assumed that they were not considered in this study either. Other studies have calculated estimates of doses to the public from radionuclides present in landfills. In some cases, the highest potential doses were found to result from intruder scenarios and well drilling scenarios some years after a landfill has closed. Both external and internal doses result from scenarios describing a residence being built on a landfill, where excavation is used for a basement and the soil is spread out over the yard and used for residential gardens. Doses can also result when a core is brought up from well-drilling on the site and dispersed on the surface. For large scale landfill sites, animal pathways may be needed (i.e., radionuclide uptake by plants, plants eaten by animals, animals used as food for humans living nearby).

More recent computer codes can be used to perform the exposure and pathway analyses. For instance, the GENII code and its many modules have been used extensively for dose calculations using many different types of scenarios associated with landfills and incinerators. External and internal doses can be obtained; air, water, and plant and animal food pathways are included. The GENII-S code incorporates a Latin Hypercube uncertainty and sensitivity analysis capability into the GENII code. The EPA's code, CAP88-PC, can be used for modeling the transport and fate of radioactive air emissions and doses to the public surrounding incinerators. This code also includes the food pathways in the dose calculations. Use of such codes should provide more "realistic" dose estimates.

The authors of the 1987 ORAU report they realize that they were not experts in using the dispersion and groundwater models, and that they probably did not use the most up-to-date models for the 1987 time frame. They stated that now they realize the project would have benefitted from having specialists in these areas involved on the project.

S13. Page 7, 2nd ¶: The incineration of medical wastes was used to select values for the last two parameters described in the comment above. The incineration process used for medical wastes may be quite different than that used for municipal wastes (i.e., which is a more likely occurrence for GLDs). Different fractions of the radionuclides can be released through stack emissions under the different incineration conditions. Thus, more research into appropriate partitioning values is needed.

The IAEA (IAEA Safety Series 111-P-1.1 [IAEA 1992]) has taken one

approach to this problem of lack of knowledge about specific radionuclide partitioning by conservatively assuming that 100% of the initial radioactivity is retained in all three resultant phases. For incinerator scenarios, these phases are slag, fly ash, and flue gases in stack emissions. This triple accounting approach will maximize the potential importance of the scenarios associated with each possibility. A similar approach should be considered as a possible way to estimate the effect of radionuclide partitioning when no or limited partitioning data is available.

A different approach that may be used is a slight variation of the IAEA method. An assumption is made that 100% of the radionuclides will end up in each of the three resulting incinerator phases. But, instead of summing the dose results from all three resultant phases, only the maximum dose for a given resultant phase is chosen and reported.

The ORAU authors stated that most of the literature available at the time the report was prepared dealt with medical incinerators. Thus, the only parameter values they felt comfortable in using were taken from those pertaining to the available literature.

S14. Page 7, last ¶: A metal recycling scenario was only considered for ^{60}Co . However, for a more realistic assessment, any of the materials could possibly find their way to a melter. Thus, all nuclides involved for GLDs should be considered in the dose assessments.

The ORAU authors agreed that the recycling section of the 1987 ORAU report needs to be updated.

S15. Page 8, 2nd ¶: Inhalation of suspended particles is not considered as an exposure pathway for the landfill burial scenario. As mentioned above in other comments, this scenario should be included for both worker scenarios, intruder scenarios, and well-drilling scenarios. For these scenarios, inhalation may be the major exposure pathway leading to internal doses for some of the radionuclides considered.

The ORAU authors contend that they relied heavily on NUREG/CR-1775 as a general basis for their report. Since this particular exposure scenario was not included in that report, they did not feel it necessary to include it in the 1987 ORAU report.

S16. Page 8, last ¶: Values for leach rates from landfills are given. More up-to-date information on nuclide-specific leach rates needs to be reviewed and

considered.

See response to S12.

S17. Page 9, 2nd ¶: *It is arbitrarily assumed that the dose to the average individual is one-half that of the MEI. The difference in dose to the average individual versus the dose to the MEI should be dependent on the radionuclide, its half-life, the uptake pathways considered, hold-up time through the scenarios, time in a given location, amount of a given food eaten per year, etc., and generally is not a constant number as assumed. A more in-depth development of scenarios and pathways is needed in this study.*

The ORAU authors agreed that the conversion to average individual dose from MEI dose should be a factor of 1/10 or less.

S18. Page 30, last ¶: *For internal dose calculations, a fraction of 0.3 (30%) was chosen as the maximum amount of the initial activity that can be taken in. Since this value can vary with radionuclide, the value(s) need to be validated and changed as needed.*

See response to S1.

S19. Page 30, last ¶: *It is stated that hydrogen gas is usually converted to tritiated water in the atmosphere, and thus is not considered in the dose calculations. Results of recent research indicate that this is not the case; elemental tritium gas (HT) is not readily converted to tritiated water in the atmosphere, but rather enzymes present in soil microbes are necessary for this conversion to readily occur. HT gas release experiments performed in Canada and France indicate that a significant portion of the dose resulting from a release of HT gas to the atmosphere is the secondary plume of tritiated water released from the soil after the primary plume of HT has contacted the soil. Also, a small portion of the HT gas that is inhaled is absorbed into the blood, and is converted to tritiated water by gut flora. The significant portion of the dose from inhalation of HT gas is associated with the tritiated water that is formed from the HT by this mechanism and should be considered. (NOTE: The ICRP has not yet accepted these items in its dose commitment scheme.)*

Alternatively, a review of the paper "Maximum Permissible Amounts of Accidentally Released Tritium from an Environmental Experiment to Meet Dose Limits for Public Exposure", by Taeschner, Bunnenberg, and Gulden, in Fusion Technology, August 1991, reporting on the 1986

French experiments found that the authors contend that the kinetics of the reaction,



avored the formation of tritiated water (HTO) in all environments with sufficient mass of water available, including that found in moist air. Following an HT release under dry air conditions, the absorption of HT in the soil, conversion to HTO, and re-release to the air will dominate the overall exposure to tritium (the dose due to exposure to the initial HT plume will be several orders of magnitude lower). Under moist conditions, the overall exposure will consist of roughly equal parts exposure to the initial HTO plume (following rapid conversion of HT to HTO in the atmosphere) and exposure to HTO absorbed in the soil and reemitted to the atmosphere over the following few days.

No response was made by the ORAU authors on this comment.

S20. Page 32,
2nd ¶: The "realistic individual" doses from tritium are calculated assuming uptake of 10%. This value needs to be researched and updated. This comments applies generally to all radionuclides considered in this study.

No response was made by the ORAU authors on this comment.

S21. Page 114: No scenarios are considered for the class J devices with natural uranium or thorium. However, the doses associated with these two radionuclides could be significant and should be determined.

No response was made by the ORAU authors on this comment.

B.3. MAJOR CONCLUSIONS PERTAINING TO THE REVIEW OF THE 1987 ORAU REPORT

The meeting with the authors of the 1987 ORAU report proved to be a significant help in identifying the constraints under which the study was done. The report was designed to address only the development of scenarios in which the public could receive doses from the improper transfer or disposal of generally licensed devices. It was their understanding that they were not charged with the task of establishing the total number of reported incidents with these devices or with calculating doses specific to actual incidents involving these devices. As a result, we now understand that many of the parameter values in question during this review were only used as benchmark values by the ORAU authors with the knowledge that more definitive values would be needed for the next phase of the analysis.

One item not addressed above is that while the committed dose equivalents to organs and

tissues (the units used in the ORAU report) is important, doses should be reported in units of committed effective dose equivalent (CEDE) and total effective dose equivalent (TEDE = sum of external and internal doses). This would have made comparison of the resultant doses between the different devices much easier.

Even with these caveats, it is still very difficult to justify the use of 0.30 as the maximum fraction-taken-in value used in the 1987 ORAU report for calculation of internal doses. First, if only worst case scenarios were to be considered, the maximum fraction-taken-in should be 1.0, that is assume that all of the source is taken into the body. Second, as discussed in Appendix B, in the review of reports of over 60 actual accidents, in no case did an intake fraction ever reach the maximum 30% level used in the ORAU study. Actual observed values ranged from $2\text{E}-4$ to $2\text{E}-8$ for individuals involved in the accidents, and ranged from $7\text{E}-11$ to $2\text{E}-15$ for the Goiânia cleanup workers (IAEA 1998b). If the doses for the various scenarios were recalculated using a defensible range of values for the fraction-taken-in, with all other parameter values the same, the CEDE and TEDE will be significantly lower than those that would result from using 0.30 value.

B.4. GENERAL LISTING OF NEW DEVELOPMENTS IN INTERNAL DOSIMETRY SINCE THE 1987 GLD REPORT

- *There still is no regulatory limit established for collective dose to either workers or the public.*
- *The revision to NRC 10CFR20 published in Federal Register and goes into effect January 1, 1994. This updates the methodological basis for the report dose limits from ICRP 2 (1959) to ICRP 26/30 (1977;1979). The dose limits in units of CEDE and TEDE.*
- *Age-dependence of doses is being looked at more closely*
- *Refinements to biokinetic models (eg. 1992 Leggett Air model, etc.) have been made*
- *New lung models are being developed by ICRP and NCRP subcommittees*
- *ICRP60/61 methodologies (1991) changed recommended dose limits to 20 mSv per year for occupational exposures and 1 mSv in a year for exposures to the public. The ICRP 26/30 limits were 50 mSv per year for occupational exposures.*

APPENDIX C: COMMENTS ON THE ORAU REPORT BY J. S. DUKELOW

Original Review Comments appear in the Appendix in italics. Information gained during the October 1993 visit to Oak Ridge to confer with the authors of the ORAU report appears interspersed in standard non-italic font. Additional relevant information gained since the October trip appears in a separate section at the end of this Appendix, also in non-italic font.

C.1. GENERAL COMMENTS

Page i -- The appropriate "worst case" assumption for exposure to the external radiation field probably ought to be direct contact of the encapsulated source with the body for some number of hours. This case is plausible and is likely to produce more severe consequences than the report's assumption of 20 weeks at 100 cm (particularly for alpha and beta sources).

The report assumes a "worst case" for ingestion as ingestion or inhalation of 30% of the radioactive. No justification is given for using 30%; an obvious "worse" than worst case is inhalation or ingestion of 100% of the radioactive material. Assuming 100% is not unreasonable if one remembers that some of the previous sealed source incidents involved sources falling into the hands of small children.

We still consider these comments to be reasonable. For many of the sealed source accidents described in the body of the present report, the victim "finding" the sealed source has put it in his pocket. For the ingestion cases, several of the accidents have involved small children finding and playing with sealed sources or sealed source material. A soft beta sealed source would require a delicate "shield window", which would be unlikely to survive stomach acid. On the other hand, the analysis provided in the text of the present report establishes that these worst case assumptions are not at all representative of exposures resulting in the known sealed source accidents.

Page 2 -- Some additional explanation is needed of the interior structure of Figure 1. For instance, one could argue that the "incinerator" block belongs on the downstream side of "salvage dealer" in addition to or instead of the upstream side. One of the things a salvage dealer might do with materials not deemed of interest for metal recovery is incineration. On the other hand, the pathway "trash handler to incinerator to salvage dealer" presumes that the trash handler sends to the salvaged dealer incinerator ashes containing intact or dispersed sources, all of which seems implausible.

Page 3 -- The "probabilities" described in Section 2.2 are properly called "conditional probabilities". Also, Section 2.2 ought to say that the probability of reaching a final state by way of a particular pathway is the product of all the pathway segment conditional probabilities and that the probability of reaching the final state is the sum of all those pathway probabilities.

The fact that the pathway segment conditional probabilities are all (or mostly) "educated

guess" deserves more emphasis, as well as some description of the basis used to make those educated guesses.

Page 5 -- There is a discussion of increasing published dose rates by 25-45 per cent to account for "electron production in the stainless steel walls" of the capsule (presumably Compton scattering, pair production, and photoelectric absorption). Is this the entire rationale or should it include: "bremsstrahlung radiation resulting from the deceleration of beta particles, electrons, and positrons in the stainless steel"?

The second paragraph refers to dose rates at a "depth of 7 mg/cm²". Cember's phrase "density thickness" is more understandable; at any rate, some explanation should be provided for the reader who is not a health physicist.

The last two paragraphs of Section 2.3.1 discuss internal dose resulting from "intact" sources. How does this differ from the internal dose resulting from dispersed sources? Wilmot's 1981 report on spent fuel transportation accidents may not be a good guide for release fractions for improper handling of sealed sources because of the significant differences in the barriers to release and in the purposeful nature of behavior that can be assumed in sealed source mishandling incidents.

Page 7 -- No basis is given for the assumption that the population near the incinerator is 73000.

Page 9 -- The population of the US is 250M, so the average population served by a landfill is $250000000/18500 = 13500$.

Page 13, Table 4 -- The table would be clearer if the last column were "Fraction" instead of "Percentage". Thus, for Class A-1, the last column would show 5.0×10^{-5} .

Table 4 suffers from a significant censoring problem. It does not reflect those sources that have been mishandled in some fashion, but for which the mishandling has not been detected. This is significant, because in known incidents, the detection was either accidental or announced by the severe radiological consequences. Thus, the portion of the sample space that leads to low consequences will be significantly under-represented in Table 4, which is nonetheless used to assign probabilities to event initiators.

The text of the present report discusses the censoring issue in more detail and proposes a methodology for estimating the amount of censoring in the available data, and thus, for arriving at estimates of the initiating event probabilities for various types of diversion or mishandling of Generally Licensed Sources.

C.2. WHAT IS NEEDED FOR REGULATORY APPLICATION, BUT MISSING FROM THE ORAU REPORT?

The main thing missing from the ORAU report, but required for regulatory application is a set of realistic estimates of the risk associated with the production and use of generally licensed sources. To obtain these realistic estimates of risk, we will need to make some realistic estimates of the frequencies associated with initiating events, the conditional probabilities associated with the transitions from state to state in the ORAU block diagrams, and best estimates of the mishandling incident source terms. Wilmot (1981) does not strike me as a good basis for those best estimate source terms, but I don't have an alternative to offer at the moment.

C.3 ADDITIONAL INFORMATION GAINED SUBSEQUENT TO OCTOBER MEETING WITH ORAU AUTHORS

The present report now provides a detailed proposal for the estimation of initiating event probabilities, taking the data censoring into account. These probabilities, together with best-estimate calculations of the public health consequences of a diverted/mishandled source ending up in a particular "final status", provide the basis for calculation of the public health risk associated with Generally Licensed Sources of a particular type.

Although we had criticized the ORAU report for not including the initiating event probabilities and realistic estimates of the public health consequences of a diverted/mishandled source reaching a specific "final status", it became clear during the October meeting that the ORAU report authors were aware of this difficulty. They had been specifically constrained by the Scope of Work on their project not to estimate those quantities. Such constraints are not unreasonable in a first-cut analysis of whether any changes are needed in the regulation of generally licensed devices; had all of the worst case consequences calculated by the ORAU authors been acceptably low, no further analysis would be needed to support a decision to maintain current regulation of generally licensed devices. With some of the worst case consequences being unacceptably high, it may be necessary to collect the additional data needed to calculate the risk, i.e., the probability-weighted consequences, of generally licensed device diversion/mishandling.

APPENDIX D: COMMENTS ON 1987 ORAU REPORT BY G.R. CICOTTE

D.1. REGULATORY CHANGES

The USNRC made no significant changes in its regulations affecting the quantities or categories of generally licensed radioactive sources since 1987. There have also been other changes in the reporting, record keeping, and enforcement portions of the applicable regulations. These are summarized as follows:

- *The new 10 CFR 20 (56 FR 23360-23472, May 21, 1991) resulted in changes to the incident reporting requirements of 10 CFR 30, 10 CFR 31, and 10 CFR 32, invoking a choice between the old part 20 (§20.1-20.601) and the new part 20 (§20.1001-20.2401).*
- *As part of the new part 20, general licensees were specifically limited to doses of 10% of the limits in either 10 CFR 20.101(a), i.e., 125 mrem (1.25 mSv) per calendar quarter, or 10 CFR 20.1201(a), i.e., 500 mrem (5 mSv) per calendar year.*
- *Record retention requirements were clarified (53 FR 19240-19246, May 27, 1988)*
- *The new incident reporting requirements summarized in 56 FR 40757-40767 were imposed on general licensees, by invoking 10 CFR 30.50 in 10 CFR 31.2, and specifically on general licensees for Am-241 calibration and reference sources, as specified in 10 CFR 31.8.*
- *Enforcement authority was placed on general licensees, specifically authorizing injunctions and orders deemed necessary by NRC, and providing criminal penalties for willful violations. (57 FR 55063-55075)*

D.2. EFFECTS ON CONCLUSIONS IN ORAU REPORT OF CHANGED AND EXISTING REGULATIONS

The changes in the regulations should result in reporting of a greater fraction of the incidents which occur. The ratios of incident causes appear to have changed as described below.

The regulations require that in order to use generally licensed devices in accordance with 10 CFR 31.5, the source must not be likely to result in a dose in excess of 125 mrem (1.25 mSv) per quarter, which correlates to 500 mrem (5 mSv) per year. The Executive Summary of the ORAU Report states in part that: "... dose equivalents received by members of the general public who purchase contaminated products would most likely not exceed 500 mrem/yr (0.005 Sv/yr) in most cases." No further action would be necessary to protect the public if the goal is to meet current regulatory limits as called out in 10 CFR 20.101(a) or 10 CFR 20.1201(a). The change to the goal dose of 1 mSv per year to a member of the public may result in the need to regulate the manufacture of licensed sources to assure that exceeding 1 mSv per year

is unlikely.

D.3. OTHER CONSIDERATIONS RELATED TO THE ORAU REPORT

The ORAU Report addressed the following generally licensed uses:

1. *Certain measuring, gauging, or controlling devices as authorized by 10 CFR 31.5 - Po-210, Am-241, Ra-226, H-3, Kr-85, Co-60, Cs-137, Sr-90, Tl-204, Ru-106, Pm-147, C-14, Pb-210, Ni-63, Cm-244, Cd-109, and Fe-55.*
2. *Luminous safety devices for use in aircraft as authorized by 10 CFR 31.7 - H-3 and Pm-147.*
3. *Calibration or reference sources as authorized by 10 CFR 31.8 - Am-241.*

The ORAU Report did not appear to address the following generally licensed uses:

1. *Isotopes [byproduct material for certain in vitro clinical or laboratory testing] as authorized by 10 CFR 31.11(a)(1) through (7).*
2. *Ice detection devices containing Sr-90 as authorized by 10 CFR 31.10(a).*
3. *Beta- and/or gamma-emitting materials in measuring, gauging or controlling devices containing radioisotopes other than those specifically listed in Table 1 of the ORAU Report, authorized in 10 CFR 31.5.*
4. *Alpha-emitting materials in measuring, gauging or controlling devices containing radioisotopes other than U-238, Am-241, Ra-226, or Pu-239, as authorized in 10 CFR 31.5.*

It may be appropriate to consider whether scenarios for items 1 and 2 above should be addressed in any report developed for the same purpose as the ORAU Report.

Generally licensed sources authorized in 10 CFR 31.5 are constrained to meet certain dose limits, even if damaged or misused. The general license in 10 CFR 31.5 requires that the sources must have been manufactured or produced through adherence to a specific license issued pursuant to the conditions specified by 10 CFR 32.51. 10 CFR 32.51 requires that individual sources must not cause doses to any person [not necessarily just a member of the public] in excess of the organ doses listed in 10 CFR 32.24. The organ dose limits have not changed since June 13, 1969, and are summarized below:

<i>Part of Body</i>	<i>normal use of one unit (rem)</i>	<i>normal use of all units in one location (rem)</i>	<i>failure - low probability of dose (rem)</i>	<i>failure - negligible probability of dose (rem)</i>
<i>Whole body; head and trunk; active blood- forming organs; gonads; or lens of eye.....</i>	<i>0.001</i>	<i>0.01</i>	<i>0.5</i>	<i>15</i>
<i>Hands and forearms; feet and ankles; localized areas of skin averaged over areas no larger than 1 square centimeter.....</i>	<i>0.015</i>	<i>0.15</i>	<i>7.5</i>	<i>200</i>
<i>Other organs.....</i>	<i>0.003</i>	<i>0.03</i>	<i>1.5</i>	<i>50</i>

D.4. REFINEMENT OF SELECTED SCENARIOS IN THE ORAU REPORT

The ORAU Report hypothesizes a maximum skin contact time of three hours (2.3.1). The limited time was based on the assumption that most of the devices cannot be carried in an individual person's pocket due to size considerations. The great majority of reported incidents involving generally licensed devices were related to portable static eliminators (task 3 report), which are often small enough to be carried in a pocket, e.g., recording equipment static eliminator brushes.

The dispersed source scenario assumes that dispersed material in a landfill would, if previously incinerated, leach into surface or ground water supplies at a rate of 1.0 per year (2.3.2). Since most landfills now have leaching requirements which result in much greater time for breakthrough and there is some natural filtration by the landfill material itself, inclusion of this consideration could reasonably be expected to reduce the available dose significantly.

The metals recycling section of the ORAU Report assumes continuous contact over a year for a postulated maximum dose of 360 millirem (3.6 mSv) based on a distance from contact which reduces the dose to 0.1 of that calculated for continuous contact. In addition, Table 6 of the ORAU Report states that whole body exposures to all categories are based on 20 weeks at 100 cm distance. A better approach might be to assume continuous contact over a reasonable fraction of the time, with the appropriate distance factor. Two possibilities are household products:

1. *Furniture - assume an occupancy factor combined with a distance factor, and account for some shielding by protective coatings such as paint (for beta emitters).*
2. *Hand-held appliances - assume contact for a postulated usage time, e.g., 30 minutes per day in contact with the ORAU projected dose rate of 410 μ R/h for Co-60.*

When asked about the 20 weeks at 100 cm criterion, the ORAU teams indicated that it was based on an analysis of the Juarez, Mexico accident, which indicated that a fifth of the time was spent in that configuration.

The scenario development section for static eliminators (3.1.2) asserts that static eliminators are unlikely to be recycled. In general, recycling has increased since 1987, and is expected to continue to increase (although remaining at a relatively low fraction of total recyclables), resulting in more of these devices entering the recycle scenario.

D.5. ADEQUACY OF THE ORAU STUDY

Given the information on which the ORAU study was based, the conclusions regarding doses appeared adequate, with some exceptions. However, there are two additional caveats to this statement. First, the study clearly states that a large number of assumptions were conservatively made, due to lack of data regarding specific disposition pathways. Second, the study clearly states that the "worst case" scenarios are excessively conservative, e.g., in section 3.7.3.1, the report states an assumption of ingestion/inhalation of 30% of the activity after damage to the source, even though the report considered the possibility of access to the source to be "extremely unlikely."

D.6. NEED FOR REGULATORY ACTION

In the opinion of this reviewer, there are two major considerations regarding the need for regulatory action. The first is related to the reliability of ORAU study estimates. The second is related to the change in public dose. The change to occupational doses do not appear to reflect a need for regulatory action.

1. *Based on the number of assumptions which were required to be made, the estimates given are significantly in doubt, in particular as they relate to the probabilities of misuses resulting in skin contact or internal exposure. That is, the estimates cannot necessarily be given sufficient credence to justify regulatory action based only on the statement of a high estimate. In order to use the conservative values in the ORAU study as the basis for regulatory action, it would be necessary to confirm as realistic information the estimates given, or determine representative values with greater assurance as to accuracy.*
2. *The public dose limit is introduced in 10 CFR 20.1301, "Dose limits for individual*

members of the public." The new limit is 100 mrem (1 mSv) total effective dose equivalent (TEDE). Thus, many of the conclusions in the ORAU study related to the old limit would need to be reassessed relative to the new limit. Estimates meeting all the following (or similar) criteria would need to be addressed through additional restrictions on generally licensed quantities or categories:

- a. The estimated TEDE exceeds the limit of 1 mSv.
- b. The probability of occurrence of the scenario is accurate within a margin which indicates the limit would be exceeded with a specified confidence level, e.g., when the estimate exceeds the limit by a factor of 2, the accuracy is within a factor of 2 to a specified confidence of 95%.
- c. The dose estimate for the pathway itself is accurate within a margin which indicates the limit would be exceeded with a specified confidence level, e.g., when the estimate exceeds the limit by a factor of 2, the accuracy is within a factor of 2 to a specified confidence of 95%.
- d. The combined effect of b. and c. is accurate within a margin which indicates the limit would be exceeded with a specified confidence level, e.g., when the estimate exceeds the limit by a factor of 2, the accuracy of b. and c. combined is within a factor of 2 to a specified confidence of 90%.
- e. Any arbitrary assumptions made in the estimate have been supplanted by estimates based on actual information.

D.7 ADDITIONAL INFORMATION OBTAINED IN MEETING WITH ORAU TEAM AND SUBSEQUENTLY

The ORAU team considered the pathways represented in the probability networks used in their report to represent a condensation of hundreds or thousand of possible pathways. Their intent was to display only the pathways responsible for the "first" 95% or so of the risk.

When asked why Po-210 was considered a significant hazard, given its short half-life (138 days), the ORAU team responded that it was considered an ingestion hazard due to the large number of Po-210 source and the loose control exercised over them.

The ORAU team indicated that sensitivity analysis was not part of the scope of their project.

When asked what they intended by the term "intact" sources, the ORAU team indicated that it varied somewhat, on a case-by-case basis, but that it was generally synonymous with "localized". External dose was not calculated for dispersed sources, with the exception of recycled material, such as scrap.

The ORAU team indicated that the "average" dose for intact sources was arbitrarily defined to be half of the worst-case dose.

The ORAU team was asked how population statistics were applied to external sources. They indicated that maximum and average population figures were applied only to dispersed sources (i.e., no external dose was assumed for populations, except for the case of recycled material).

The ORAU team was asked to describe any special considerations applicable to their dose calculations: 1) absorption was not used, 2) Maximum external dose was assumed to be 0.3 times the worst case for an intact source, 3) the most likely external dose was assumed to be 10^{-6} times the worst case for an intact source, 4) encapsulated source doses were calculated based on NCRP 40, and 5) non-encapsulated source dose were calculated based on beta dose, using the methodology of Kocher and Eckerman.

The assumptions in Section 2.3.2 of the ORAU report are based on NUREG/CR-1775.

The symbol nC in Section 2.3.2.2 of the ORAU report refers to nanoCoulombs; the distance 0.1 is in meters.

Re Section 2.3.2.3: Intrusion at landfills was not considered. The value $e^{-\lambda t}$ refers to the mean leach rate.

Re Section 2.3.1: The 50 year dose equivalent from ICRP 30 did not refer to the source remaining in place.

APPENDIX E TO
FINAL REVIEW OF THE 1987 REPORT BY
OAK RIDGE ASSOCIATED UNIVERSITIES,
"IMPROPER TRANSFER/DISPOSAL SCENARIOS FOR
GENERALLY LICENSED DEVICES:"

REVISED TASK 3 TECHNICAL LETTER REPORT:
EVALUATION OF HISTORICAL SEALED SOURCE DEVICE EXPERIENCE

NRC JOB CODE L2536
PNL No. 20278

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1.0 EXECUTIVE SUMMARY

Data and records provided to the Pacific Northwest Laboratory (PNL) by Steven L. Baggett and Sterling Bell of the U.S. Nuclear Regulatory Commission (NRC) have been reviewed to establish the historical experience of sealed source device use and reported events of improper transfer or disposal.

2.0 SUMMARY OF FILES AND REPORTS USED IN THIS WORK

Documents supplied to PNL are listed in Section 6.0, References. The original work under review in this project is referred to as the "ORAU Report" (Stabin et al. 1987).

Computer database files (in dBase III format for DOS) received and reviewed by PNL are listed in Attachment 1.

3.0 NUMBER OF SEALED SOURCE LICENSES AND DEVICES OF EACH TYPE EXISTING IN EACH YEAR FOR WHICH RECORDS ARE READILY AVAILABLE

3.1 REVIEW OF SEALED SOURCE DEVICE REGISTRY

The Sealed Source Device Registry (SSDR) provided basic information about the design, construction, uses and authorized maximum activity for each type of device. Devices are categorized in the fashion of Table 1 of the ORAU Report, included here as Attachment 2.

For detailed, source-specific risk analyses, it would be necessary to determine the numbers of sources in each category by isotope, date placed in service, and activity. Use of design information regarding shielding would help to determine external doses in cases of improper transfer or disposal in which the source was not removed from the shield. Review in this level of detail was beyond the scope of the present project.

3.2 REVIEW OF NUMBER OF GENERALLY-LICENSED DEVICES

Four hardcopy reports from "General License Database System" were provided to the PNL reviewers attached to a letter from Steven L. Baggett to Daniel J. Strom dated November 2, 1993 (Baggett 1993). Two of the reports were entitled "General License Database System Report for Peer Review." The first of these, dated 10/20/93, was 6 pages long and contained sealed source registrations in the years 1987-1992. The second, dated 10/20/93, was 2 pages long and contained additional sealed source registrations in the years 1991-1992. Column headings on the two reports were year, isotope, device code, number of devices, number of general licensees, total activity, and average activity/device. (Device codes are shown in Table Task-3-2. Activities are in millicuries (mCi).)

These two reports were keyed and analyzed as a review of historical data for a revision of the Task 3 report. The ORAU report was limited to devices of less than 20 mCi, but the "General License Database System Report for Peer Review" printouts contained many sources of activities significantly greater than 20 mCi, so these data are difficult to use directly in a risk analysis.

These two printouts are summarized over the 6-year period in Tables Task-3-1A, Task-3-1B, and Task-3-1C. In each table, the Device Codes are those identified in the printouts as being licensed under 10 CFR 31.5 and listed in Appendix C of Regulatory Guide 10.10 (NRC 1987), with "W7" denoting self-luminous sources licensed under 10 CFR 31.7. The tables are identical except for the sort order. Rows are labeled by nuclide and device code. Rows in Table Task-3-1A are sorted by nuclide (alphabetically, the way the data were received) and within nuclide by Device Code; rows in Table Task-3-1B are sorted by Device Code and within Device Code by nuclide in order of increasing atomic number; and rows in Table Task-3-1C are sorted by fraction of total ingestion ALIs contributed by a nuclide-Device Code combination.

In order to get an idea of the steady-state activity for sources, the activities of sources for years 1987-91 were decay-corrected to 1992. For some sources, e.g., 138-day ^{210}Po , this means that the 1987 sources had essentially disappeared. Each table shows the total number of devices registered between 1987 and 1992, the decay-corrected sum of the activities by nuclide and device code.

To rank the relative hazards of the nuclide-device code combinations, the ingestion ALIs for each row were divided by the total number of ingestion ALIs contained in sources registered during the 6-year span. Using the decay correction, there were 3.37×10^9 ingestion ALIs in these 325,681 sources in 1992. The fractions are expressed in parts per million (ppm) to make the numbers easier to compare.

Table Task-3-1C shows that the ^{241}Am D (gamma gauge) sources account for nearly half of the ingestion ALIs (489,184 ppm or 49%), with ^3H W self-luminous sources accounting for 26% of the ALIs (using the admittedly incorrect, that is, too high by one or more orders of magnitude, ALI for $^3\text{H}_2\text{O}$, not the unlisted ALI for ^3H -labeled luminous materials). Promethium-147 E gauges account for 9%, followed by ^{244}Cm U devices (4.1%), ^{252}Cf E and U devices (2.9% and 2.6%), and ^{210}Po static eliminators (2.5%). Tritium self-luminous light sources licensed under 10 CFR 31.7 account for about 1%, as do ^3H gas sources.

Also shown in the tables are crude risk assessment numbers in the last 5 columns: the average number of ingestion ALIs per device; the average activity per device; the average source strength, Γ_A , in rems per hour at 1 meter from an unshielded source with the average activity; the committed effective dose equivalent from ingesting 1/10,000 (10^{-4}) of the source; and the dose equivalent one would receive by spending 1000 hours at 1 meter from an unshielded source. A justification for use of the factors, 10^{-4} fraction-taken-in and 1000 hours used in the last two columns, is given in the main body of this report.

A number of difficulties were found in the SDDR printouts. These included invalid isotopes ("K85," "KR-95," "KR84 4," "CE137," "CS137 SR"), invalid device codes (GAUGE, blank, V, and 31.7U), and an invalid number (0) of general licensees that occurs on 6 occasions. The obvious corrections were made for these cases (K85 changed to Kr-85; zeros changed to ones for general licensees). There were also some puzzling uses of isotopes which may be explainable as keying errors: ^3H , ^{55}Fe , and ^{90}Sr gamma gauges; ^{55}Fe , ^{60}Co , ^{137}Cs , ^{210}Po , and ^{241}Am beta gauges; ^{63}Ni and ^{137}Cs neutron sources; Device Code I sources with activities less than 30 mCi; and ^{90}Sr gas sources. There was no basis for correcting these latter errors, so they remain as reported when included in the Tables Task-3-1A and 1B.

Table Task-3-1A. Summary of "General License Database System Report for Peer Review." Sorted by Atomic Number, Device Code.

Nuclide	Dev - Ice Code	Device Definition	'87 thru '92		sum of Fractions of Total Ing. ALIs	Average Ing. ALIs/ Device	Average Activity/ Device (mCi)	Average Source Strength, Gamma*A (rem/h @1m)	Dose from Ingestion of 10 ⁻⁴ of Source (rems)	Dose Specifying 1000 hours at 1 m from Source (rems)
			Total Number of Devices, '87-'92	Sum of Activity decayed to 1992 (mCi)						
H-3	D	Gamma Gauges	1178	1.38E+07	1,364	3,909	11,727	-	2.0	-
H-3	E	Beta Gauges	2390	2.47E+07	2,438	3,439	10,317	-	1.7	-
H-3	N	Ion Gen, Chromatog	1378	5.82E+08	556	1,360	4,079	-	0.68	-
H-3	O	Ion Gen, Static Elim.	1392	1.51E+07	1,494	3,617	10,852	-	1.8	-
H-3	R	Gas Sources	10771	1.02E+08	10,066	3,150	9,451	-	1.6	-
H-3	S	Foil Sources	25	1.25E+03	0.12	17	50	-	0.0083	-
H-3	T	Other	20	2.93E+02	0.029	5	15	-	0.0024	-
H-3	W	Self-Lum Light Src	242505	2.66E+08	262,841	3,654	10,961	-	1.8	-
H-3	W7	31.7 Self-Lum Light Src	18519	1.07E+08	10,596	1,929	5,786	-	1.0	-
C-14	E	Beta Gauges	6	6.00E-01	0.0020	1	0.10	-	5.55E-04	-
C-14	N	Ion Gen, Chromatog	10	1.00E-02	0.000033	0	0.0010	-	5.55E-06	-
C-14	R	Gas Sources	42	2.10E+00	0.0069	1	0.050	-	2.78E-04	-
C-14	T	Other	33	7.25E+00	0.024	2	0.22	-	0.0012	-
C-14	W	Self Lum Light Src	267	3.87E+01	0.12	2	0.14	-	7.64E-04	-
C-14	Y	Calibrators	24	1.20E-01	0.00040	0	0.0050	-	2.78E-05	-
Sc-46	N	Ion Gen, Chromatog	167	4.91E+02	0.16	3	2.9	0.0034	0.0016	3.4
Ti-44	N	Ion Gen, Chromatog	13	6.43E+02	0.64	165	49	0.0072	0.082	7.2
Fe-55	D	Gamma Gauges	57	1.10E+03	0.036	2	19	-	0.0011	-
Fe-55	E	Beta Gauges	13	4.84E+02	0.015	4	38	-	0.0020	-
Fe-55	N	Ion Gen, Chromatog	18	7.67E+02	0.025	5	43	-	0.0024	-
Fe-55	T	Other	12	5.60E+02	0.018	5	47	-	0.0026	-
Fe-55	U	X-Ray Fluorescence	811	6.51E+04	2	9	80	-	0.0045	-
Co-60	D	Gamma Gauges	41	4.13E+03	2	201	101	0.14	0.10	138
Co-60	E	Beta Gauges	3	3.52E+01	0.021	23	12	0.016	0.012	16
Ni-63	E	Beta Gauges	22	2.17E-01	7.15E-06	0	0.010	-	5.48E-07	-
Ni-63	H	Gen Neut Src Apps	22	3.30E+02	0.011	2	15	-	8.33E-04	-
Ni-63	N	Ion Gen, Chromatog	3451	2.59E+05	9	8	75	-	0.0042	-
Ni-63	O	Ion Gen, Static Elim.	56	8.28E+02	0.027	2	15	-	8.21E-04	-

Table Task-3-1A continued. Sorted by Atomic Number and Device Code.

Nuclide	Dev - ice Code	Device Definition	Total Number of Devices, '87-'92	Sum of Activity decayed to 1992 (mCi)	'87 thru '92 sum of Fractions of Total Ing. ALIs		Average Ing. ALIs/ Device	Average Activity/ Device (mCi)	Average Source Strength, Gamma*A (rem/h @1m)	Dose from Ingestion of 10 ⁻⁴ of Source (rems)	Dose from Spending 1000 hours at 1 m from Source (rems)
					Sum of Activity decayed to 1992 (ppm)	Total Ing. ALIs					
Ni-63	S	Foil Sources	8	6.99E+01	0.0023		1	12	-	6.47E-04	-
Ni-63	T	Other	12	1.19E-01	3.93E-06		0	0.010	-	5.52E-07	-
Ni-63	U	X-Ray Fluorescence	8	3.91E+01	0.0013		1	4.9	-	2.72E-04	-
Ni-63	W	Self-Lum Light Src	19	1.20E+05	4		702	6,320	-	0.35	-
Kr-85	D	Gamma Gauges	291	1.20E+05	0		0	413	0.00E+00	-	0.00E+00
Kr-85	E	Beta Gauges	623	2.43E+06	0		0	3,905	0.0061	-	6.1
Kr-85	N	Ion Gen, Chromatog	135	6.94E+04	0		0	514	8.05E-04	-	0.80
Kr-85	O	Ion Gen, Static Elim.	128	3.44E+02	0		0	2.7	4.21E-06	-	0.0042
Kr-85	R	Gas Sources	2	2.11E+03	0		0	1,054	0.0017	-	1.7
Kr-85	T	Other	11	2.20E+03	0		0	200	3.13E-04	-	0.31
Kr-85	U	X-Ray Fluorescence	13	5.44E+01	0			4.2	6.55E-06	-	0.0065
Sr-90	D	Gamma Gauges	104	2.28E+04	226		7,224	220	-	3.7	-
Sr-90	F	Beta Gauges	1153	3.94E+05	3,895		11,387	342	-	5.7	-
Sr-90	I	Calib Src A>30 mCi	1	5.00E-01	0.0049		17	0.50	-	0.0083	-
Sr-90	R	Gas Sources	135	2.44E+02	2		60	1.8	-	0.030	-
Sr-90	T	Other	182	8.63E+01	0.85		16	0.47	-	0.0079	-
Sr-90	Y	Calibrators	2	2.41E-03	0.000024		0	0.0012	-	2.01E-05	-
Ru-106	D	Gamma Gauges	2	4.00E+01	0.059		100	20	-	0.050	-
Ru-106	E	Beta Gauges	1	9.65E-04	1.43E-06		0	0.0010	-	2.41E-06	-
Cd-109	D	Gamma Gauges	44	5.27E+01	0.052		4	1.2	2.21E-04	0.0020	0.22
Cd-109	E	Beta Gauges	85	4.60E+02	0.45		18	5.4	0.0010	0.0090	1.0
Cd-109	N	Ion Gen, Chromatog	8	2.85E+01	0.026		11	3.3	6.11E-04	0.0055	0.61
Cd-109	T	Other	10	4.90E+01	0.048		16	4.9	9.03E-04	0.0082	0.90
Cd-109	U	X-Ray Fluorescence	567	6.25E+02	0.62		4	1.1	2.03E-04	0.0018	0.20
I-129	T	Other	24	2.96E+01	0.0029		0	1.2	1.55E-04	2.05E-04	0.16
Cs-137	D	Gamma Gauges	2493	2.20E+06	6,533		8,833	883	0.34	4.4	337
Cs-137	E	Beta Gauges	31	6.05E+05	1,795		195,228	19,523	7.5	98	7,455
Cs-137	H	Gen Neut Src Apps	45	3.56E+03	11		791	79	0.030	0.47	30

Table Task-3-1A continued. Sorted by Atomic Number and Device Code.

Nuclide	Dev - Ice Code	Device Definition	Total Number of Devices, '87-'92	Sum of Activity decayed to 1992 (mCi)	'87 thru '92 sum of Fractions of Total Ing. ALIs decayed to 1992 (ppm)	Average Ing. ALIs/ Device	Average Activity/ Device (mCi)	Average Source Strength, Gamma*A (rem/h @1m)	Dose from Ingestion of 10 ⁻⁴ at 1 m from Source (rems)	Dose from Spending 1000 hours at 1 m from Source (rems)
Cs-137	I	Calib Src A>30 mCi	17	6.49E+02	1.9	382	38	0.015	0.19	15
Cs-137	N	Ion Gen, Chromatog	11	3.13E+02	0.93	284	28	0.011	0.14	11
Cs-137	T	Other	1080	1.40E+06	4,139	13,163	1,316	0.50	6.6	503
Cs-137	U	X-Ray Fluorescence	18	2.60E+03	8	1,445	144	0.055	0.72	55
Cs-137	W	Self-Lum Light Src	5	2.25E+03	7	4,503	450	0.17	2.3	172
Cs-137	Y	Calibrators	109	4.64E+01	0.14	4	0.43	1.62E-04	0.0021	0.16
Ba-133	I	Calib Src A>30 mCi	93	1.53E+00	0.00023	0	0.016	7.51E-06	4.12E-06	0.0075
Ba-133	N	Ion Gen, Chromatog	5	8.79E-02	0.000013	0	0.018	8.01E-06	4.39E-06	0.0080
Ba-133	T	Other	644	5.88E+02	0.087	0	0.91	4.16E-04	2.28E-04	0.42
Pm-147	D	Gamma Gauges	11	6.20E+03	0.46	141	564	1.51E-06	0.070	0.0015
Pm-147	E	Beta Gauges	2685	1.09E+09	80,974	101,660	406,639	0.0011	51	1.1
Pm-147	T	Other	1	5.90E+02	0.044	147	590	1.58E-06	0.074	0.0016
Pm-147	W	Self-Lum Light Src	1	4.42E+04	3	11,055	44,220	1.18E-04	5.5	0.12
Eu-152	T	Other	56	1.08E+00	0.00040	0	0.019	1.44E-05	1.21E-05	0.014
Tl-204	E	Beta Gauges	538	4.05E+06	601	3,764	7,528	0.0084	1.9	8.4
Po-210	E	Beta Gauges	98	3.85E+01	4	131	0.39	2.07E-09	0.065	2.07E-06
Po-210	O	Ion Gen, Static Elm.	27235	2.55E+05	25,251	3,125	9.4	4.94E-03	1.6	4.94E-05
Po-210	U	X-Ray Fluorescence	535	9.23E+03	912	5,749	17	9.09E-08	2.9	9.09E-05
Bi-210	E	Beta Gauges	1	2.79E-89	1.04E-92	0	2.79E-89	-	1.75E-92	-
Ra-226	I	Calib Src A>30 mCi	1	2.00E-02	0.0030	10	0.020	2.85E-05	0.0050	0.029
Ra-226	S	Foil Sources	53	1.24E+00	0.18	12	0.023	3.34E-05	0.0058	0.033
Ra-226	T	Other	12	1.20E-01	0.018	5	0.010	1.42E-05	0.0025	0.014
Pu-238	D	Gamma Gauges	2	5.84E+01	19	32,425	29	0.0023	16	2.3
Pu-238	U	X-Ray Fluorescence	12	3.52E+02	116	32,620	29	0.0023	16	2.3
Am-241	D	Gamma Gauges	2261	1.32E+06	489,184	729,319	583	0.18	365	183
Am-241	E	Beta Gauges	82	7.89E+04	29,240	1,202,034	962	0.30	601	302
Am-241	H	Gen Neut Src Apps	3	1.36E+03	504	565,999	453	0.14	283	142
Am-241	I	Calib Src A>30 mCi	9	2.17E-01	0.080	30	0.024	7.55E-06	0.015	0.0076

Table Task-3-1A continued. Sorted by Atomic Number and Device Code.

Nuclide	Dev - ice Code	Device Definition	Total Number of Devices, '87-'92	Sum of Activity decayed to 1992 (mCi)	'87 thru '92 sum of Fractions of Total Ing. ALIs decayed to 1992 (ppm)	Average Ing. ALIs/ Device	Average Activity/ Device (mCi)	Average Source Strength, Gamma*A (rem/h @1m)	Dose from Ingestion of 10 ⁻⁴ of Source (rems)	Dose from Spending 1000 hours at 1 m from Source (rems)
Am-241	N	Ion Gen, Chromatog	1	1.30E-01	0.048	162	0.13	4.07E-05	0.081	0.041
Am-241	O	Ion Gen, Static Elim.	107	1.47E+03	545	17,184	14	0.0043	8.6	4.3
Am-241	S	Foil Sources	2	2.60E-01	0.10	162	0.13	4.07E-05	0.081	0.041
Am-241	T	Other	7	2.00E+01	7	3,572	2.9	8.98E-04	1.8	0.90
Am-241	U	X-Ray Fluorescence	350	6.97E+04	25,830	248,772	199	0.062	124	62
Am-241	Y	Calibrators	6	6.79E-05	0.000025	0	1.13E-05	3.55E-09	7.08E-06	3.55E-06
Cm-244	D	Gamma Gauges	18	2.16E+03	642	135,307	135	0.0087	68	8.7
Cm-244	U	X-Ray Fluorescence	239	1.35E+05	40,171	566,579	567	0.036	283	36
Cf-252	H	Gen Neut Src Apps	16	1.60E+00	0.24	50	0.10	4.18E-06	0.025	0.004
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Table Task-3-1B. Summary of "General License Database System Report for Peer Review." Sorted by Device Code and Nuclide.

Nuclide	Dev - ice Code	Device Definition	Total Number of Devices, '87-'92	Sum of Activity decayed to 1992 (mCi)	'87 thru '92 sum of Fractions of Total Ing. ALIs		Average Ing. ALIs/ Device	Average Activity/ Device (mCi)	Average Source Strength, Gamma*A (rem/h @1m)	Dose from Ingestion of 10 ⁻⁴ of Source (rems)	Dose from Spending 1000 hours at 1 m from Source (rems)
					decayed to 1992 (ppm)	decayed to 1992 (ppm)					
H-3	D	Gamma Gauges	1176	1.38E+07	1,364	3,909	11,727	-	2.0	-	-
Fe-55	D	Gamma Gauges	57	1.10E+03	0.036	2	19	-	0.0011	-	-
Co-60	D	Gamma Gauges	41	4.13E+03	2	201	101	0.14	0.10	138	-
Kr-85	D	Gamma Gauges	291	1.20E+05	0	0	413	1.53E-04	-	1.53E-01	-
Sr-90	D	Gamma Gauges	104	2.28E+04	226	7,324	220	-	3.7	-	-
Ru-100	D	Gamma Gauges	2	4.00E+01	0.059	100	20	-	0.050	-	-
Cd-109	D	Gamma Gauges	44	5.27E+01	0.052	4	1.2	2.21E-04	0.0020	0.22	-
Cs-137	D	Gamma Gauges	2493	2.20E+06	6,533	8,833	883	0.34	4.4	337	-
Pm-147	D	Gamma Gauges	11	6.20E+03	0.46	141	584	1.51E-06	0.070	0.0015	-
Pu-238	D	Gamma Gauges	2	5.84E+01	19	32,425	29	0.0023	16	2.3	-
Am-241	D	Gamma Gauges	2261	1.32E+06	489,184	729,319	583	0.18	365	183	-
Cm-244	D	Gamma Gauges	16	2.16E+03	642	135,307	135	0.0087	68	8.7	-
H-3	E	Beta Gauges	2390	2.47E+07	2,438	3,439	10,317	-	1.7	-	-
C-14	E	Beta Gauges	8	6.00E-01	0.0020	1	0.10	-	5.55E-04	-	-
Fe-55	E	Beta Gauges	13	4.64E+02	0.015	4	36	-	0.0020	-	-
Co-60	E	Beta Gauges	3	3.52E+01	0.021	23	12	0.016	0.012	16	-
Ni-63	E	Beta Gauges	22	2.17E-01	7.15E-06	0	0.010	-	5.48E-07	-	-
Kr-85	E	Beta Gauges	623	2.43E+06	0	0	3,905	0.0061	-	6.1	-
Sr-90	E	Beta Gauges	1153	3.94E+05	3,895	11,387	342	-	5.7	-	-
Ru-106	E	Beta Gauges	1	9.65E-04	1.43E-06	0	0.0010	-	2.41E-06	-	-
Cd-109	E	Beta Gauges	85	4.60E+02	0.45	18	5.4	0.0010	0.0090	1.0	-
Cs-137	E	Beta Gauges	31	6.05E+05	1,795	195,228	19,523	7.5	98	7,455	-
Pm-147	E	Beta Gauges	2685	1.09E+09	80,974	101,660	406,639	0.0011	51	1.1	-
Tl-204	E	Beta Gauges	538	4.05E+06	601	3,764	7,528	0.0084	1.9	8.4	-
Po-210	E	Beta Gauges	98	3.85E+01	4	131	0.39	2.07E-09	0.065	2.07E-06	-
Bi-210	E	Beta Gauges	1	2.79E-89	1.04E-92	0	2.79E-89	-	1.75E-92	-	-
Am-241	E	Beta Gauges	82	7.89E+04	29,240	1,202,034	962	0.30	601	302	-
Ni-63	H	Gen Neut Src Apps	22	3.30E+02	0.011	2	15	-	8.33E-04	-	-

Table Task-3-1B continued. Sorted by Device Code and Nuclide.

Nuclide	Dev - ice	Device Definition	Total Number of Devices, '87-'92	Sum of Activity decayed to 1992 (mCi)	'87 thru '92 sum of Fractions of Total Ing. ALIs decayed to 1992 (ppm)	Average Ing. ALIs/ Device	Average Activity/ Device (mCi)	Average Source Strength, Gamma*A (rem/h @1m)	Dose from Ingestion of 10 ⁻⁴ of Source (rems)	Dose from Spending 1000 hours at 1 m from Source (rems)
Cs-137	H	Gen Neut Src Apps	45	3.56E+03	11	791	79	0.030	0.40	30
Am-241	H	Gen Neut Src Apps	3	1.36E+03	504	565,999	453	0.14	283	142
Cf-252	H	Gen Neut Src Apps	16	1.60E+00	0.24	50	0.10	4.18E-06	0.025	0.0042
Sr-90	I	Calib Src A>30 mCi	1	5.00E-01	0.0049	17	0.50	-	0.0083	-
Cs-137	I	Calib Src A>30 mCi	17	6.49E+02	1.9	382	38	0.015	0.19	15
Ba-133	I	Calib Src A>30 mCi	93	1.53E+00	0.00023	0	0.016	7.51E-06	4.12E-06	0.0075
Ra-226	I	Calib Src A>30 mCi	1	2.00E-02	0.0030	10	0.020	2.85E-05	0.0050	0.029
Am-241	I	Calib Src A>30 mCi	9	2.17E-01	0.080	30	0.024	7.55E-06	0.015	0.0076
H-3	N	Ion Gen, Chromatog	1378	5.62E+06	556	1,360	4,079	-	0.68	-
C-14	N	Ion Gen, Chromatog	10	1.00E-02	0.000033	0	0.0010	-	5.55E-06	-
Sc-46	N	Ion Gen, Chromatog	167	4.91E+02	0.16	3	2.9	0.0034	0.0016	3.4
Ti-44	N	Ion Gen, Chromatog	13	6.43E+02	0.64	165	49	0.0072	0.082	7.2
Fe-55	N	Ion Gen, Chromatog	18	7.67E+02	0.025	5	43	-	0.0024	-
Ni-63	N	Ion Gen, Chromatog	3451	2.59E+05	9	8	75	-	0.0042	-
Kr-85	N	Ion Gen, Chromatog	135	6.94E+04	0	0	514	8.05E-04	-	0.80
Cd-109	N	Ion Gen, Chromatog	8	2.65E+01	0.026	11	3.3	6.11E-04	0.0055	0.61
Cs-137	N	Ion Gen, Chromatog	11	3.13E+02	0.93	284	28	0.011	0.14	11
Ba-133	N	Ion Gen, Chromatog	5	8.79E-02	0.000013	0	0.018	8.01E-06	4.39E-06	0.0080
Am-241	N	Ion Gen, Chromatog	1	1.30E-01	0.048	162	0.13	4.07E-05	0.081	0.041
H-3	O	Ion Gen, Static Elim.	1392	1.51E+07	1,494	3,617	10,852	-	1.8	-
Ni-63	O	Ion Gen, Static Elim.	56	8.28E+02	0.027	2	15	-	8.21E-04	-
Kr-85	O	Ion Gen, Static Elim.	128	3.44E+02	0	0	2.7	4.21E-06	-	0.0042
Po-210	O	Ion Gen, Static Elim.	27235	2.55E+05	25,251	3,125	9.4	4.94E-08	1.6	4.94E-05
Am-241	O	Ion Gen, Static Elim.	107	1.47E+03	545	17,184	14	0.0043	8.6	4.3
H-3	R	Gas Sources	10771	1.02E+08	10,066	3,150	9,451	-	1.6	-
C-14	R	Gas Sources	42	2.10E+00	0.0069	1	0.050	-	2.78E-04	-
Kr-85	R	Gas Sources	2	2.11E+03	0	0	1,054	0.0017	-	1.7
Sr-90	R	Gas Sources	135	2.44E+02	2	60	1.8	-	0.030	-

Table Task-3-1B continued. Sorted by Device Code and Nuclide.

Nuclide	Device Code	Device Definition	Total Number of Devices, '87-'92	Sum of Activity decayed to 1992 (mCi)	'87 thru '92 sum of Fractions of Total Ing. ALIs decayed to 1992 (ppm)		Average Ing. ALIs/ Device	Average Activity/ Device (mCi)	Average Source Strength, Gamma*A (rem/h @1m)	Dose from Ingestion of 10 ⁻⁴ of Source (rems)	Dose from Spending 1000 hours at 1 m from Source (rems)
H-3	S	Foil Sources	25	1.25E+03	0.12	17	50	-	0.0083	-	-
Ni-63	S	Foil Sources	6	6.99E+01	0.0023	1	12	-	6.47E-04	-	-
Ra-226	S	Foil Sources	53	1.24E+00	0.18	12	0.023	3.34E-05	0.0018	0.033	-
Am-241	S	Foil Sources	2	2.60E-01	0.10	162	0.13	4.07E-05	0.081	0.041	-
H-3	T	Other	20	2.93E+02	0.029	5	15	-	0.0024	-	-
C-14	T	Other	33	7.25E+00	0.024	2	0.22	-	0.0012	-	-
Fe-55	T	Other	12	5.60E+02	0.018	5	47	-	0.0026	-	-
Ni-63	T	Other	12	1.19E-01	3.93E-06	0	0.010	-	5.52E-07	-	-
Kr-85	T	Other	11	2.20E+03	0	0	200	3.13E-04	-	0.31	-
Sr-90	T	Other	182	8.63E+01	0.85	16	0.47	-	0.0079	-	-
Cd-109	T	Other	10	4.90E+01	0.048	16	4.9	9.03E-04	0.0082	0.90	-
I-129	T	Other	24	2.96E+01	0.0029	0	1.2	1.55E-04	2.05E-04	0.16	-
Cs-137	T	Other	1060	1.40E+06	4,139	13,163	1,316	0.50	6.6	503	-
Ba-133	T	Other	644	5.88E+02	0.087	0	0.91	4.16E-04	2.28E-04	0.42	-
Pm-147	T	Other	1	5.90E+02	0.144	147	590	1.58E-06	0.074	0.0016	-
Eu-152	T	Other	56	1.08E+00	0.00040	0	0.019	1.44E-05	1.21E-05	0.014	-
Ra-226	T	Other	12	1.20E-01	0.018	5	0.010	1.42E-05	0.0025	0.014	-
Am-241	T	Other	7	2.00E+01	7	3,572	2.9	8.96E-04	1.8	0.90	-
Fe-55	U	X-Ray Fluorescence	811	6.51E+04	2	9	80	-	0.0045	-	-
Ni-63	U	X-Ray Fluorescence	8	3.91E+01	0.0013	1	4.9	-	2.72E-04	-	-
Kr-85	U	X-Ray Fluorescence	13	5.44E+01	0	0	4.2	6.55E-06	-	0.0065	-
Cd-109	U	X-Ray Fluorescence	567	6.25E+02	0.62	4	1.1	2.03E-04	0.0018	0.20	-
Cs-137	U	X-Ray Fluorescence	18	2.60E+03	8	1,445	144	0.055	0.72	55	-
Po-210	U	X-Ray Fluorescence	535	9.23E+03	912	5,749	17	9.09E-08	2.9	9.09E-05	-
Pu-238	U	X-Ray Fluorescence	12	3.52E+02	116	32,620	29	0.0023	16	2.3	-
Am-241	U	X-Ray Fluorescence	350	6.97E+04	25,830	248,772	199	0.062	124	62	-
Cm-244	U	X-Ray Fluorescence	239	1.35E+05	40,171	566,579	567	0.036	283	36	-
H-3	W	Self-Lum Light Src	242505	2.66E+09	262,841	3,654	10,961	-	1.8	-	-

Table Task-3-1B continued. Sorted by Device Code and Nuclide.

Nuclide	Dev - ice Code	Device Definition	Total Number of Devices, '87-'92	'87 thru '92 sum of Fractions of		Average Ing. ALIs/ Device	Average Activity/ Device (mCi)	Average Strength, Gamma*A (rem/h @1m)	Dose from Ingestion of 10 ⁻⁴ of Source (rems)	Dose from Spending 1000 hours at 1 m from Source (rems)
				Sum of Activity decayed to 1992 (mCi)	Total Ing. ALIs decayed to 1992 (ppm)					
C-14	W	Self-Lum Light Src	267	3.67E+01	0.12	2	0.14	-	7.64E-04	-
Ni-63	W	Self-Lum Light Src	19	1.20E+05	4	702	6,320	-	0.35	-
Cs-137	W	Self-Lum Light Src	5	2.25E+03	7	4,503	450	0.17	2.3	172
Pm-147	W	Self-Lum Light Src	1	4.42E+04	3	11,055	44,220	1.18E-04	5.5	0.12
H-3	W7	31.7 Self-Lum Light Src	18519	1.07E+08	10,596	1,929	5,786	-	1.0	-
C-14	Y	Calibrators	24	1.20E-01	0.00040	0	0.0050	-	2.78E-05	-
Sr-90	Y	Calibrators	2	2.41E-03	0.000024	0	0.0012	-	2.01E-05	-
Cs-137	Y	Calibrators	109	4.64E+01	0.14	4	0.43	1.62E-04	0.0021	0.16
Am-241	Y	Calibrators	6	6.79E-05	0.000025	0	1.13E-05	3.55E-09	7.08E-06	3.55E-06
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Table Task-3-1C. Summary of "General License Database System Report for Peer Review." Sorted by "Fractions of Total Ing. ALIs."

				'87 thru '92 sum of Fractions of Total Ing. ALIs				Average Source Strength, Gamma*A (rem/h @1m)	Dose from Ingestion of 10 ⁻⁴ at 1 m from Source (rems)	Dose from Spending 1000 hours at 1 m from Source (rems)
Nuclide	Dev - Ice Code	Device Definition	Total Number of Devices, '87-'92	Sum of Activity decayed to 1992 (mCi)	Sum of Activity decayed to 1992 (ppm)	Average Ing. ALIs/ Device	Average Activity/ Device (mCi)			
Am-241	D	Gamma Gauges	2261	1.32E+06	489,184	729,319	583	0.18	365	183
H-3	W	Self-Lum Light Src	242505	2.66E+09	262,841	3,654	10,961	-	1.8	-
Pm-147	E	Beta Gauges	2685	1.09E+09	80,974	101,660	408,639	0.0011	51	1.1
Cm-244	U	X-Ray Fluorescence	239	1.35E+05	40,171	568,579	567	0.036	283	36
Am-241	E	Beta Gauges	82	7.89E+04	29,240	1,202,034	962	0.30	601	302
Am-241	U	X-Ray Fluorescence	350	6.97E+04	25,830	248,772	199	0.062	124	62
Po-210	O	Ion Gen, Static Elim.	2723	2.55E+05	25,251	3,125	9.4	4.94E-08	1.6	4.94E-05
H-3	W7	31.7 Self-Lum Light Src	18519	1.07E+08	10,596	1,929	5,786	-	1.0	-
H-3	R	Gas Sources	10771	1.02E+08	10,066	3,150	9,451	-	1.8	-
Cs-137	D	Gamma Gauges	2493	2.20E+06	6,533	8,833	983	0.34	4.4	337
Cs-137	T	Other	1060	1.40E+06	4,139	13,163	1,316	0.50	6.6	503
Sr-90	E	Beta Gauges	1153	3.94E+05	3,895	11,387	342	-	5.7	-
H-3	E	Beta Gauges	2390	2.47E+07	2,438	3,439	10,317	-	1.7	-
Cs-137	E	Beta Gauges	31	6.05E+05	1,795	195,228	19,523	7.5	98	7,455
H-3	O	Ion Gen, Static Elim.	1392	1.51E+07	1,494	3,617	10,852	-	1.8	-
H-3	D	Gamma Gauges	1176	1.38E+07	1,364	3,909	11,727	-	2.0	-
Po-210	U	X-Ray Fluorescence	535	9.23E+03	912	5,749	17	9.09E-08	2.9	9.09E-05
Cm-244	D	Gamma Gauges	16	2.18E+03	642	135,307	135	0.0087	68	8.7
Tl-204	E	Beta Gauges	538	4.05E+06	601	3,764	7,528	0.0084	1.9	8.4
H-3	N	Ion Gen, Chromatog	1378	5.62E+06	556	1,360	4,079	-	0.68	-
Am-241	O	Ion Gen, Static Elim.	107	1.47E+03	545	17,184	14	0.0043	8.6	4.3
Am-241	H	Gen Neut Src Apps	3	1.36E+03	504	565,999	453	0.14	283	142
Sr-90	D	Gamma Gauges	104	2.28E+04	226	7,324	220	-	3.7	-
Pu-238	U	X-Ray Fluorescence	12	3.52E+02	116	32,620	29	0.0023	16	2.3
Pu-238	D	Gamma Gauges	2	5.84E+01	19	32,425	29	0.0023	16	2.3
Cs-137	H	Gen Neut Src Apps	45	3.56E+03	11	791	79	0.030	0.40	30
Ni-63	N	Ion Gen, Chromatog	3451	2.59E+05	9	8	75	-	0.0042	-
Cs-137	U	X-Ray Fluorescence	18	2.60E+03	8	1,445	144	0.055	0.72	55

Table Task-3-1C continued. Sorted by "Fractions of Total Ing. ALIs."

		'87 thru '92					Average	Dose	Dose from	
		Total		sum of		Average	Source	from	Spending	
		Number	Sum of	Fractions of		Activity/	Strength,	Ingestion	1000 hours	
Dev -		of	Activity	Total Ing.		Device	Gamma*A	of 10^-4	at 1 m from	
Nuclide	Code	Devices,	decayed to	decayed to	Average Ing.	(mCi)	(rem/h	of Source	Source	
		'87-'92	1992 (mCi)	1992 (ppm)	ALIs/ Device		@1m)	(rems)	(rems)	
Am-241	T	Other	7	2.00E+01	7	3,572	2.9	8.96E-04	1.8	0.90
Cs-137	W	Self-Lum Light Src	5	2.25E+03	7	4,503	450	0.17	2.3	172
Ni-63	W	Self-Lum Light Src	19	1.20E+05	4	702	6,320	-	0.35	-
Po-210	E	Beta Gauges	98	3.85E+01	4	131	0.39	2.07E-09	0.065	2.07E-06
Pm-147	W	Self-Lum Light Src	1	4.42E+04	3	11,055	44,220	1.18E-04	5.5	0.12
Co-60	D	Gamma Gauges	41	4.13E+03	2	201	101	0.14	0.10	138
Sr-90	R	Gas Sources	135	2.44E+02	2	60	1.8	-	0.030	-
Fe-55	U	X-Ray Fluorescence	811	6.51E+04	2	9	80	-	0.0045	-
Cs-137	I	Calib Src A>30 mCi	17	6.49E+02	1.9	382	38	0.015	0.19	15
Cs-137	N	Ion Gen, Chromatog	11	3.13E+02	0.93	284	28	0.011	0.14	11
Sr-90	T	Other	182	8.63E+01	0.85	16	0.47	-	0.0079	-
Ti-44	N	Ion Gen, Chromatog	13	6.43E+02	0.64	165	49	0.0072	0.082	7.2
Cd-109	U	X-Ray Fluorescence	567	6.25E+02	0.62	4	1.1	2.03E-04	0.0018	0.20
Pm-147	D	Gamma Gauges	11	6.20E+03	0.46	141	564	1.51E-06	0.070	0.0015
Cd-109	E	Beta Gauges	85	4.80E+02	0.45	18	5.4	0.0010	0.0090	1.0
Cf-252	H	Gen Neut Src Apps	16	1.80E+00	0.24	50	0.10	4.18E-06	0.025	0.0042
Ra-226	S	Foil Sources	53	1.24E+00	0.18	12	0.023	3.34E-05	0.0058	0.033
Sc-46	N	Ion Gen, Chromatog	167	4.91E+02	0.16	3	2.9	0.0034	0.0016	3.4
Cs-137	Y	Calibrators	109	4.64E+01	0.14	4	0.43	1.62E-04	0.0021	0.16
H-3	S	Foil Sources	25	1.25E+03	0.12	17	50	-	0.0083	-
C-14	W	Self-Lum Light Src	267	3.67E+01	0.12	2	0.14	-	7.64E-04	-
Am-241	S	Foil Sources	2	2.60E-01	0.10	162	0.13	4.07E-05	0.081	0.041
Ba-133	T	Other	644	5.88E+02	0.087	0	0.91	4.16E-04	2.28E-04	0.42
Am-241	I	Calib Src A>30 mCi	9	2.17E-01	0.080	30	0.024	7.55E-06	0.015	0.0076
Ru-106	D	Gamma Gauges	2	4.00E+01	0.059	100	20	-	0.050	-
Cd-109	D	Gamma Gauges	44	5.27E+01	0.052	4	1.2	2.21E-04	0.0020	0.22
Cd-109	T	Other	10	4.50E+01	0.048	16	4.9	9.03E-04	0.0082	0.90
Am-241	N	Ion Gen, Chromatog	1	1.30E-01	0.048	162	0.13	4.07E-05	0.081	0.041

Table Task-3-1C continued. Sorted by "Fractions of Total Ing. ALIs."

			'87 thru '92		sum of		Average		Average	Dose	Dose from
			Total		Fractions of		Activity/		Source	from	Spending
			Number	Sum of	Total Ing.	Average Ing.	Device	Strength,	Ingestion	1000 hours	
			of	Activity	ALIs	ALIs/ Device	Device	Gamma*A	of Source	at 1 m from	
			Devices,	decayed to	decayed to		(mCi)	(rem/h	(rems)	Source	Source
Nuclide	Dev -	Device Definition	'87-'92	1992 (mCi)	1992 (ppm)			@1m)			(rems)
Code	ice										
Pm-147	T	Other	1	5.90E+02	0.044	147	590	1.58E-06	0.074	0.0016	
Fe-55	D	Gamma Gauges	57	1.10E+03	0.036	2	19	-	0.0011	-	
H-3	T	Other	20	2.93E+02	0.029	5	15	-	0.0024	-	
Ni-63	O	Ion Gen, Static Elim.	56	8.28E+02	0.027	2	15	-	8.21E-04	-	
Cd-109	N	Ion Gen, Chromatog	8	2.65E+01	0.026	11	3.3	6.11E-04	0.0055	0.61	
Fe-55	N	Ion Gen, Chromatog	18	7.67E+02	0.025	5	43	-	0.0024	-	
C-14	T	Other	33	7.25E+00	0.024	2	0.22	-	0.0012	-	
Co-60	E	Beta Gauges	3	3.52E+01	0.021	23	12	0.016	0.012	16	
Fe-55	T	Other	12	5.80E+02	0.018	5	47	-	0.0026	-	
Ra-226	T	Other	12	1.20E-01	0.018	5	0.010	1.42E-05	0.0025	0.014	
Fe-55	E	Beta Gauges	13	4.64E+02	0.015	4	36	-	0.0020	-	
Ni-63	H	Gen Neut Src Apps	22	3.33E+02	0.011	2	15	-	8.33E-04	-	
C-14	R	Gas Sources	42	2.10E+00	0.0069	1	0.050	-	2.78E-04	-	
Sr-90	I	Calib Src A>30 mCi	1	5.00E-01	0.0049	17	0.50	-	0.0083	-	
Ra-226	I	Calib Src A>30 mCi	1	2.00E-02	0.0030	10	0.020	2.85E-05	0.0050	0.029	
I-129	T	Other	24	2.96E+01	0.0029	0	1.2	1.55E-04	2.05E-04	0.16	
Ni-63	S	Foil Sources	6	6.99E+01	0.0023	1	12	-	6.47E-04	-	
C-14	E	Beta Gauges	6	6.00E-01	0.0020	1	0.10	-	5.55E-04	-	
Ni-63	U	X-Ray Fluorescence	8	3.91E+01	0.0013	1	4.9	-	2.72E-04	-	
Eu-152	T	Other	56	1.08E+00	0.00040	0	0.019	1.44E-05	1.21E-05	0.014	
C-14	Y	Calibrators	24	1.20E-01	0.00040	0	0.0050	-	2.78E-05	-	
Ba-133	I	Calib Src A>30 mCi	93	1.53E+00	0.00023	0	0.016	7.51E-06	4.12E-06	0.0075	
C-14	N	Ion Gen, Chromatog	10	1.00E-02	0.000033	0	0.0010	-	5.55E-06	-	
Am-241	Y	Calibrators	6	6.79E-05	0.000025	0	1.13E-05	3.55E-09	7.08E-06	3.55E-06	
Sr-90	Y	Calibrators	2	2.41E-03	0.000024	0	0.0012	-	2.01E-05	-	
Ba-133	N	Ion Gen, Chromatog	5	8.79E-02	0.000013	0	0.018	8.01E-06	4.39E-06	0.0080	
Ni-63	E	Beta Gauges	22	2.17E-01	7.15E-06	0	0.010	-	5.48E-07	-	
Ni-63	T	Other	12	1.19E-01	3.93E-06	0	0.010	-	5.52E-07	-	

Table Task-3-1C continued. Sorted by "Fractions of Total Ing. ALIs."

Nuclide	Dev - ice Code	Device Definition	Total Number of Devices, '87-'92	Sum of Activity decayed to 1992 (mCi)	'87 thru '92 sum of Fractions of Total Ing. ALIs decayed to		Average Ing. ALIs/ Device	Average Activity/ Device (mCi)	Average Source Strength, Gamma*A (rem/h @1m)	Dose from Ingestion of 10^-4 of Source (rems)	Dose from Spending 1000 hours at 1 m from Source (rems)
						1992 (ppm)					
Ru-106	E	Beta Gauges	1	9.65E-04	1.43E-06	0	0.0010	-	2.41E-06	-	
Bi-210	E	Beta Gauges	1	2.79E-89	1.04E-92	0	2.79E-89	-	1.75E-92	-	
Kr-85	D	Gamma Gauges	291	1.20E+05	0	0	413	0.00E+00	-	0.00E+00	
Kr-85	E	Beta Gauges	623	2.43E+06	0	0	3,905	0.0061	-	6.1	
Kr-85	N	Ion Gen, Chromatog	135	6.94E+04	0	0	514	0.05E-04	-	0.80	
Kr-85	O	Ion Gen, Static Elim.	128	3.44E+02	0	0	2.7	4.21E-06	-	0.0042	
Kr-85	R	Gas Sources	2	2.11E+03	0	0	1,054	0.0017	-	1.7	
Kr-85	T	Other	11	2.20E+03	0	0	200	3.13E-04	-	0.31	
Kr-85	U	X-Ray Fluorescence	13	5.44E+01	0	0	4.2	6.55E-06	-	0.0065	
			325681								

Table Task-3-2 shows the total number of sources in each Device Code category. Since many of these contain activities greater than 20 mCi, the numbers of sources used as a basis for the ORAU Report would be less than these numbers reported here.

Table Task-3-2. Summary of All Database (Baggett 1993).

Device Code ^a	Device Definition ^b	Number of Devices ^a
31.5D	Gamma Gauges	24,679
31.5E	Beta Gauges	18,336
31.5H	General Neutron Source Applications	110
31.5I	Calibration Sources, A > 30 mCi	317
31.5N	Ion Generators, Chromatography	9,474
31.5O	Ion Generators, Static Eliminators	37,084
31.5R	Gas Sources	11,146
31.5S	Foil Sources	827
31.5T	Other	5,277
31.5U	X-Ray Fluorescence	3,763
31.5W	Self-Luminous Light Source	310,667
31.5Y	Calibrators	577
31.7W	Self-Luminous Light Source	71,315
TOTAL		493,572

^aAttachment to letter dated Nov. 2, 1993, from S.L. Baggett to D.J. Strom, entitled "Summary of all Database."

^bU.S. Nuclear Regulatory Commission Regulatory Guide 10.10, Appendix C (NRC 1987).

Taylor (1939) estimated that there were "approximately 30,000 general licenses in non-Agreement States using about 400,000 devices, and about twice this many in Agreement States." Using Taylor's overall factor of 3 and the figure of 493,572 provided by NMSS, one is led to conclude that about $3 \times 493,572$, or roughly 1.5 million generally licensed sources exist in the USA.

Registration rates for the various Device Codes are shown in Table Task-3-3. Production figures are presented in the "Enclosure 1"¹ and ORAU report (see Attachment 2, Table 1, from the

¹The data identified as "Enclosure 1" data, entitled "Estimated number of generally licensed devices and materials," dated 4/21/87, have been ascribed by S.L. Baggett to "part of an older Commission paper or NRC staff report." The exact provenance of this data table is uncertain, except that it was ultimately supplied to PNL by NMSS for this review.

ORAU Report). Production appears to have remained relatively constant for the major categories of sources identified in the ORAU report. Thus, a constant annual production was assumed to estimate the total number of remaining sources (with the assumption of decay for short-lived sources), as shown in Table Task-3-4. There is remarkable agreement between the 1987-92 figures and the data reported in earlier work.

The data in Tables Task-3-3 and Task-3-4 do not include isotopes, dates, and activities, and thus, are of very limited use in risk analysis.

The 1987 production and accumulation of total quantities of generally licensed sources is given in the "Enclosure 1" table, as shown in Table Task-3-5. A correspondence between the "Enclosure 1" categories and the ORAU categories is given where possible.

Table Task-3-3. Annual device registration rates based on 6-year averages (1987-1992) from "General License Database System Report for Peer Review."

Code	Device Definition	per year	per 6 years
D	Gamma Gauges	1083	6498
E	Beta Gauges	1289	7731
H	General Neutron Source Applications	14	86
I	Calibration Sources, A > 30 mCi	20	121
N	Ion Generator, Chromatography	866	5197
O	Ion Generator, Static Eliminator	4820	28918
R	Gas Sources	1625	10950
S	Foil Sources	14	86
T	Other	347	2084
U	X-Ray Fluorescence	426	2553
W	Self-Luminous Light Source	40466	242797
W7	31.7 Self-Luminous Light Source	3087	18519
Y	Calibrators	24	141
	TOTALS	54280	325681

Table Task-3-4. Annual Production Rate and Numbers of Sources in Non-Agreement States from the "Enclosure 1" table dated 4/21/87 and from Table 4 of the 1987 ORAU Report by ORAU Report Category.

ORAU Category	"Encl. 1" 4/21/87 Annual Production Rate	"Encl. 1" 4/21/87 Total as of 1987	ORAU 1987 Table 4
A-1 Static Eliminators: Hand-Held/Portable/Small Brushes (10 CFR 31.3)	7000	20000	20000
A-2 Static Eliminators or Detectors: In Equipment or Process Line (Very High Toxicity Sources) (10 CFR 31.5)	80000	160000	170000
A-3 Static Eliminators or Detectors: In Equipment or Process Line (Low Toxicity Sources) (10 CFR 31.5)	120	9600	
B Gamma Gauges (10 CFR 31.5)	337	16000	4200
C-1 Beta Gauges: Backscatter Type (10 CFR 31.5)	800	7000	8000
C-2 Beta Gauges: Transmission Type (10 CFR 31.5)	-	-	
D Gas Chromatographs (10 CFR 31.5)			8000
E-1 X-Ray Fluorescence Analyzers: Very High Toxicity Sources (10 CFR 31.5)	90	720	720
E-2 X-Ray Fluorescence Analyzers: Moderate Toxicity Sources (10 CFR 31.5)	-	-	
F Sources for Checking Detector Operation or Calibration and Analytical Reference Sources (10 CFR 31.5)	-	-	-
G-1 Self-Luminous Devices (10 CFR 31.5)	20000	180000	180000
G-2 Self-Luminous Devices in Aircraft (10 CFR 31.7)	7600	90000	90000
H Analytical Instruments Containing Small Calibration or Reference Sources (10 CFR 31.5)	> 600 (Liq Scint)	7000 (Liq Scint)	7000
I Sources for Checking Detector Operation or Calibration and Analytical Reference Sources (10 CFR 31.8)	60	480	2000
J Small Quantities of Source Material (10 CFR 40.22)	70 kg	2000 kg	-
K Sources for Checking Detector Operation or Calibration and Analytical Reference Sources (10 CFR 70.19)	Pu-239	NA	-
TOTAL (except J,K)	-	506745	487920

Table Task-3-5. Estimated Number of Generally Licensed Devices and Materials ("Enclosure 1," 4/21/87). ORAU Categories are given where there is a clear correspondence between the "Enclosure 1" categories (i.e., Regulatory Guide 10.10; NRC 1987) and the ORAU categories.

CFR Sec.	Device Code	Device Type	Number of Devices Sold per Year	Total Number of Devices	ORAU Category (if applicable)
31.3		Static Eliminator	7000	20000	A-1
31.5		Aerosol Neutralizer	120	9600	A-3
31.5	E	Beta Backscatter Gauge	800	7000	C-1
31.5	N	Electron Capture Detector	900	8000	D
31.5		Electrostatic Voltmeter	890	3000	H
31.5		Fill Level Gauge	600	4200	B
31.5		Fuel Densitometer Emitter	200	945	B
31.5	D	Gauging Devices (Part I)	337	16000	B
31.5		In Flight Blade Inspection Systems	200	1000	B
31.5		Liquid Scintillation Spectrometers	600	7000	H
31.5	W	Self-Luminous Exit Signs	20000	180000	G-1
31.5		Static Eliminators/Meters	80000	160000	A-2
31.5		X-Ray Fluorescence Spectrometer	90	720	E-2
31.7		Self-Luminous Aircraft Signs	7600	80000	G-2
31.8		Calibration or Reference Sources	60	480	
40.22		Source Material (Depleted Uranium)	140	2000	J
TOTAL			119397	497945	

The data provided by NMSS (Baggett 1993) shows 71,315 devices in Device Code 31.7W, self-luminous devices in aircraft. This is in reasonably good agreement with the 10 CFR 31.7 entry in the "Enclosure 1" Table (80,000) and with the ORAU estimate for Category G-2 (90,000).

The "Enclosure 1" Table, produced in 1987, shows 497,945 devices, and the NMSS table, produced 6.5 years later in 1993, shows 493,572. This is a remarkable coincidence with the ORAU Table 4 Total of 487,920 devices. One could safely conclude that there are about a half a

million generally licensed devices in existence under NRC General License, with perhaps twice that many in Agreement States (Taylor 1989).

In general, it has not been possible to establish a correspondence among the ORAU categories, the categories in the "Enclosure 1" Table of 4/21/87, and the data provided by NMSS (Baggett 1993). This has been confirmed in the letter dated March 30, 1994, from S.L. Baggett to D.J. Strom. To apply the ORAU risk analysis methods, it will be necessary to resolve what numbers of devices of what designs, sources, and activities are in use. This will require significant additional investigation and direct access to the database.

4.0 TYPES, FREQUENCIES, AND RELATIVE SEVERITIES OF IMPROPER TRANSFER/DISPOSAL OCCURRENCES

NMSS inspection report summaries (Piner 1990, Wheaton 1993) and the operational experience reports and bulletins issued by the NRC's Office of Analysis and Evaluation of Operational Data (AEOD) provided information on the types, frequencies, and relative severities of improper transfer/disposal occurrences having actual or potential public dose implications. These data were accessed through written reports and the Nuclear Regulatory Event Report (NRER) database files listed in Attachment 1. These files covered event reports during the period 1980 through 1992.

The NRER database was searched for occurrences of anything leading to events concerning generally licensed materials. Such events were found with a code of "GEN" or "GL" in the LIC_NO field, and in a couple of other instances. Many of the entries, however, either did not involve generally-licensed devices or materials, or involved materials whose range of activities exceeded the activity limits in the ORAU Report. The latter groups were categorized as "L," and tabulated with the other ORAU Report Categories. Several hundred entries under "NL" (not licensed) were also reviewed, revealing no events involving generally-licensed devices.

Results of the survey are summarized in Table Task-3-6 (number of incidents) and Table Task-3-7 (number of sources or devices). Table Task-3-6 shows incident rates by rows labeled with ORAU Category. Column entries are numbers of incidents of each kind that the PNL reviewers found in the NRER database. Since a given incident may involve more than one source, Table Task-3-7 shows number of devices involved in the incidents listed in Table-3-6 by rows labeled with ORAU Category. Many event reports covered multiple sources, including various nuclides (^{60}Co , ^{210}Po and ^{210}Pb in one case, and ^{241}Am , ^{210}Po , and ^{85}Kr in another case). By far the most common occurrences were in Category A2, static elimination sources, with ^{210}Po being the most commonly-involved nuclide.

In the tables, an incident can be classified in more than one category of outcome, such as lost and recovered, or damaged and leaking.

In most cases, the NRER database gives no information concerning severity. Sometimes a description like "significant event" occurs. On only a rare occasion is a dose to a person mentioned. Occasionally a description like "completely destroyed in a fire resulting in contamination" occurs.

Table Task-3-6. Number of incidents involving generally-licensed devices. Data are from the NRER database files listed in the attachment, for the period 1980-1992. Category "L" involves sources whose activities exceed the upper limits on activity in the ORAU Report.

ORAU Cat.	Incidents*	Lost	Stolen	Land-fill	Scrap-yard	Damaged	Leaking	Recovered	Mis-use
A1	3	2				1			
A2	49	28	2	5	1	9	12	2	1
A3	1							1	
B	6	4	1		1	1		1	1
C1	1						1		
C2	1	1		1					
D	1			1					
E1	2	2							
E2	2	1				1			
F									
G1	11	1	5		2	2	2	1	
G2	1	1							
H	3	1			1		1	1	
I									
J	3	1					1		
K									
L	30	8	4	1	5	6	2	5	4
Total	114	50	12	8	10	20	19	11	6
Total w/o L	84	42	8	7	5	14	17	6	2

*Incidents may be less than row total because some reports are cataloged multiple times, e.g., "stolen and recovered."

Table Task-3-7. Number of devices involved in the incidents listed in Table Task-3-6. Data are from the NRER database files listed in the attachment, for the period 1980-1992 (there were no entries for 1980-82, so this is a 10-year period). Category "L" involves sources whose activities exceed the upper limits on activity in the ORAU Report.

ORAU Cat.	Devices*	Lost	Stolen	Land- fill	Scrap- yard	Dam- aged	Leaking	Recover- ed	Mis- use
A1	14	9				5			
A2	89	54	2	5	3	28	27	4	3
A3	1								
B	9	6	1		1	2		1	3
C1	1						1		
C2	1	1		1					
D	1			1					
E1	2	2							
E2	2	1				1			
F									
G1	57	1	43		2	9	9	1	
G2	1	1							
H	5	1			1		3	1	
I									
J	4	2					1		
K									
L	113	12	51	1	7	35	34	6	7
Total	300	90	97	8	14	80	75	13	13
Total w/o L	187	78	46	7	7	45	41	7	6

*Devices may be less than row total because some reports are cataloged multiple times, e.g., "stolen and recovered."

There were 41 incidents tabulated in the ORAU report (Table 4 in Stabin et al. 1987, p. 13). Our analysis, displayed in Tables Task-3-6 and Task-3-7, shows 114 incidents involving 300 devices over a 10-year period, for rates of 11.4 incidents per year and 30 device-incidents per year. The ORAU incidents were based on only a few years worth of data, and included data from agreement state reports. It is also unclear whether the ORAU Report included numbers of devices involved (see Table Task-3-7.) Thus, the two totals are not strictly comparable. If the "L" incidents listed in Table Task 3-7 are not counted, the rates become 8.4/year and 18.7/year.

Incident breakdown by nuclide and ORAU source category is shown in Table Task-3-8. It is difficult to justify the relevance of the "L" category for this study. Ignoring L incidents, many of

which may be improperly classified in the database, the bottom row gives the total number of incidents for each nuclide. The nuclides, ^{210}Po and ^{241}Am , dominate with 44 and 17 occurrences, respectively, out of a total of 84 incidents. Clearly, row totals show that incidents are dominated by static elimination sources (49 of 84 generally-licensed incidents), with self luminous devices following at 11.

Table Task-3-8. NRER Incident Breakdown by Nuclide and Source Category. Table entries are number of incidents per 10 years.

ORAU Cat.	^{241}Am	^{109}Cd	^{60}Co	^{137}Cs	DU ^{238}U	^3H	^{85}Kr	^{63}Ni	^{147}Pm	^{210}Po	^{90}Sr	Z	Total
A1										3			3
A2	4									40		5	49
A3							1						1
B	1			5									6
C1									1				1
C2											1		1
D								1					1
E1	2												2
E2		2											2
F													0
G1						5	5						11
G2									1				1
H				1				1				1	3
I													0
J					3								3
K													0
L	10		2	6		5			3	1		2	30
Total	17	2	2	12	3	10	7	2	5	44	1	9	114
Total w/o L	7	2	0	6	3	5	7	2	2	43	1	6	84

5.0 CONCLUSIONS AND NEED FOR FURTHER WORK

The NRER data base gives a good idea of how many incidents have been reported. It would be good to find a formal method of estimating the number of unreported incidents, and for this a survey of field regulatory personnel may be the best approach.

A better study of the actual numbers of sources in use and in storage would be desirable. A more formal survey of manufacturers would provide a good basis for risk estimates for short-lived nuclides, such as Po-210, which appears to dominate the numbers of sources.

The readily available information examined here should provide an adequate basis for an order-of-magnitude risk analysis, which would be better than many risk analyses in non-radiation fields.

6.0 REFERENCES

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TASK 3-ATTACHMENT 1: DOS FILES RECEIVED AND REVIEWED BY PNL

The following list contains the most recent versions of files reviewed by PNL staff for this task.

ADDRESS	DBF	176066	06-03-93	3:33p	D:\NMSS\SSD
B-BITXT	CSV	413152	04-07-93	4:56p	D:\NMSS\INCIDENT
BYCODE	NTX	14336	06-03-93	2:41p	D:\NMSS\SSD
CATALOG	CAT	439	03-17-92	10:17a	D:\NMSS\NRER
COMPANY	NTX	67584	06-03-93	2:41p	D:\NMSS\SSD
CUSTOM	DBF	25149	06-03-93	12:17p	D:\NMSS\SSD
CUSTOM1	NTX	5120	06-15-93	4:49p	D:\NMSS\SSD
CUSTOM2	NTX	10240	06-15-93	4:49p	D:\NMSS\SSD
EXP	DBF	257458	05-03-93	4:29p	D:\NMSS\NRER
EXP	XLS	158633	08-25-93	10:23a	D:\NMSS\NRER
LIC	NDX	49664	06-24-92	4:48p	D:\NMSS\NRER
NRER	ZIP	926512	01-14-93	11:08a	D:\NMSS\NRER
NRER0	DBF	186138	05-14-91	10:40a	D:\NMSS\NRER
NRER1	DBF	250880	07-19-92	8:47p	D:\NMSS\NRER
NRER2	DB	587469	04-20-93	9:34a	D:\NMSS\NRER
NRER2	DBF	306150	01-11-93	2:05p	D:\NMSS\NRER
NRER2	DBT	195584	01-11-93	2:05p	D:\NMSS\NRER
NRER8	DBF	119808	05-23-90	2:08a	D:\NMSS\NRER
NRER80-4	DBF	939378	08-20-92	6:07p	D:\NMSS\NRER
NRER85-7	DBF	769898	05-03-93	4:25p	D:\NMSS\NRER
NRER9	DBF	178818	01-04-80	1:48a	D:\NMSS\NRER
PRINUSE	DBF	1398	12-21-91	9:57a	D:\NMSS\SSD
REGNUM	NTX	73728	06-03-93	2:41p	D:\NMSS\SSD
REGNUM1	NTX	55296	06-03-93	2:41p	D:\NMSS\SSD
REGNUM2	NTX	47104	06-03-93	2:41p	D:\NMSS\SSD
SSD	DBF	2706829	06-03-93	3:33p	D:\NMSS\SSD
SSDS	EXE	324608	06-15-93	4:59p	D:\NMSS\SSD
TEMP	DBF	1155	04-26-93	7:18a	D:\NMSS\SSD
TEMP	NTX	152576	08-20-93	1:16p	D:\NMSS\SSD
UNTITLED	CAT	439	03-17-92	10:17a	D:\NMSS\NRER
YR86	DBF	38778	03-17-92	10:40a	D:\NMSS\NRER

TASK 3 ATTACHMENT 2: Table 1 from the ORAU Report (Stabin et al. 1987).

TABLE 1 CLASSES OF DEVICES FOR SCENARIO DEVELOPMENT*

APPLICABLE REGULATORY SECTION**	CLASS	DEVICE	RADIONUCLIDES AND MAXIMUM ACTIVITIES
31.3	A-1	Static Eliminators: Hand-Held/Portable/ Small Brushes	Po-210 - 0.50 mCi (18.5 MBq)
31.5	A-2	Static Eliminators or Detectors: In Equipment or Process Line (Very High Toxicity)	Po-210 - 100 mCi (3700 MBq) Am-241 - 0.0005 mCi (0.0185 MBq) Ra-226 - 0.0005 mCi (0.0185 MBq)
31.5	A-3	Static Eliminators or Detectors: In Equipment or Process Line (Low Toxicity)	H-3 - 250 mCi (9250 MBq) Kr-85 - 2 mCi (74 MBq)
31.5	B	Gamma Gauges	Co-60 - 10 mCi (370 MBq) Cs-137 - 20 mCi (740 MBq) Am-241 - 20 mCi (740 MBq) Ra-226 - 10 mCi (370 MBq)
31.5	C-1	Beta Gauges: Backscatter Type	Sr-90 - 0.025 mCi (0.925 MBq) Tl-204 - 0.10 mCi (3.7 MBq) Ru-106 - 0.025 mCi (0.925 MBq) Pm-147 - 0.050 mCi (1.85 MBq) C-14 - 0.050 mCi (1.85 MBq) Pb-210 - 0.010 mCi (0.37 MBq)
31.5	C-2	Beta Gauges: Transmission Type	Sr-90 - 20 mCi (740 MBq)
31.5	D	Gas Chromatographs	Ni-63 - 20 mCi (740 MBq) H-3 - 1000 mCi (37 GBq)
31.5	E-1	X-Ray Fluorescence Analyzers (Very High Toxicity)	Am-241 - 30 mCi (1100 MBq) Cm-244 - 100 mCi (3700 MBq)

TASK 3 ATTACHMENT 2 (continued)

TABLE 1 CLASSES OF DEVICES FOR SCENARIO DEVELOPMENT* - CONTINUED

APPLICABLE REGULATORY SECTION	CLASS	DEVICE	RADIONUCLIDES AND MAXIMUM ACTIVITIES
31.5	E-2	X-Ray Fluorescence Analyzers (Moderate Toxicity)	Cd-109 - 20 mCi (740 MBq) Fe-55 - 100 mCi (3700 MBq)
31.5	F	Calibration or Reference Sources	Cs-137 - 0.10 mCi (3.7 MBq) Co-60 - 0.01 mCi (0.37 MBq) Ra-226 - 0.004 mCi (0.15 MBq) Sr-90 - 0.001 mCi (0.037 MBq)
31.5	G-1	Self-Luminous Devices	H-3 - 5000 mCi (185 GBq) Kr-85 - 1700 mCi (62.9 GBq) C-14 - 0.10 mCi (3.7 MBq)
31.7	G-2	Self-Luminous Devices in Aircraft	H-3 - 5000 mCi (185 GBq) Pm-147 - 300 mCi (11 GBq)
31.8	H	Analytical Instruments Containing Small Calibration or Reference Sources	Cs-137 - 0.040 mCi (1.5 MBq) Ni-63 - 15 mCi (555 MBq)
31.8	I	Calibration or Reference Sources	Am-241 - 0.005 mCi (0.185 MBq)
40.22	J	Small Quantities of Source Material	U-238 and Th-232 - 15 pounds at any one time, no more than 150 pounds per calendar year
70.19	K	Calibration or Reference Sources	Pu-239 - 0.005 mCi (0.185 MBq)

* See Appendix for device descriptions
 ** Code of Federal Regulations, Title 10

APPENDIX F TO
FINAL REVIEW OF THE 1987 REPORT BY
OAK RIDGE ASSOCIATED UNIVERSITIES,
"IMPROPER TRANSFER/DISPOSAL SCENARIOS FOR
GENERALLY LICENSED DEVICES:"

TECHNICAL LETTER REPORT:
TASK 6, DEVELOPMENT OF ADDITIONAL PROBABILITY AND RISK
INFORMATION

NRC JOB CODE L2536
PNL No. 20278

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1.0 EXECUTIVE SUMMARY

Task 6 of the project "Review of Improper Transfer/Disposal Scenarios for Generally Licensed Devices Study" is entitled "Development of Additional Probability and Risk Information." For this task, ... "the individual reviewers will develop additional probability and risk information in their area of expertise to assist the NRC staff's decision making regarding the need for regulatory action. This information may include refined estimates of probabilities associated with selected disposal scenarios, assessments of the sensitivity of consequence and risk calculations to different assumptions and inputs, and quantitative estimates of individual and population risk resulting from selected improper disposal activities."

Two areas of additional risk information have been developed by the PNL reviewers. The first area of additional risk information that was identified in PNL's preliminary review of the 1987 ORAU Report is the need for a mathematical framework or "formula" for the risk of radiological accidents. This framework should address two items of methodology missing from the ORAU Report; the probabilities of initiation of accident sequences, and the use of historically-derived probability distributions of accident consequences (which include worst cases as their extremes).

The second area of additional risk information is a quantitative characterization of relevant historical accidents with sources, whether generally licensed or not. The quantitative characterization results in three numerical factors for each person involved in each accident for whom intake, whole body dose, and local dose values are published. The numerical values are "fraction taken in," "whole body time-and-proximity factor," and "local dose time-and-proximity factor." These factors depend only on human behavior and accident circumstances, not on the amount, kind, and quantity of radioactive material involved in the accident. The factors can be used to predict more realistically the radiological consequences of future accidents than the use of "worst case" factors. Furthermore, over 10 accidents involved exposures to two or more people, resulting in *distributions*, rather than point estimates, of values. Such distributions can be used as inputs to modern probabilistic risk calculations.

2.0 PROBABILISTIC MATHEMATICAL FRAMEWORK

The first area of additional risk information that was identified in PNL's preliminary review of the 1987 ORAU Report is the need for a mathematical framework or "formula" for the risk of radiological accidents. This framework uses individual and collective radiation dose as surrogates for risk, and considers both the magnitude and probability of occurrence of various doses. The PNL reviewers have identified two probability considerations as missing from the ORAU Report. The probabilities of initiation of accident sequences, and the likely (rather than worst case) consequences are included. Using historically-derived probability distributions of accident consequences (which include worst cases as their extremes) enhances the realism of risk estimates calculated from postulated accidents.

2.1 ASSESSMENT OF THREE KINDS OF DOSES FOR USE AS SURROGATES FOR HUMAN HEALTH RISKS

Risk is conventionally defined as

$$\text{Risk} = \text{Probability} \times \text{Severity} . \quad (1)$$

There may be several separate components to the probability term: probability of an accident happening, probability of a given dose resulting when the accident happens, and probability of that dose resulting in a stochastic health effect, for example. For human health risks due to radiation exposure, various dose quantities multiplied by suitable health risk coefficients may be used as surrogates *severity* (ICRP 1991). At low doses, *severity* may connote the *likelihood* of a severe effect such as cancer occurring in an individual. At higher doses exceeding thresholds for deterministic effects, "severity" has a more conventional meaning for an individual, such as how serious a burn is.

Severity in Equation 1, for incidents involving sealed sources, can be defined both as individual doses and collective doses, that is, the sum of all doses accruing to all individuals in a given incident.

An individual tissue or organ dose equivalent, if below 50 rems, carries no risk of deterministic (formerly "non-stochastic") health effects (such as radiation burns, developmental abnormalities, etc.), but represents some degree of risk for stochastic effects (i.e., cancer and heritable ill-health). Such individual tissue or organ dose equivalents from internal and external exposure can be combined to form a total effective dose equivalent (TEDE), which is a modern surrogate for stochastic risk in individuals. Current risk estimates are on the order of 4×10^{-2} per sievert (4×10^{-4} per rem) for adverse stochastic health outcomes in workers, and perhaps 5×10^{-2} per sievert (5×10^{-4} per rem) in the general public (ICRP 1991).

Under the linear, non-threshold dose response hypothesis for stochastic effects used for radiation protection purposes, individual TEDE values can be summed to make collective total effective dose equivalent ("collective dose"). Collective dose is a surrogate for collective risk of adverse stochastic health outcomes in populations.

For improper transfer and disposal scenarios for generally licensed devices, it is also necessary to consider the possibility of individual tissue or organ doses that exceed thresholds for deterministic effects. Such doses result in certain injury, whether sub-clinical, mild, severe, or fatal, to the individual receiving the dose. Doses above a few tenths of a sievert (a few tens of rems) should be expressed in absorbed dose units, i.e., grays (or rads), specifying the radiation type, since the relative biological effectiveness of high linear energy transfer (LET) radiation (e.g., neutrons and α particles) at such dose levels is significantly less than the quality factor used for limitation of stochastic effects (ICRP 1991). Furthermore, no clinical effects may occur whatsoever from protracted irradiation

significantly exceeding the traditional deterministic threshold of 0.5 Gy, since significant repair can occur between damaging events on a microscopic scale. For this reason, *committed* doses of long-lived, tenaciously-retained radionuclides will be poor predictors of deterministic effects.

Thus there are three dose endpoints that should be considered in a risk analysis: Distributions of individual TEDEs, the collective TEDE, and distributions of individual tissue or organ dose above thresholds, such as 0.5 Gy for acute irradiation and perhaps 1 Gy or more for protracted irradiation.

$$\text{Individual Stochastic Risk} = \text{Probability} \times \text{TEDE}$$

$$\text{Collective Stochastic Risk} = \text{Probability} \times \text{Collective TEDE} \quad (2)$$

$$\text{Individual Deterministic Risk} \propto \text{Probability of Dose Above a Threshold} \times \text{Dose Effect Function}$$

The probabilities of various doses being received from a given improper transfer/disposal scenario are related both to the probability of the scenario occurring and the probability distribution of doses resulting from the scenario. Finally, risks are summed over all scenarios.

Ecological risk is the risk to ecosystems, habitats, and potential loss of access and usability of land and environmental resources. Although improper transfer and disposal scenarios for generally licensed sources may result in ecological risks, they are not considered here.

2.2 PROBABILISTIC RISK METHODOLOGY

Since 1987, many changes have occurred in probabilistic risk methodologies. Recent summaries of these techniques are provided by IAEA (1989), Finkel (1990), and Morgan and Henrion (1990). In addition, the advent of user-friendly Monte Carlo simulation software for probabilistic health risk analysis, such as Crystal Ball (™Decisioneering, Inc., Denver, CO), makes it feasible to perform probabilistic risk assessments for this kind of work.

2.1.1 Risk Networks

The draft ORAU report contains, for each class of Generally Licensed Sources, a "risk network" connecting the Initial Events (listed in Table 2 of the report) with the Final Status of Device conditions (listed in Table 3). Figure 1 is an example risk network from the draft report. Each of these networks starts with the assumption that a device of that class has been improperly transferred/disposed. Along the left side of the network is a collection of Initial Events, or states that the device can be found in after improper transfer/disposal. Along the right side of the network are the Final Status of Device conditions. In between is a collection of transition conditions and a collection of paths leading (from left to right) from the Initial Events, perhaps through one or more of the transition conditions, ending in one of

CLASS A-2 EQUIPMENT STATIC ELIMINATORS

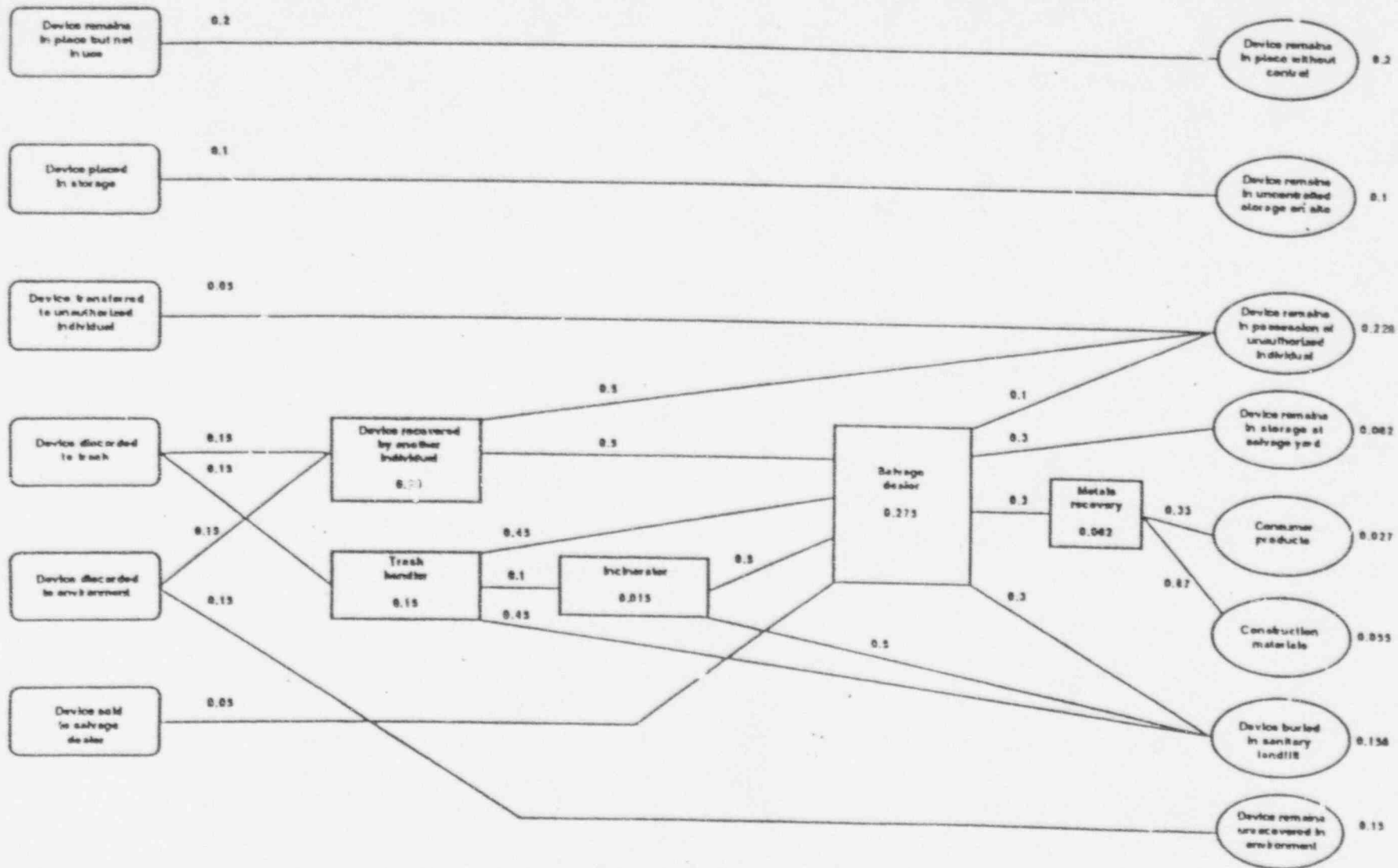


Figure 2.2.1. An example of a risk network from the ORAU Report (Figure 3).

the Final Status of Device conditions. Associated with each of the path segments between two of the condition boxes is the conditional probability of the transition from the left box (i.e., the left end of the path segment) to the right box. Finally, along the right side of the network, associated with each of the Final Status of Device conditions is the conditional probability of ending up in that state, obtained by adding up the probabilities associated with each of the distinct paths through the risk network that terminate in that state.

2.2.1 Adequacy of ORAU Report Risk Networks

We feel that the conditional probabilities assigned (using engineering judgment) to the path segments by the authors of the ORAU draft report are generally reasonable. There is, however, a structural aspect of these risk networks with which we take issue: all of the networks have a pathway leading from "trash handler" to "incinerator" and thence on to either "salvage dealer" (with conditional probability 0.5) or to the Final Status of Device condition "Device buried in sanitary landfill" (with conditional probability 0.5). This seems to presuppose that the trash handler sifts the incinerator ashes for metal/ceramic slag which is sent on to the salvage dealer, and that half the time (i.e., probability 0.5) the remains of the source are incorporated in that slag. The probability 0.5 of "source-in-slag" seems high. In addition, for those cases in which the bulk of the source is volatilized in the incinerator and released to the atmosphere, the box "incinerator" has effectively become a Final Status of Device condition for which health consequences should be assessed. Several potentially significant scenarios, such as an intact source out of a shield, and potentially significant consequences, such as doses to workers (rather than the public), have been omitted. Finally, one could argue that the incinerator box ought to be "downstream" of the salvage dealer, with incineration as one of the salvage dealer's options for dealing with items that incorporate both salvageable (i.e., metals) and non-salvageable materials.

What is missing from this picture, if we desire to estimate the risks associated with the improper transfer/disposal of various types of Generally Licensed Devices? Missing is the probability of entering the risk network in the first place (i.e., the probability that a device of that type will be improperly transferred/disposed) and the consequences associated with each of the Final Status of Device conditions.

2.2.2 Censoring of Scenario Probabilities by Under-Reporting

Table 4 of the draft ORAU report contains a computation of the fraction of devices of various types that have been improperly transferred/disposed, based on very limited data. These values cannot be directly used as improper transfer/disposal probabilities because of an obvious censoring problem. Improperly transferred/disposed devices show up in Table 4 only if the improper transfer/disposal was detected in some fashion. This detection could occur if an appropriately labelled device was found somewhere it didn't belong, if an inspection of records and device inventory discovers that a device is missing, or if someone's medical symptoms can be tied directly to the improper transfer/disposal of a specific device. If a device was buried in a landfill or incinerated improperly, that improper transfer/disposal

is unlikely to be detected, since inspections and audits of holders of General Licenses seem to sample only a small fraction of the total population of General Licensees.

2.2.3 Correcting Probabilities for Under-Reporting of Incidents

One way of dealing with this censoring would be to obtain an estimate for the probability $P(d)$ of the event d , where d denotes the detection, in some fashion, of the improper transfer/disposal of a particular Generally Licensed Device.

We can then take the fraction of devices of that class known to be improperly transferred/disposed, F (from Table 4), and increase it by the factor $1/P(d)$ to obtain the total fraction of devices of that type improperly transferred/disposed. That is, we assume that F is telling us only about that subset of the set of improperly transferred/disposed devices for which the improper transfer/disposal is detected in some fashion; that subset is only the $P(d)$ -part of the whole set of improperly transferred/disposed devices.

In other cases (see, for instance, the discussion of industrial process line static eliminators in Section 3.2 of the ORAU report), there is enough information to directly estimate the fraction of devices that are improperly transferred/disposed, without reference to Table 4. In this case, all of the devices are out on lease and the distributor simply offers to charge for an additional year's lease if the device isn't returned for legally authorized disposal.

Suppose we are going to estimate $P(d)$. We might consider d to be the union or "sum" of two events:

- a = the event that the improper transfer/disposal of the source is detected by observation of the label, by radiation detection, by inspection of records and inventory, etc.; and
- b = the event that radiation sickness or injury is recognized and tied back to the improper transfer/disposal of the source.

We can then calculate

$$P(d) = P(a) + P(b) - P(ab), \quad (3)$$

where ab is the set intersection of the events a and b . If $P(a)$ and/or $P(b)$ are relatively small, say 0.1 or less, then the term $P(ab)$ will be second order and can be ignored.

The probability $P(a)$ will depend on a variety of factors, including the durability and intrusiveness of the labelling, the likelihood of "fortuitous" radiation detection, the pervasiveness of inspections and audits, the unwieldiness of the individual sources, etc. By fortuitous radiation detection, we mean something similar to discovery of high indoor radon levels in the Reading Prong because a Pennsylvania nuclear power plant worker set off one of the portal detectors as he was arriving for work or the outside world's first knowledge of the

Chernobyl accident when area detectors outside a Swedish nuclear plant started alarming. We could also reasonably expect $P(a)$ to vary by type of source.

Reviewing the history of improper transfer/disposal of sealed sources, even "strong", specifically licensed sealed sources used in radiography or radiotherapy, we see dozens of incidents in which a sealed source improper transfer/disposal is first detected by recognition of radiation sickness or injury, followed by an investigation to determine the cause of that sickness. These detections are noted by the entry "Med" for "medical" detection of the incident with the subsequent re-establishment of control over the source.

We know of no such incidents involving Generally Licensed sources or equivalent amounts of radioactivity. Because of this, $P(b)$ is likely to be significantly smaller than $P(a)$ for most generally licensed sources. It would be reasonable to model $P(b)$ as directly proportional to the worst-case and/or the average human health consequences of improper transfer/disposal of that type of source.

We propose a model for $P(b)$ of the following type:

$$P(b) = k_a \cdot f(H_{T,\max}) + (1 - k_a) \cdot g(H_{T,\text{avg}}), \quad (4)$$

where $H_{T,\max}$ = dose to the critical organ for the maximally exposed individual;
 $H_{T,\text{avg}}$ = dose to the critical organ for the average individual;

$$f(H_{T,\max}) = \begin{cases} 0 & \text{if } H_{T,\max} \leq H_{th}; \\ \frac{H_{T,\max} - H_{th}}{H_{cr} - H_{th}} & \text{if } H_{th} < H_{T,\max} < H_{cr}; \\ 1 & \text{if } H_{T,\max} \geq H_{cr}, \end{cases} \quad (5)$$

H_{th} is the average individual threshold dose for developing clinical symptoms;

H_{cr} is the critical threshold dose for certain diagnosis of clinical symptoms;

$g(H_T)$ is defined similarly to f , using values $H_{th,pop}$ appropriate for failure to detect and $H_{cr,pop}$ for sure detection of the incident in a whole population exposed to a given dose H_T ; and

k_a = an appropriately defined weighting factor for balancing between detection based on a symptomatic maximally-exposed individual and detection based on symptoms in a population; $0 \leq k_a \leq 1$.

2.3 RECOMMENDATIONS FOR IMPLEMENTING THE PROBABILISTIC RISK METHODOLOGY

To carry out the program defined in the preceding paragraphs, we need to do the following:

- 1) For each Generally Licensed sealed source type x , reassess the improper transfer/disposal fractions, F_x , defined in Table 4 of the ORAU draft report, on the basis of additional data, as available. The annual *rates* of incidents of improper transfer or disposal as a function of source category, including probabilities of incidents not being reported were not adequately assessed in the ORAU Report. Improved data will be available from the revised Task 3 report.
- 2) For each Generally Licensed sealed source type x (i.e., A-1, A-2, A-3, B, etc.) and for each radionuclide that can be utilized for devices of that type, produce estimates of either $P(Mx)$ directly or of $P(ax)$ and $P(bx)$. If $P(ax)$ and $P(bx)$ are estimated, then $P(dx) = P(ax) + P(bx)$ and $P(Mx) = F_x/P(dx)$. If Mx is the event that a device of Type x is improperly transferred/disposed, we can use the probability $P(Mx)$ to "enter" the corresponding risk network (whether that probability is estimated directly or obtained by estimating the probability $P(d)$ and factoring up the fraction, F , from Table 4). The probability $P(Mx)$ will propagate through the network and each of the Final Status probabilities (from the ORAU draft) will be multiplied by $P(Mx)$ to give the probability that a particular device of that class will end up in that Final Status of Device. We can then multiply that probability by the associated health consequences, $C_{x,fs}$, for device type x and that Final Status fs , to obtain the risk with a single device of that type ending up in that particular Final Status. Those risks can be summed for all of the Final Statuses to obtain the risk associated with a single device of that type, which can then be multiplied by the number of such devices extant to obtain the total risk to the public associated with that type of device.
- 3) For each Final Status of Device end state, produce estimates of the human health consequences resulting from a device ending up in that state, either using information in the draft ORAU report or additional information, as necessary.
- 4) Define the risk associated with a particular type x of device in a particular end-state as:

$$R_{x,fs} = P(M_x) \cdot p_{x,fs} \cdot (\text{consequences associated with the device } x \text{ final state } fs), \quad (6)$$

where $p_{x,fs}$ is the end state probability from the corresponding ORAU draft report risk network.

- 5) Define the total risk, R_x , associated with a single device of type x as the sum of all the $R_{x,fs}$ over the set of final states fs . Finally, the total risk associated with devices of type x is the product of the number of devices of type x and $R_{x,fs}$.
- 6) We can do a simple uncertainty analysis by replacing all of the estimated probabilities and the probabilities given in the ORAU draft report by probability distributions and using random variable arithmetic to propagate those distributions through the calculations described above. The quick and dirty part would use just the mean and

variance of the distribution (or the mean and variance of the log-transformation of the distribution) and the Central Limit theorem to replace sums of random variables by the normal distribution with the appropriate mean and variance and replace products of random variables by the lognormal distribution with the appropriate mean and variance. These approximations give good results for random variables with a central tendency and arithmetic calculations consisting of several sums and products (roughly speaking the more sums and products, the better the approximation).

3.0 QUANTITATIVE ASSESSMENTS OF HISTORICAL IMPROPER TRANSFER/DISPOSAL INCIDENTS AND ACCIDENTS FOR INPUT TO PROBABILISTIC RISK METHODOLOGY

Many new risk assessment tools have been developed since 1987 when the ORAU Report was finalized. Furthermore, the PNL reviewers believe that the probabilistic nature of risk assessment results rather than worst case scenarios need to be emphasized. We have analyzed historical accidents involving sources to yield quantitative characterizations of human behavior in accident situations as input to probabilistic risk assessments. Nuclear weapons, nuclear fuel cycle, and criticality accidents and accidents involving accelerators or x-ray machines have not been analyzed, since it is difficult to determine the relevance of these to improper transfer/disposal scenarios for generally licensed devices.

3.1 DISTRIBUTIONS OF RESULTS RATHER THAN POINT ESTIMATES

We have performed a quantitative characterization of relevant historical accidents with sources, whether generally licensed or not, for those accidents for which necessary data are available in the literature. The quantitative characterization results in three numerical factors for each person involved in each accident for whom intake, whole body dose, and local dose values are published. The numerical values are "fraction taken in," "whole body time-and-proximity factor," and "local dose time-and-proximity factor." For the most part, these factors depend only on human behavior and accident circumstances, not on the amount, kind, and quantity of radioactive material involved in the accident. The factors can be used to more realistically predict the radiological consequences of future accidents than the use of "worst case" factors. Furthermore, over 10 accidents involved exposures to two or more people, resulting in *distributions*, rather than point estimates, of values. Such distributions can be used as inputs to modern probabilistic risk calculations.

3.2 TIME-AND-PROXIMITY FACTORS FOR WHOLE-BODY AND LOCAL EXTERNAL IRRADIATION

There is a need for risk analysis for accidents involving single radionuclide radioactive sources (as opposed to nuclear reactor or nuclear weapons accidents). Such risk analysis requires knowledge of the probabilities and severities of such accidents. Historically, accidents have involved anywhere from one to several thousand people.

Information for risk analysis of small, generally-licensed sources can be derived from accidents, usually involving large sources, that have already happened.

The objectives of this study were to

- Develop generalized, quantitative descriptions of human behavior and interactions with radiation sources from study of historical accidents;
- Evaluate applicability, advantages, disadvantages, and limitations of the approach; and
- Identify additional information needed to apply these risk estimates to quantitative risk assessments for informed regulatory decision making.

Historical records of accidental human interactions with radiation sources are used to develop distributions of external radiation exposure factors. These factors retain information about human behavior from the accidents, but are independent of source strength or the radionuclide(s) involved. For external exposures, we define a "time-and-proximity" exposure factor, F_p , with and without provision for breach of source shielding, for each person involved in an accident. The distributions of exposure factors can be used to predict ranges of possible radiation doses to individuals from a variety of accident scenarios involving different radionuclides, activities, and device designs. Use and limitations of the exposure factors and their distributions are discussed below.

One limitation is the possibility that the exposure factor distributions based on accident information are altered when the accidents are discovered. For example, deaths or symptoms of acute radiation syndrome may lead to an investigation, discovery of an accident, and termination of exposure. Distributions of factors from such an accident may be of limited applicability to accidents that go undiscovered.

3.2.1 Mathematical Description of Time and Proximity Factors

For an unshielded point source, dose equivalent H depends on exposure time t (hours), distance r (meters), source strength A (activity in Ci) and isotope (through Γ in rem/hr m²/Ci or Sv/h m²/Bq):

$$\begin{aligned}
 H &= \int \dot{H} dt \\
 &= \int \frac{\Gamma A dt}{r(t)^2} \\
 &= \Gamma A \frac{t}{r^2}.
 \end{aligned} \tag{7}$$

For each individual i , exposed in an incident for whom the dose equivalent H is known, the

source isotope and activity are known, one can calculate a time-and-proximity factor for the incident:

$$F_{p,i} = \frac{H_i}{\Gamma A} = \frac{t}{r^2} \quad (8)$$

This factor is independent of both source strength and radionuclide involved.

Distributions of time-and-proximity factors can then be applied to similar accidents to determine external exposures, even those involving a radioactive source of different isotope and activity. One simple definition of F_p is the number of hours one would have to spend at one meter from the source to receive a dose equivalent of H .

If the source remained partially or wholly shielded, then an additional factor should be introduced:

$$F_{p,i} = \frac{H_i}{F_s \Gamma A} = \frac{t}{F_s r^2} \quad (9)$$

where F_s is the fraction transmitted through a shield, a number less than 1. F_s can be taken as the dose rate at 1 m from the source (in its shield) to the unshielded dose rate at 1 m.

When sources are sub-divided, the full activity is still used in the calculations because the human interaction is what we want to characterize, not the immediate source. So doses from the 1984 Mexican accident, for example, are attributable to the entire source.

3.2.2 Theoretical Limits Are Not Useful

The theoretical upper limit on F_p is

$$\frac{1 \text{ lifetime}}{(\text{very close})^2} \quad (10)$$

where (very close) represents a small distance from the source, e.g., 0.01 m². For a weak source, this represents about 10⁹ to 10¹⁰ hours at one meter, a quantity so large as to be useless. However, in tens of historical accident cases, a person (and in two of the worst accidents, in Mexico in 1962 [5 fatalities] and Morocco in 1984 [8 fatalities], a child) has found an industrial radiography source and put it in a "hip pocket." In many cases, the "very close" is 1 cm or so, and exposure times have been up to several months. When large sources are involved, such as industrial radiography sources, these cases result in local radiation burns. The ratio of the average bone marrow dose to the dose at the site of the radiation burn is dependent on "how far the bone marrow is away from the hip pocket." In many cases, the bone marrow to burn site dose ratio is over 100, sometimes over 1000. In other words, one needs to look at truly potential exposures which may result in a high dose,

but are realistic.

3.2.3 Historically-Derived Values of Time-and-Proximity Factors

Table 3.2.1 shows an analysis of 42 incidents involving external exposure. Accidents are characterized by year of occurrence, by nuclide, source type and activity. The number of people involved (broken down as public, workers, and cleanup workers) is given. Each accident is characterized by whether it involved brief or protracted whole-body irradiation, brief or protracted localized irradiation, whether there were intakes, whether the source was removed from the shield, and whether the source was damaged. References to the literature are given. Specific dose equivalent rate constants are tabulated, along with dose rates at 1 meter from the sources. The table contains average and geometric mean values of whole-body time-and-proximity factors, as well as minimum, maximum, and standard deviation and geometric standard deviation values, where appropriate. Also tabulated are maximum values of skin dose or local irradiation time-and-proximity factors.

These 42 incidents were chosen because of the availability of data on source identity, source activity, whole body and/or local doses to individuals, incident descriptions. The NRC's Office of Analysis and Evaluation of Operational Data (AEOD) incident database does not, in general, contain the information needed for this kind of analysis. Nuclides include ^{241}Am (1 accident), ^{60}Co (20), ^{137}Cs (4; note that Goiânia is listed 4 times for various analyses), ^{131}I (1), and ^{192}Ir (16). Accidents included 21 industrial radiography sources, 8 sterilization facilities, 5 teletherapy sources, 3 experimental sources, 2 brachytherapy sources, 1 defense incident, and 1 "radiation station." Members of the public were involved in 14 incidents, in numbers ranging from 1 person to 4000 persons, totalling roughly 7000. Four accidents (Mexico 1983 [4000], Morocco 1984 [28], Brazil 1987 [2800], and Pennsylvania 1992 [94]) account for virtually all of this total 7000 people. There were 29 incidents involving exposure at work, and 6 incidents involved exposure to cleanup or recovery workers.

The weighted average of the whole body is 1.28 hours at a meter. This is far below the 3360 hours exposure time assumed as a worst case in the ORAU Report (i.e., 20 weeks \times 7 days per week \times 24 hours per day = 3360 [page i, Stabin et al. 1987]). Ignoring the Texas child-abuse case of 1972, whole-body time-and-proximity factors ranged from essentially zero to 686 hours at a meter, the latter deriving from the 1962 Mexican incident (which formed the basis for the ORAU Report value of 3360). For only accidents exposing the public, time-and-proximity factors averaged 117 hours at a meter (averaging over accidents) and 1.37 hours at a meter (using a weighted average over all accident victims, for which accidents with many victims dominate the average).

The 16 available maximum local time-and-proximity factors formed a highly skewed distribution ranging from 0.05 to 24000 hours at a meter, with an average of 3100, a standard deviation of 6900, and a geometric standard mean of 91. The average of 3100 is comparable to the 1962 Mexico ^{60}Co accident whole body factor.

Four accidents in particular are especially interesting, due to their nature and the potential for generalization to improper transfer and disposal scenarios for generally licensed devices. These are the November 1992 Indiana, Pennsylvania accident; the 1983-84 Mexican ^{60}Co accident, and the 1987 Goiânia, Brazil accident, and the 1990 Korean source shipment accident.

The November 1992 Indiana, Pennsylvania case of a medical misadministration is particularly instructive for improper transfer and disposal scenarios for generally licensed devices (NRC 1993a, NRC 1993b). Doses are listed in the Appendix. This was a case of complete loss of control of the source, and the persons exposed did not know about it until the source was essentially "out of harm's way." The ^{192}Ir source activity was 3.7 Ci ($1.37\text{E}11$ Bq). The specific dose equivalent rate constant, Γ , is $0.48 \text{ rem/hr m}^2/\text{Ci}$. The central (i.e., halfway between the minimum and maximum doses) estimates of doses ranged from 0.0006 to 18.9 rems, with an average of 3.0 ± 4.1 rems, a geometric mean of $0.29 \text{ rems} \times 25$ (i.e., a GSD of 25). The time-and-proximity factors ranged from 0.0003 to 8.6 for the 85 cases that the NRC (1993b) evaluated. A preliminary investigation of the distribution of time-and-proximity factors for this incident shows that, while the distribution is very broad and skewed, it is not lognormal (Figure 3.2.1). There were a number of persons who evidently received roughly the same doses due to similar duties involving patient care, so these may need to be treated separately from doses to the others involved.

The 1983-84 Mexican ^{60}Co accident (IAEA 1989) was also a complete loss of control of a large source ($1.7\text{E}13$ Bq (450 Ci) of ^{60}Co ; $\Gamma = 3.6\text{E}-13 \text{ Sv/hr m}^2/\text{Bq}$ ($1.3 \text{ rem/hr m}^2/\text{Ci}$)). This incident resulted in an estimated 4000 persons exposed. From the data, the largest F_p was 1.2, the mean of lognormal fits (with GSDs of 13.8 (uniform weighting) and 22.5 (Finney weighting)) were 0.004 to 0.008. In this situation, symptoms of acute radiation syndrome would have prevented much larger F_p values, since the $F_p = 1.2$ corresponds to 700 rems.

The 1987 Goiânia, Brazil accident presents four different populations for analysis, and it is difficult to determine whether some individuals may appear in more than one of these populations. Depending on how many persons are included, the time-and-proximity factors for the public plus the 9 workers involved with the source average 0.20, 0.11, or 0.0034 hours at a meter, with the former population including 97 persons, the next, 249 persons, and the last, 2812 persons. The maximum whole-body time-and-proximity factor was 1.46, and minimum factors on the order of 10^{-9} . The maximum local irradiation time-and-proximity factor was 4.2 hours at a meter. Figure 3.2.2 shows a histogram and cumulative density function for whole-body time-and-proximity factors from the Goiânia accident for the 46 highest dose cases, with a roughly logarithmic horizontal axis.

The 1992 Korean Accident (NUREG 1405, NRC 1990) includes dose estimates for 24 individuals. These are clearly lognormally-distributed as shown in Figure 3.2.3. Doses are listed in the Appendix. The average time-and-proximity factor was 1.3 ± 3.7 hours at a meter, with a geometric mean of 0.11×7.9 (GSD = 7.9), and a range of 0.0063 to 14.7.

A Time-and-Proximity Factor analysis has been performed for 40 accidents involving 231 individual doses. The complete results (231 lines of data) are presented in the Appendix. The results are shown in Figure 3.2.4 and given in Table 3.2.2.

3.2.4 Censoring of factors for high-dose accidents by the appearance of clinical symptoms of acute irradiation

In many cases, accidents were discovered by the appearance of clinical symptoms of acute irradiation. In many cases, persons stopped receiving any more dose because they died. In such cases, it can be concluded that the accident did not run its normal course. A plot of time-and-proximity factors versus source strength, ΓA , in sieverts per hour at 1m from an unshielded source, is shown in Figure 3.2.5. It shows a decreasing relationship between the time-and-proximity factor and source strength. In most cases, the incidents involving a source with activity greater than 10,000 Ci resulted in fatalities. For many of the strong source accidents, medical symptoms were the first sign of an accident. The column in Table 3.2.1 labeled "How Terminated" shows "Med" if medical symptoms appeared, and "Rad" if the accident was discovered by other means, usually by radiation measurements or by discovery of missing sources through inventory or malfunction.

Only a few accidents appear to have run their full course, that is, delivered all the dose they ever would have. The 1992 Indiana, PA accident and the 1990 source shipment from Korea fall into this category. In each case, the accident was essentially over when it was discovered. These accidents are particularly valid for assessment of improper transfer and disposal scenarios.

3.2.5 Collective Time-and-Proximity Factors for Accidents

The collective dose can be determined using a Collective Time-and-Proximity Factor for an accident. Values of Time-and-Proximity Factors are given in Table 3.2.6 for the accidents at Goiânia, Brazil; Indiana, PA; and the Korea-USA incident based on collective dose due to external irradiation (UNSCEAR 1993; NRC 1992; NRC 1990). Notice that the Indiana, PA accident was an order of magnitude more serious than the Brazilian accident, and the Korea-USA shipment incident was a factor of 3 more serious, in terms of Time-and-Proximity Factors. That is, had the Brazilian source been involved in the Indiana or Korea accidents, the collective doses would have been 10 times and 3 times higher, respectively, than they were at Goiânia. Using a similar rationale, if any of these three accidents had occurred with a small, generally licensed source, the collective dose would have been lowest for the Goiânia-like accident, and highest for the Indiana, PA-like accident.

3.2.6 Conclusions

Distributions of Time-and-Proximity Factors from historical accidents can be used in probabilistic risk analyses for both whole-body and local irradiation from external sources. An analysis of 42 accidents for which source identity and strength are available show that the

average accident victim gets a whole body dose equal to that from being at 1 meter from the accident's unshielded source for an hour. The average accident is characterized by a value of 46 hours at a meter. In other words, the population-weighted average is about 1 hour at a meter, while the accident-weighted average is 117 hours at a meter. Clearly, the accidents with large numbers of victims (e.g., Goiânia and Juarez) dominate the former average. The maximum value seen for whole-body doses is about 700 hours at a meter. The average, geometric mean, and maximum values for local irradiation are 3100, 60 and 24,000 hours at a meter, respectively.

Such distributions should be used in probabilistic risk analyses to determine likely distributions of risks or doses from improper transfer and disposal scenarios for generally licensed devices.

Accidents that were terminated due to the appearance of clinical symptoms of acute irradiation have less value for risk analyses than accidents that were terminated by other means, or never terminated.

The current AEOD incident database does not contain the kinds of information needed to perform this analysis. It is recommended that the database either be modified to include this information, or a separate database be created. There is a great deal of work to be done to refine these preliminary analyses, extend them to additional accidents, and develop the logical framework for extrapolating to other kinds of sources and scenarios.

Table 3.2.1. Radiation Accidents involving external exposures. See text.

Code	Year	Nuclide	Source Type	Activity, A (Ci)	Activity, A (Bq)	#Public	#Workers	#Cleanup Workers How Terminated	WB-brief	WB-protracted	Local-brief	Local-protracted	Intakes	Removed from Shield?	Source Damaged?	Reference	Gamma (Sv/h [m ² /Bq])
ALG78	78	Ir-192	IndRad	25	9.25E+11	7	0	0 Med	1	1				1		Jammet et al. 1980a, 1980b	1.60E-13
AUS70	70	Ir-192	IndRad	22	8.14E+11	0	2	0 Rad	1							Brown and McNeill 1971	1.60E-13
BAN85	85	Ir-192	IndRad	50	1.85E+12	0	1	0 Med	1							Jalil and Mollia 1989	1.60E-13
BRA87	87	Cs-137	Tele	1400	5.18E+13	83	9	Med	1	1	1		1	1	1	Ramalho et al. 1988 Tables 1&2 (97 persons for whom cytogenetic dosimetry was done)	9.25E-14
BRA87	87	Cs-137	Tele	1400	5.18E+13	240	9	Med		1				1	1	IAEA 1988: lognormal fit to 50 individual doses from Fig. 9 and 199 zeroes (249 total, p. 117)	9.25E-14
BRA87	87	Cs-137	Tele	1400	5.18E+13	2803	9	Med	1	1	1		1	1	1	Lushbaugh et al. IAEA-CN-51-92 p401	9.25E-14
BRA87	87	Cs-137	Tele	1400	5.18E+13			583 Med		1				1	1	IAEA 1988 p. 116 External doses for 583 Cleanup Workers	9.25E-14
CA79	79	Ir-192	IndRad	28	1.04E+12	0	11	0 Rad	1					1		Ross 1980	1.60E-13
CZE66	66	I-131	Medical	2.25	8.33E+10	0	0	16 Rad		1				1	1	Carach et al. 1967	7.63E-14
CZE73	73	Co-60	Tele	2973	1.10E+14	0	2	0 Rad	1							Klener et al. 1986	3.70E-13
FRG68	68	Ir-192	IndRad	7.8	2.89E+11	0	1	0 Med	1		1					Chone et al. 1970	1.60E-13
FRG72	72	Ir-192	IndRad	29.7297	1.10E+12	0	1	0 Rad	1							UNSCEAR 1988 p. 416	1.60E-13
FRG81	81	Co-60	Tele	2594.59	9.60E+13	0	2	0 Rad	1		1					Stephan et al. 1983	3.70E-13
IND68	68	Ir-192	IndRad	1.4	5.18E+10	0	1	0 Rad	1		1			1		Annamalai et al. 1978	1.60E-13
ISR90	90	Co-60	Steril	340541	1.26E+16	0	1	0 Med	1		1					IAEA 1993	3.70E-13
ITA75	75	Co-60	Steril	36000	1.33E+15	0	1	0 Rad	1		1					Parmentier et al. 1980	3.70E-13
JCH59	59	Co-60	IndRad	1.75	6.48E+10	1	0	0 Rad	1		1			1		Elliott 1960	3.70E-13
JPN71	71	Ir-192	IndRad	5.26	1.95E+11	5	1	0 Rad		1	1			1		Hirashima et al. 1980	1.60E-13
KOR90	90	Ir-192	IndRad	4	1.48E+11	0	19	5 Rad	1	1				1		NUREG-1405	1.60E-13
KY76	76	Ir-192	IndRad	78	2.89E+12	0	1	0 Rad	1		1			1		Jacobson et al. 1977	1.60E-13
LA78	78	Ir-192	IndRad	100	3.70E+12	0	1	0 Med						1		Scot. 1980	1.60E-13

Table 3.2.1. (cont.) Radiation Accidents involving external exposures. See text.

Code	Year	Nuclide	Source Type	Activity, A (Ci)	Activity, A (Bq)	#Public	#Workers	#Cleanup Workers	How Terminated	WB-brief	WB-protracted	Local-brief	Local-protracted	Intakes	Removed from Shield?	Source Damaged?	Reference	Gamma (Sv/h [m ² /Bq])
MEX62	62	Co-60	IndRad	5	1.85E+11	6	0	0	Med		1	1			1		Andrews 1963	3.70E-13
MEX83	83	Co-60	Tele	430	1.59E+13	4000	100	?	Rad						1	1	Listar 1984; IAEA 1989	3.70E-13
MOR84	84	Ir-192	IndRad	16.2162	6.00E+11	28	0	0	Med		1	1			1		Marshall 1984	1.60E-13
NJ74	74	Co-60	Steril	120000	4.44E+15	0	1	0	Rad	1							Bariotta 1980	3.70E-13
NJ77	77	Co-60	Steril	500000	1.85E+16	0	1	0	Rad	1							Bariotta 1980	3.70E-13
NOR82	82	Co-60	Steril	65720	2.43E+15	0	1	0	Med	1							Flatby et al. 1983	3.70E-13
NY83	83	Co-60	IndRad	25	9.25E+11	0	10	0	Rad		1				1	1	NRC I&E Notice 83-16	3.70E-13
PA92	92	Ir-192	Brachy	3.7	1.37E+11	94	0	0	Rad	1	1	1			1		NUREG-1480	1.60E-13
PRC63	63	Co-60	IndRad	10	3.70E+11	7	0	0	Med						1		Gen-yao et al. 1980	3.70E-13
PRC80	80	Co-60	Steril	53000	1.96E+15	0	1	0	Med	1					1		Gen Yao Ye et al. 1990	3.70E-13
PRC85	85	Cs-137	IndRad?	10	3.70E+11	3	0	0	Med		1	1			1		Gen Yao Ye et al. 1990	9.25E-14
PRC86	86	Co-60	(1)	6888	2.55E+14	2	0	0	?	1							Gen Yao Ye et al. 1990	3.70E-13
PRC87	87	Co-60	Steril	89000	3.29E+15	0	1	0	?	1							Gen Yao Ye et al. 1990	3.70E-13
PRC92	92	Co-60	Expt	12	4.44E+11	18	0	0	Med		1	1			1		Nenot 1993	3.70E-13
SAF77	77	Ir-192	IndRad	6.75676	2.50E+11	3	0	0	?Med	1		1			1		UNSCEAR 1988 p. 416	1.60E-13
SAL89	89	Co-60	Steril	18000	6.66E+14	0	3	0	?Med	1		1					Littlefield et al. 1991	3.70E-13
SCO69	69	Ir-192	IndRad	25	9.25E+11	0	1	0	Rad	1		1					Harrison et al. 1973	1.60E-13
TN71	71	Co-60	Expt	7700	2.85E+14	0	1	0	Rad	1							Wade 1972, Vodopick and Andrews 1980	3.70E-13
TRK76	76	Co-60	Tele	2260	8.36E+13	0	5	0	Rad	1		1			1		Yalcintas et al. 1980	3.70E-13
TX72	72	Cs-137	IndRad	4	1.48E+11	1	0	0	Med		1		1		1		Collins 1980	9.25E-14
UK77	77	Ir-192	IndRad	21.6216	8.00E+11	0	1	0	Rad	1							UNSCEAR 1988 p. 416	1.60E-13
UK81	81	Cs-137	Brachy	0.12	4.14E+09	0	0	1	Rad	1							Heaton and Murray 1982	9.25E-14
WA76	76	Am-241	Dfns	343.337	1.27E+13	0	1	10	Rad						1	1	McMurray 1983	7.57E-15
WI61	61	Co-60	Expt	200	7.40E+12	0	1	0	Rad	1		1			1		Ross et al. 1962	3.70E-13
			(1) "Radiation Station"															

Table 3.2.1. (cont.) Radiation Accidents involving external exposures. See text.

Code	Year	Nuclide	Source Type	Activity, A (Ci)	Activity, A (Bq)	#Public	#Workers	#Cleanup Workers How Terminated	WB-brief	WB-protracted	Local-brief	Local-protracted	Intakes	Removed from Shield?	Source Damaged?	Reference	Gamma (Sv/h [m ² /Bq])
Minimum				0.12	4.44E+09	0	0	0									7.57E-15
Maximum				500000	1.85E+16	4000	100	583									3.7E-13
Average				2.8E+04	1.029E+15	166	4.545	15									2.38E-13
Std Dev				9.1E+04	3.358E+15	726.9	15.24	90.97									1.24E-13
GeoMean				1.9E+02	6.85E+12												0
Median				50	1.85E+12	0	1	0									1.6E-13
Mode				1400	5.18E+13	0	1	0									3.7E-13
Number				45	45	44	44	41	31	15	16	5	3	28	8		45

Table 3.2.1. (cont.) Radiation Accidents involving external exposures. See text.

Code	Gamma*Act. (Sv/h @ 1m)	TIME AND PROXIMITY FACTORS, F(t,p)						Whole Body DOSES (Sv)						Skin DOSE	
		Average (h @ 1m)	Std Dev (h @ 1m)	Minimum (h @ 1m)	Maximum (h @ 1m)	Geometric Mean (h @ 1m)	Geo. Std. Deviation	Average	Standard Deviation	Minimum	Maximum	Geometric Mean	Geo. Std. Deviation	Max Local F(t,p) (h @ 1m)	Max Local Dose (Sv)
ALG78	0.148	88.108	88.176	6.7568	270.27	47.297	4	13.04	13.05	1	40	7	4	675.68	100
AUS70	0.13024	1.6892		1.2285	3.5319	2.083		0.22		0.16	0.46	0.271293			
BAN85	0.296	8.4459		6.7568	10.135	8.2753		2.5		2	3	2.44949		81.081	24
BRA87	4.7915	0.2024	1.9827	0.0209	1.4609	0.0605	4.7	0.97	9.5	0.1	7	0.29	4.7	4.1741	20
BRA87	4.7915	0.1119	3.2975	2E-06	1.4609	0.0038	13	0.536	15.8	1.02E-05	7	0.0182	13.49	4.1741	20
BRA87	4.7915	0.0034	0.6261	1E-09	1.4609			0.0163	3	7.00E-09	7			4.1741	20
BRA87	4.7915	0.0002	0.0003	5E-06	0.0033	0.0001	2.8	1.07E-03	1.48E-03	2.426E-05	0.016	6.2E-04	2.82	4.1741	20
CA79	0.16576	0.7264	1.5601	0.0121	5.2787	0.1647	5.8	0.204	0.2586	0.002	0.875	0.0273	5.83	24131	4000
CZE66	0.00635	0.0236		0.0079	0.0394	0.0176		0.00015		0.00005	0.00025	0.000112			
CZE73	40.7	0.0184		0.0025	0.0344	0.0092		0.75		0.1	1.4	0.374166		2.457	100
FRG68	0.04618	32.484		21.656	43.313	30.627		1.5		1	2	1.414214		4331.3	200
FRG72	0.176	1.7045						0.3							
FRG81	35.52	0.0084		0.0056	0.0113	0.008		0.3		0.2	0.4	0.282843			
IND68	0.00829	156.85						1.3							
ISR90	4662	0.0032		0.0021	0.0043			15		10	20			15685	130
ITA75	492.84	0.0284						14						0.0487	24
JOH59	0.02396	1.0435						0.025						375.67	9
JPN71	0.03114	13.167	15.222	3.2114	42.711	8.4138	2.7	0.41	0.474	0.1	1.33	0.262	2.66		
KOR90	0.02368	1.2965	3.7162	0.0063	14.717	0.1098	7.9	0.0307	0.088	0.00015	0.3485	0.0026	7.87		
KY76	0.46176	2.0573		0.8663	3.2484	1.6775		0.95		0.4	1.5	0.774597		32.484	15
LA78	0.592	0.0845						0.05						168.92	100

Table 3.2.1. (cont.) Radiation Accidents involving external exposures. See text.

Code	Gamma*Act. (Sv/h @ 1m)	TIME AND PROXIMITY FACTORS, F(t,p)						Whole Body DOSES (Sv)						Skin DOSE	
		Average (h @ 1m)	Std Dev (h @ 1m)	Minimum (h @ 1m)	Maximum (h @ 1m)	Geometric Mean (h @ 1m)	Geo. Std. Deviation	Average	Standard Deviation	Minimum	Maximum	Geometric Mean	Geo. Std. Deviation	Max Local F(t,p) (h @ 1m)	Max Local Dose (Sv)
MEX62	0.06845	446.17	184.51	175.31	686.63	408.04	1.7	30.54	12.63	12	47	27.93	1.7		
MEX83	5.8867	0.0041	0.1276	8E-09	1.1891	0.0001	14	0.024	0.751	5.0E-08	7	7.7E-04	13.8		
MOR84	0.096	123.02	75.417	10.417	260.42	92.188	2.6	11.81	7.24	1	25	8.85	2.6		
NJ74	1642.8	0.0025						4.1							
NJ77	6845	0.0003						2.1							
NOR82	899.707	0.0445						40							
NY83	0.34225	0.0044		0.0015	0.0058	0.0029		0.0015		0.0005	0.002	0.001			
PA92	0.0219	1.3313	1.8718	0.0003	8.6286	0.1329	25	0.02916	0.041	5.6E-06	0.189	0.00291	25.3		
PRC63	0.1369	170.2	227.17	14.609	584.37	75.237	4.1	23.3	31.1	2	80	10.3	4.1		
PRC80	725.57	0.0072				0.0072		5.22				5.22			
PRC85	0.03423	262.97	29.218	233.75	292.18	261.88		9	1	8	10	8.952809	1.12		
PRC86	94.2967	0.0323		0.0276	0.0371	0.032		3.05		2.6	3.5	3.016621			
PRC87	1218.41	0.0011		0.0011	0.0012	0.0012		1.35		1.35	1.46	1.403923			
PRC92	0.16428	22.583	33.479	1.887	121.74	11.748	2.9	3.71	5.5	0.31	20	1.93	2.89		
SAF77	0.04	11.925	14.825	2.5	29	6.75	3.6	0.477	0.593	0.1	1.16	0.27	3.63	2500	100
SAL89	246.42	0.0215	0.0108	0.013	0.0337	0.0198	1.6	5.3	2.67	3.2	8.3	4.89	1.62		
SCO69	0.148	4.0541						0.6						1351.4	200
TN71	105.413	0.0247						2.6							
TRK76	30.9394	0.0001	0.0001	2E-06	0.0003	3E-05	10	0.00346	0.00411	0.000047	0.00978	0.000945	10.1		
TX72	0.01369	730.46						10							
UK77	0.128	0.7813						0.1							
UK81	0.00041	0.487						0.0002							
WA76	0.09617	0.052						0.005							
WI61	2.738	0.9131						2.5							

Table 3.2.1. (cont.) Radiation Accidents involving external exposures. See text.

Code	Gamma*Act. (Sv/h @ 1m)	TIME AND PROXIMITY FACTORS, F(t,p)						Whole Body DOSES (Sv)						Skin DOSE	
		Average (h @ 1m)	Std Dev (h @ 1m)	Minimum (h @ 1m)	Maximum (h @ 1m)	Geometric Mean (h @ 1m)	Geo. Std. Deviation	Average	Standard Deviation	Minimum	Maximum	Geometric Mean	Geo. Std. Deviation	Max Local F(t,p) (h @ 1m)	Max Local Dose (Sv)
Minimum	0.00041	0.0001	0.0001	1E-09	0.0003	3E-05	1.6	0.00015	0.00148	7E-09	0.00025	0.000112	1.12	0.0487	9
Maximum	6845	730.46	227.17	233.75	686.63	408.04	25	40	31.1	12	80	27.93	25.3	24131	4000
Average	3.8E+02	46	38	17	82	34	6.7	4.62	5.76	1.57	10.21	3.07	6.37	3084.5	317.6
Std Dev	1.2E+03	133	67	53	176	90	6.3	8.47	8.19	3.08	17.94	5.79	6.26	6848.8	984.1
GeoMea	1.3E+00	0.32						0.43		0.018	1.42	0.19	4.45	91.2	59.7
Median	3.0E-01	0.49	3.51	0.01	3.25	0.1213	4.1	0.95	1.835	0.16	2	0.332083	4	125	62
Mode	4.8E+00			6.76	1.46			2.5		0.1	7			4.2	100
Number	45	45	18	29	29	28	16	45	18	29	29	28	17	16	15

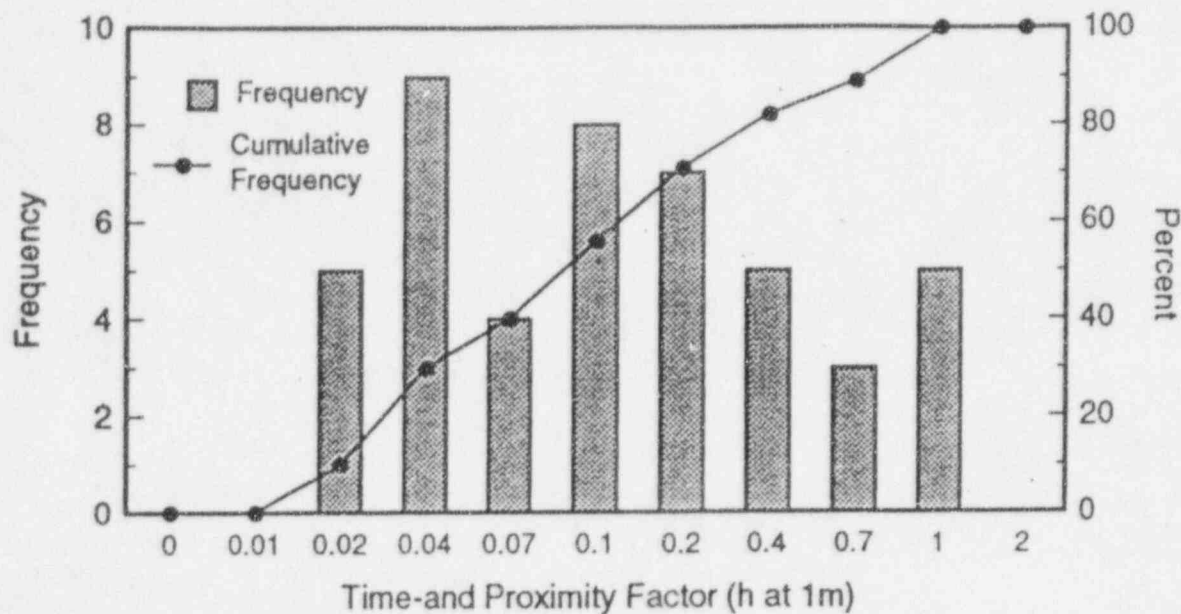


Figure 3.2.1. Frequency distribution of time-and-proximity factors from the 46 highest dose cases from the 1987 Goiânia accident on a roughly logarithmic horizontal scale. (IAEA, 1988a; Figure 9)

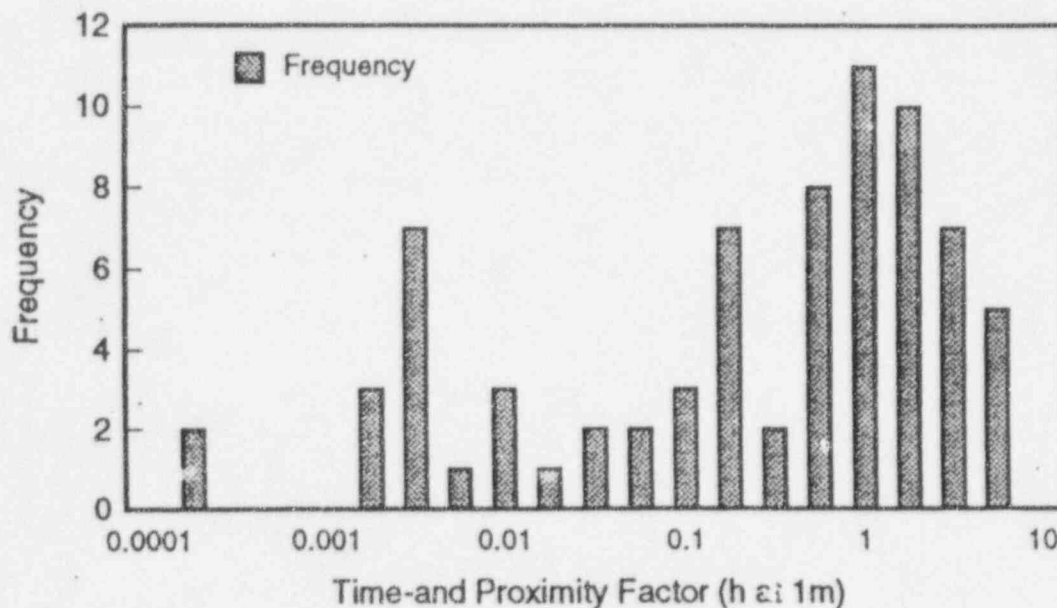


Figure 3.2.2. Frequency Distribution of Time-and-Proximity Factors for the 1992 Indiana, PA ¹⁹²Ir therapy misadministration accident on a logarithmic horizontal scale. (NRC, NUREG-1480, 1993b)

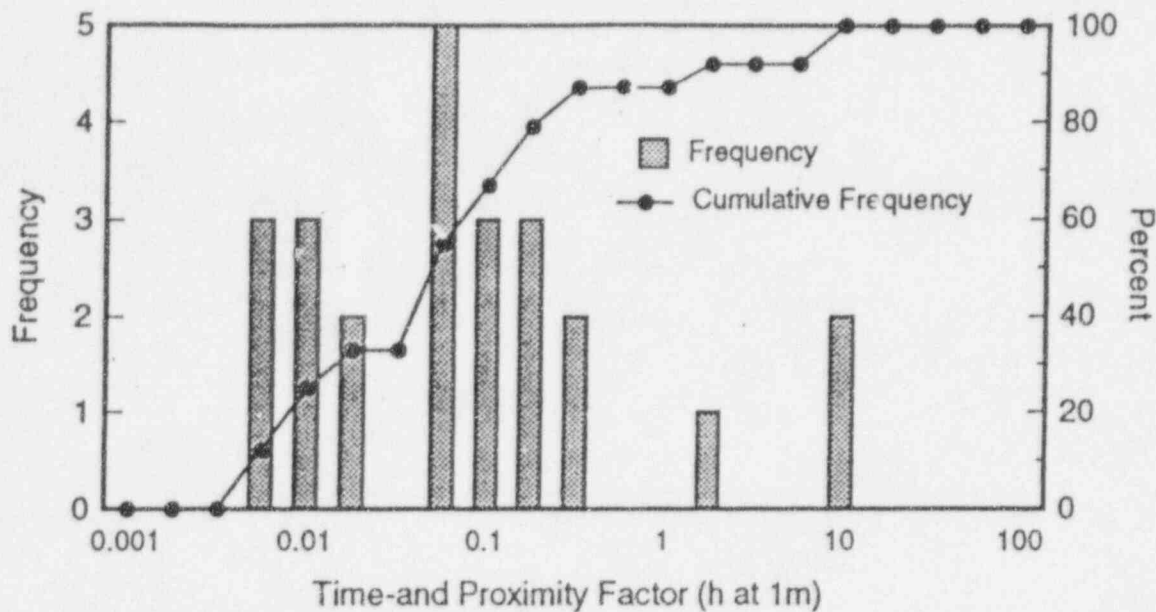


Figure 3.2.3. Frequency and Cumulative Frequency of Time-and-Proximity Factors for the 1990 Korea-USA ^{192}Ir source shipment accident on a logarithmic horizontal scale. (NRC, NUREG-1405, 1990)

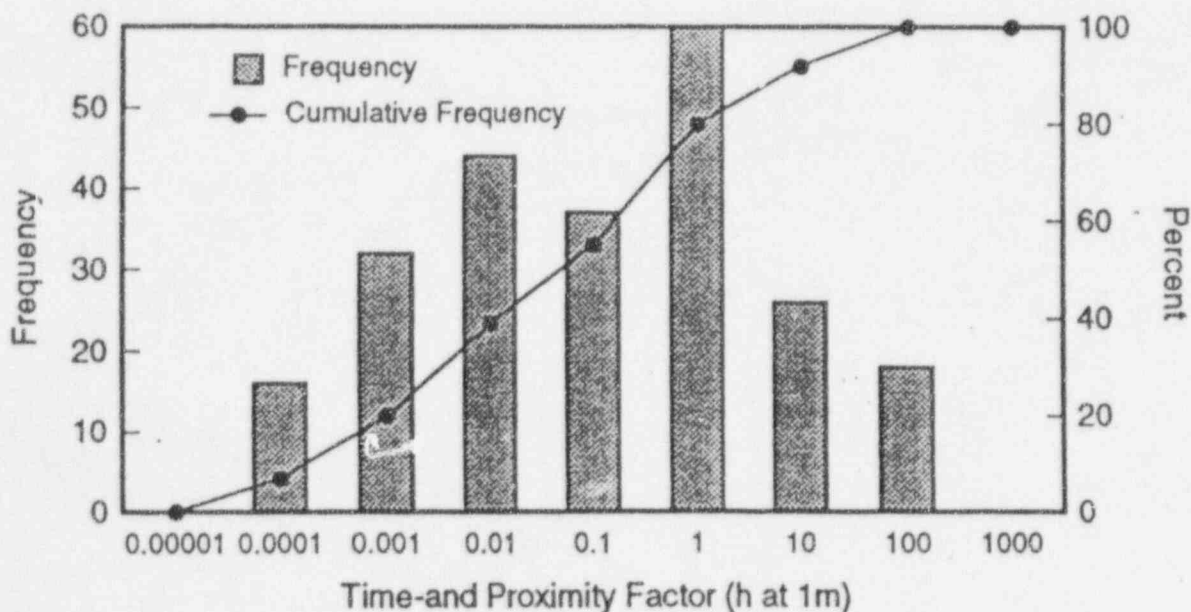


Figure 3.2.4. Analysis of 40 radiation accidents involving sealed sources, involving 231 individuals for whom doses were available. The Goiânia and Juarez accidents are excluded because the large numbers of victims of those accidents would dominate the graphs.

Table 3.2.2. Summary of Whole Body Time-and-Proximity Factors from a survey of 40 accidents involving sealed sources. The accidents at Goiânia, Brazil, and Juarez, Mexico are not included. The very large numbers of persons involved would dominate the summary statistics.

Quantity	Value	Units
Number of accidents	40	-
Number of individual doses	231	-
Average	30	hours at a meter
Standard Deviation	100	hours at a meter
Geometric Mean	0.37	hours at a meter
Geometric Standard Deviation	43	-
Median	0.46	hours at a meter
Mode	0.00068	hours at a meter
Minimum	0.000112	hours at a meter
Maximum	730	hours at a meter

Table 3.2.3. Collective Time-and-Proximity Factors for several sealed source accidents.

Population Group	Collective External Dose (Person-Gy)	Source Activity (Bq)	Source Strength (Sv/h at 1 m)	Collective Time-and-Proximity Factor (hours at 1 m)
All persons in Goiânia	56.3*	5.09E13	4.71	11.96
4 who died in Goiânia	14.9*	5.09E13	4.71	3.17
85 persons Indiana, PA	2.5	1.37E11	0.022	113
24 persons, Korea - USA	0.74	9.2E10	0.024	31.2

*Collective doses from ¶172, UNSCEAR 1993. These doses are probably too low by a factor of 2, from comparison with Figure 9 of IAEA 1988b (74 person-Gy listed there among 50 most highly exposed).

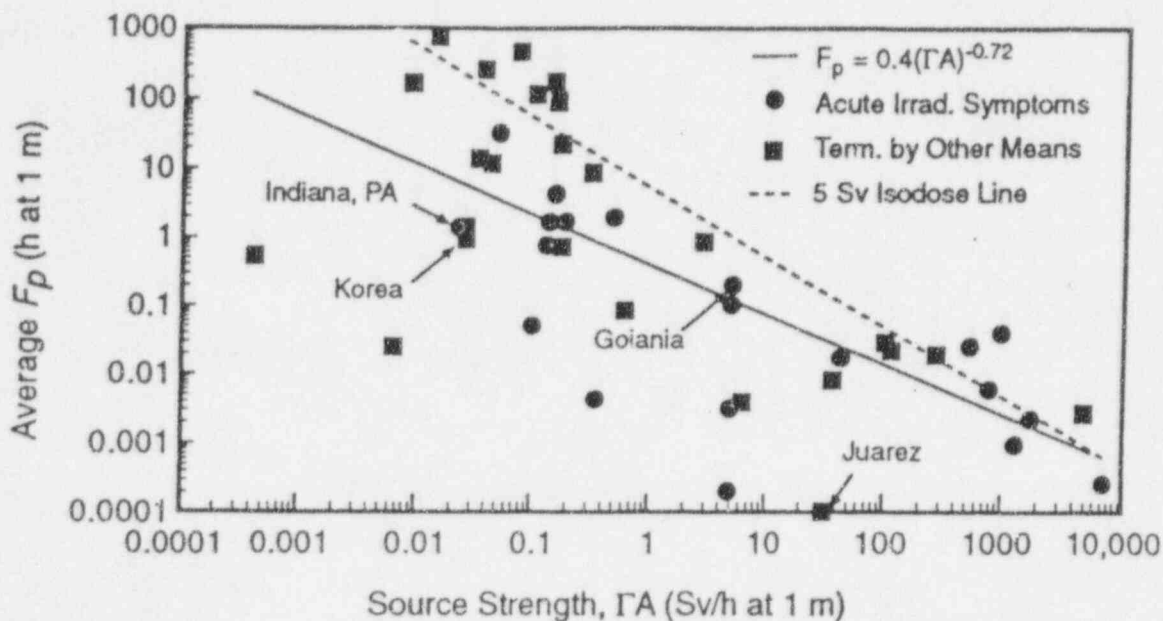


Figure 3.2.5. Average whole-body time-and-proximity factors (F_p) for 41 sealed source accidents as a function of source strength on a logarithmic horizontal scale. Most points shown above the 5-Sv (500 rem) isodose line were fatal accidents.

3.3 INTERNAL EXPOSURES

There are substantial differences between cases involving only external irradiation and cases where internal contamination has resulted. First, it is relatively rare for internal contamination to occur in non-reactor-type accidents. Second, early clinically recognizable symptoms of the internal radiation exposure are rarely seen in internal contamination cases. In most instances, any deterministic effects, and certainly the stochastic effects, will display themselves only at a time in the distant future, if at all. As a result, accidents may go unreported. Third, medical treatment procedures can be used to induce decorporation of the internal radionuclide burden, resulting in a reduction in radiation dose.

3.3.1 Theoretical Basis for Fraction-Taken-In

Radioactive material may gain entry to the body by inhalation, ingestion, or absorption through skin or wounds. The amount of radioactive material, or fraction of the total radioactive material available, that enters the body is a major factor in the radiation dose resulting from the intake. The approach taken in this report is to perform a quantitative, probabilistic characterization of historical radiation accidents in which the fraction-taken-in is known or can be calculated and then to generalize the results to generally licensed radioactive

devices.

It should be noted that since, in many cases, it is generally not known that an accident has occurred, the majority of early biological data needed to calculate radiation doses from the incident will not have been collected. In some cases, the lack of early biological data has hampered the use of historical records for determining the fraction-taken-in values.

For internal contamination cases, Brodsky (1980) has proposed the fraction-taken-in approach to characterizing internal contamination cases. Depending on how the data are reported in the literature, the fraction-taken-in may be available directly, or may be calculated from reported doses. In the former case,

$$F_{t,i} = \frac{A_i}{A}, \quad (11)$$

where $F_{t,i}$ denotes the fraction-taken-in for the i th person;
 A_i denotes the activity taken in by the i th person; and
 A denotes the source activity.

If activities taken in are not published, the fraction-taken-in may be calculated from 50-year committed effective dose equivalent to the i th person using

$$H_{50,i} = \frac{A F_{t,i}}{ALI} \cdot 5 \text{ rems}, \quad (12)$$

where ALI is the 1980 stochastic annual limit on intake (EPA 1988); and
5 rems is the 50 year committed effective dose equivalent resulting from an intake of
1 ALI .

Solving for $F_{t,i}$, we have

$$F_{t,i} = \frac{H_{50,i}}{5 \text{ rems}} \frac{ALI}{A}, \quad (13)$$

where $H_{50,i}$ denotes the 50 year committed effective dose equivalent received by individual i ;
 ALI denotes the annual limit on intake; and
5 rems is the committed effective dose equivalent associated with an intake of a 1980
 ALI .

If 1991 ALIs (based on 2 rems effective dose) are used, $F_{t,i}$ is

$$F_{t,i} = \frac{E_i}{2 \text{ rems}} \frac{ALI}{A}, \quad (14)$$

where E_i denotes the effective dose to the i th person from the incident.

3.3.2 Results of Analysis of Accidents for Fraction-Taken-In

The activity of the radioactive source at the time of the accident is an essential piece of information. The information needed for this analysis was generally unavailable in the NRC/AEOD incident database. A literature review of available data on accidents involving radioactive materials was conducted. The results of this review are presented in Table 3.3.1. The historical cases reported in this table are only those for which the source activity data were obtainable from the literature sources available, and are the same as those included in Table 3.2.1. For each incident, the locate, date, nuclide, source activity, incident type, and reference were reported, as well as whether or not external or internal exposure(s) were involved, the number of deaths resulting from such exposure(s) and the calculated fraction-taken-in (using Eq. 11).

Of the 60 radiation accidents reviewed and reported in Table 3.3.1, only 17 incidents, or (28%), involved cases of internal contamination. The low percentage of internal contamination cases is most probably due to the fact that many of the reported radiation accidents involved sealed, radiographic sources that maintained their structural integrity during the accident. As a result, no radioactive material was available for intake into the body, and the fraction-taken-in is zero. Of the 4364 individuals involved in the accidents listed in Table 3.3.1, 90 individuals¹ (2.1%) received internal contamination as a result of the indicated accident. The calculated fraction-taken-in for these individuals ranges from two in ten-thousand to twenty in a billion ($2\text{E-}4$ to $2\text{E-}8$). In only one case of an industrial accident involving an unsealed source of tritium (Lloyd 1986) was the fraction-taken-in greater than this range (in that case it was $2\text{E-}2$).

In most cases, the distribution of the fraction-taken-in, and thus the resultant dose, is lognormally distributed. This is best illustrated in Figure 3.3.1 using data from the 1987 Goiânia, Brazil accident which involved a 1,375 Ci ^{137}Cs source (Brandao-Mello 1987). The log probability analysis of the 20 highest-exposed individuals indicated in Table 3.3.1 indicates that the distribution of fraction-taken-in values is lognormal with a mean of $2.6\text{E-}6$ and a geometric standard deviation (GSD) of 5.4 (Figure 3.3.1). When a total of 77 individuals who had internal contamination from this accident were considered (IAEA 1988b), the distribution of fraction-taken-in values was lognormal with a mean of $6.2\text{E-}6$ and a GSD of 21 (Figure 3.3.2). The committed effective dose equivalents determined for these 77 individuals are also lognormally distributed with a mean of 0.3 Gy and a GSD of 11.2 (Figure 3.3.3). When 194 of the cleanup workers at Goiânia were evaluated, their fraction-taken-in values were estimated to range from $7\text{E-}11$ to $2\text{E-}15$.

In the most recent publication of the UNSCEAR report (UNSCEAR 1993), the collective dose for all persons involved in the Goiânia incident and for the 4 individuals who died as a

¹ For the 77 cases of internal contamination resulting in the Goiânia accident (Brandao-Mello 1991b), only 20 highest internal exposures are listed.

result of the incident were given. As shown in Table 3.3.2, this data can then be used as a direct application of Eq. 13. By using the reported collective internal doses of 3.7 and 2.3, respectively, the resulting fraction-taken-in values are $9\text{E-}6$ and $5\text{E-}6$, respectively. The estimated collective doses for this incident are within the range of values calculated for individuals.

Figure 3.3.4 provides an illustration of the frequency of values for fraction-taken-in for the 60 radiation incidents reviewed. In addition, both the 0.3 maximum value and the realistic range ($10\text{E-}6$ to $10\text{E-}5$) used in the ORAU report are included on the plot. It is interesting to note that the so called 'realistic' range of values used in the ORAU report (obtained from published literature on transportation accidents) falls in the range of values observed from actual accidents. However, the maximum value of 0.3 used by ORAU falls above the observed range. Thus, the ORAU approach to calculation the doses from their selected scenarios might be applicable today, if two things are done: 1) the more realistic values for fraction-taken-in should be used, and 2) CEDE and TEDE should be calculated instead of just organ doses.

3.3.3 Conclusions and Generalizations of Results to Improper Transfer and Disposal Scenarios for Generally Licensed Devices

When considering all reported non-reactor-type radiation accidents listed in Table 3.3.1, the fraction-taken-in was found to range from $2\text{E-}4$ to $2\text{E-}8$. The internal contamination cases upon which this range of values is based resulted largely from either industrial accidents involving unsealed sources of radioactive material or intentional destruction of licensed radiation sources.

When generalizing the results of this historical review, it is difficult to support the "arbitrarily chosen" value of 0.3 for the fraction-taken-in for generally licensed devices used in the ORAU Report. Thus, the most defensible values to use for the fraction-taken-in are distributions in the range of $2\text{E-}4$ to $2\text{E-}8$, except for cases involving sealed tritium sources in generally licensed devices.

TABLE 3.3.1. Summary of Radiation Accidents Used to Determine Fraction-Taken-In Values

CODE	LOCATION	DATE	HUCLIDE	SOURCE ACTIVITY	UNITS	INCIDENT TYPE	EXTERNAL INVOLVED	INTERNAL INVOLVED	# INVOLVED	# DEATHS	FRACTION TAKEN IN	REFERENCE
UK44	UK	1944	Ra-226	0.005	CI	Brachytherapy	X	—	2	0	0	Smith 1982
JOH50	JOH	1959	Co-60	2	CI	Industrial Accident	X	—	1	0	0	Elliott 1980
WI61	WI	1961	Co-60	200	CI	Experimental Accident	X	—	1	0	0	Ross 1962
MEX62	Mexico	1962	Co-60	5	CI	Industrial Accident	X	—	5	4	0	Andrews 1963
PA63	Phil. PA	1963	S-35	1.27	CI	Explosion-Vial	X	X	1	0	1.0E-05	Mass 1963
PRC63	PR of China	1963	Co-60	10	CI	Lost source	X	—	6	2	0	Mettler 1990
USA64	U.S.A	1964	Am-241	0.05	CI	Lost source	X	X	2 a	0	1.8E-06	Cohen 1979
USA64	U.S.A	1964	Am-241	0.05	CI	Lost source	X	X	2 b	0	7.2E-07	Cohen 1979
FRG64	Germany	1964	H-3	?	CI	Tritiated Paint Accident	—	X	4	1	?	Mettler 1990
CZE66	Czechoslovakia	1966	I-131	2.25	CI	Medical/Aircraft accident	X	X	16	0	0	Carach 1966
NY67	Univ of Rochester	1967	Ir-192	2	CI	Hot cell leak	X	X	2 a	0	1.4E-04	Cool 1962
NY67	Univ of Rochester	1967	Ir-192	2	CI	Hot cell leak	X	X	2 b	0	2.1E-04	Cool 1962
USA67	USA	1967	Ir-192	70	CI	Industrial Accident	X	—	2	0	0	Bhusan 1973; Maxfield 1969
FRG68	Germany	1968	Ir-192	8	CI	Industrial Accident	X	—	1	0	0	Chone 1970
IND68	India	1968	Ir-192	1	CI	Industrial Accident	X	—	1	0	0	Annamalai 1978
ARG68	Argentina	1968	Cs-137	14	CI	Industrial Accident	X	—	1	0	0	Benhaon 1969
FRG68	Germany	1968	Ir-192	?	CI	Industrial Accident	X	—	1	0	0	Schneider 1969
IND69	India	1969	Ir-192	?	CI	Industrial Accident	X	—	1	0	0	Annamalai 1978
SCO69	SCO	1969	Ir-192	25	CI	Industrial Accident	X	—	1	0	0	Harrison 1973
UK69	UK	1969	Ir-192	24	CI	Industrial Accident	X	—	1	0	0	Harrison 1973
AUS70	AUS	1970	Ir-192	22	CI	Industrial Accident	X	—	2	0	0	Brown and McNeill 1971
UK70	UK	1970	Ir-192	22	CI	Industrial Accident	X	—	1	0	0	Purrott 1972
JPN71	Chiba, Japan	1971	Ir-192	5	CI	Industrial Accident	X	—	6	0	0	Kuratsuki 1977; Mettler 1990
TN71	Oak Ridge, TN	1971	Co-60	7,700	CI	Experimental Accident	X	—	1	0	0	Volopick 1974; Preston 1974; Fry 1990
UK71	UK	1971	Ir-192	5	CI	Industrial Accident	X	—	1	0	0	Purrott 1973
BUL72	Bulgaria	1972	Cs-137	5.3	CI	Suicide	X	—	1	1	0	Oliveria 1987
TX72	Texas	1972	Cs-137	4	CI	Intentional accident	X	—	1	0	0	Collins 1980
FRG72	Germany	1972	Ir-192	30	CI	Industrial Accident	X	—	1	0	0	Sagell 1975
CZE73	Czechoslovakia	1973	Co-60	2,973	CI	Med. Therapy Accident	X	—	2	0	0	Köner 1988
ME74	Middle East	1974	Ir-192	?	CI	Industrial Accident	X	—	1	0	0	Purrott 1976
NV74	EPA-Las Vegas	1974	Hg-203	0.03	CI	Accid. med. volatilization	?	X	2 a	0	3.0E-06	Brown 1975
NV74	EPA-Las Vegas	1974	Hg-203	0.03	CI	Accid. med. volatilization	?	X	2 b	0	2.5E-05	Brown 1975

TABLE 3.3.1. (cont.)

CODE	LOCATION	DATE	NUCLIDE	SOURCE ACTIVITY	UNITS	INCIDENT TYPE	EXTERNAL INVOLVED	INTERNAL INVOLVED	# INVOLVED	# DEATHS	FRACTION TAKEN IN	REFERENCE
N174	Parsippany, NJ	1974	Co-60	120,000	CI	Medical Steril. Accident	X	—	1	0	0	Stedley 1976; Fry 1990
IRQ75	Iraq	1975	Ir-192	62	CI	Industrial Accident	X	—	1	0	0	Lloyd 1979
ITA75	Italy	1975	Co-60	36,000	CI	Industrial Accident	X	—	2	1	0	Jammet 1980; Parmentier 1980; Oliveria 1987
USA76	USA	1976	Ir-192	95	CI	Industrial Accident	X	—	1	0	0	NUREG-0322 1977 / NUREG-0090-6 1977
USA76	USA	1976	Co-60	165	CI	Industrial Accident	X	—	1	0	0	NUREG-0322 1977 / NUREG-0090-6 1977
WA78	Hanford	1976	Am-241	340	CI	Explosion-Glovebox	X	X	1	0	3.2E-06	Held 1978; McMurray 1983; Fry 1990
KY76	Kentucky	1976	Ir-192	78	CI	Industrial Accident	X	—	1	0	0	Jacobson 1977
TRK78	Turkey	1976	Co-60	2,260	CI	Med. Therapy Accident	X	—	5	0	0	Yalcintas 1980
UK77	UK	1977	Ir-192	22	CI	Industrial Accident	X	—	1	0	0	Lloyd 1978
SA77	South Africa	1977	Ir-192	7	CI	Industrial Accident	X	—	1	0	0	Lloyd 1978; Besson 1980
NJ77	Parsippany, NJ	1977	Co-60	500,000	CI	Medical Steril. Accident	X	—	1	0	0	Stedley 1976; Fry 1990
ALG78	Algeria	1978	Ir-192	25	CI	Lost Industrial source	X	?	22	5	0	Mettler 1990
LA78	Louisiana	1978	Ir-192	100	CI	Industrial accident	X	—	1	0	0	Scott 1980
CA79	California	1979	Ir-192	26	CI	Industrial accident	X	—	11	0	0	Ross 1990
LA79	Louisiana	1979	Ir-192	?	CI	Industrial accident	X	—	3	0	0	Scott 1980
PRC80	PR of China	1980	Co-60	63,000	CI	Industrial accident	X	—	1	0	0	Ye 1990
MA80	BU Med Center	1980	Tc-99m	0.04	CI	Bolled vial to dryness	—	—	1	0	0	Evdoldmoff 1980
UK81	UK	1981	Ce-137	0.12	CI	Brachytherapy	X	—	1	0	0	Heaton and Murray 1982
NOR82	Norway	1982	Co-60	65,000	CI	Industrial Use	X	—	1	1	0	Reitan 1990
ME'83	Juarez, Mexico	1983	Co-60	450	CI	Lost Industrial source	X	—	4000	5	0	Usher 1986; IAEA 1989b
MOR84	Morocco	1984	Ir-192	16	CI	Lost Industrial source	X	X	28	8	?	Mettler 1990; Lushbaugh 1990
SW585	Switzerland	1985	H-3	50	CI	Industrial accident	—	X	5 a	0	2.0E-02	Lloyd 1986
SW585	Switzerland	1985	H-3	50	CI	Industrial accident	—	X	5 b	0	5.4E-04	Lloyd 1986
SW585	Switzerland	1985	H-3	50	CI	Industrial accident	—	X	5 c	0	7.0E-05	Lloyd 1986
SW585	Switzerland	1985	H-3	50	CI	Industrial accident	—	X	5 d	0	7.0E-05	Lloyd 1986
SW585	Switzerland	1985	H-3	50	CI	Industrial accident	—	X	5 e	0	7.0E-05	Lloyd 1986
BAN85	BAN	1985	Ir-192	50	CI	Industrial accident	X	—	1	0	0	Jell and Molis 1989
PRC85	PR of China	1985	Ce-137	10	CI	Lost Industrial Source	X	—	3	1	0	Ye 1990
PRC86	PR of China	1986	Co-60	6,868	CI	Industrial accident	X	—	2	0	0	Ye 1990

TABLE 3.3.1. (cont.)

CODE	LOCATION	DATE	NUCLIDE	SOURCE ACTIVITY	UNITS	INCIDENT TYPE	EXTERNAL INVOLVED	INTERNAL INVOLVED	# INVOLVED	# DEATHS	FRACTION TAKEN IN	REFERENCE
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 s	1	2.0E-05	Brandao-Mello 1991b
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 b	0	7.9E-06	Brandao-Mello 1991b
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 c	0	3.9E-06	Brandao-Mello 1991b
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 d	0	3.9E-06	Brandao-Mello 1991b
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 e	1	2.0E-06	Brandao-Mello 1991b
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 f	0	2.0E-06	Brandao-Mello 1991b
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 g	0	2.0E-06	Brandao-Mello 1991b
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 h	0	2.0E-06	Brandao-Mello 1991b
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 i	0	2.0E-06	Brandao-Mello 1991b
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 j	1	1.1E-06	Brandao-Mello 1991b
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 k	0	1.1E-06	Brandao-Mello 1991b
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 l	0	9.8E-07	Brandao-Mello 1991b
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 m	0	8.8E-07	Brandao-Mello 1991b
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 n	0	5.9E-07	Brandao-Mello 1991b
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 o	1	3.9E-07	Brandao-Mello 1991b
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 p	0	2.0E-07	Brandao-Mello 1991b
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 q	0	1.6E-07	Brandao-Mello 1991b
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 r	0	1.1E-07	Brandao-Mello 1991b
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 s	0	6.6E-06	Brandao-Mello 1991b
BRA87	Golanis, Brazil	1987	Ce-137	1,375	Cl	Lost Medical Source	X	X	20 t	0	1.6E-06	Brandao-Mello 1991b
PRC87	PR of China	1989	Co-60	89,000	Cl	Industrial accident	X	—	1	0	0	Ye 1990
SAL89	El Salvador	1989	Co-60	18,000	Cl	Stuck sterilizer source	X	—	3	1	0	Hurtado 1991
ISP90	Israel	1990	Co-60	340,541	Cl	Stuck sterilizer source	X	—	1	0	0	Ricks 1991; IAEA 1993
KOR90	Korea	1990	Ir-192	4	Cl	Industrial accident	X	—	24	0	0	NUREG-1405 1990
PA92	Pennsylvania	1992	Ir-192	4	Cl	Brachytherapy	X	—	94	1	0	NUREG-1480 1993

TABLE 3.3.2 Calculation of Fraction-Taken-In Using Collective Internal Doses From the Goiânia Brazil Accident

Population Group	Collective Internal Dose (Person-Gy)	Source Activity (Bq)	ALI for Cs-137 (Bq)	Collective Fraction-Taken-In
All persons in Goiânia	3.7	5.09E13	6E6	8.7E-6
4 who died	2.3	5.09E13	6E6	5.4E-6

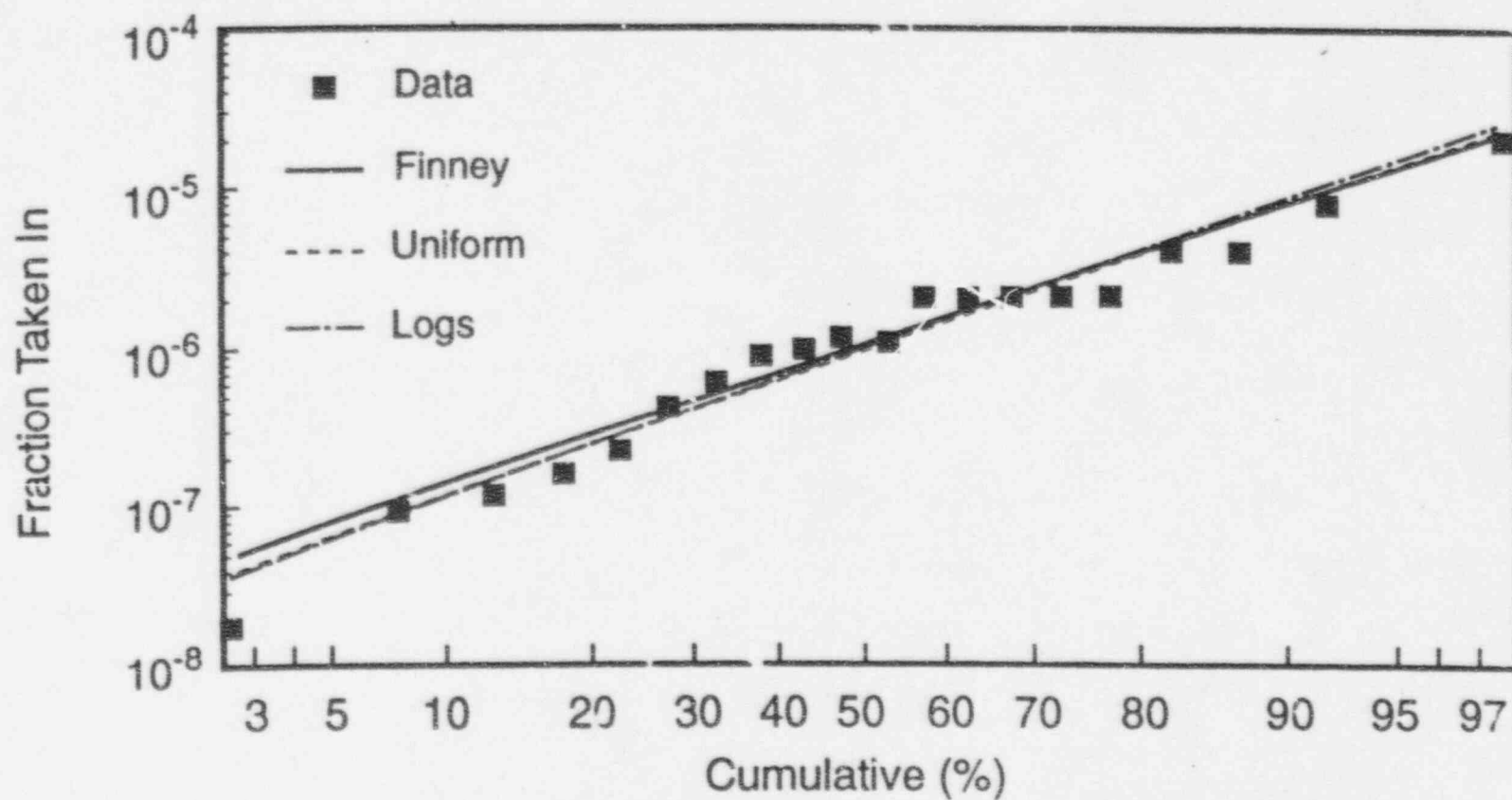


FIGURE 3.3.1. Log Probability Analysis of Fraction-Taken-In for 20 Highest-Exposed Individuals with Internal Contamination from the 1987 Goiânia ^{137}Cs Accident. Regression lines are uniform- or Finney-weighted.

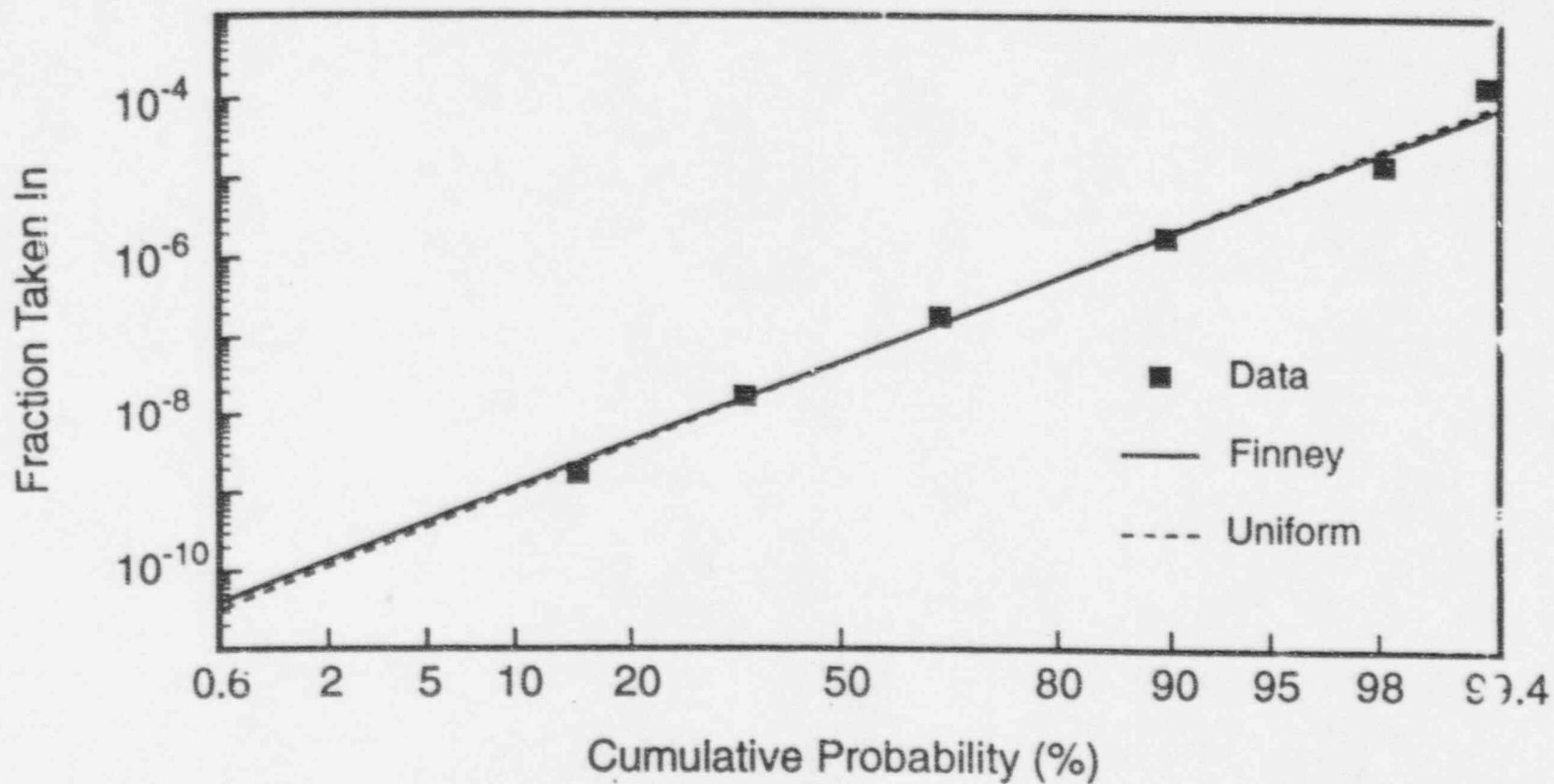


FIGURE 3.3.2. Log Probability Analysis of Fraction-Taken-In for 77 Individuals with Internal Contamination from the 1987 Goiânia ^{137}Cs Accident. Regression lines are uniform- or Finney-weighted. (IAEA, 1988a, from Figure 13)

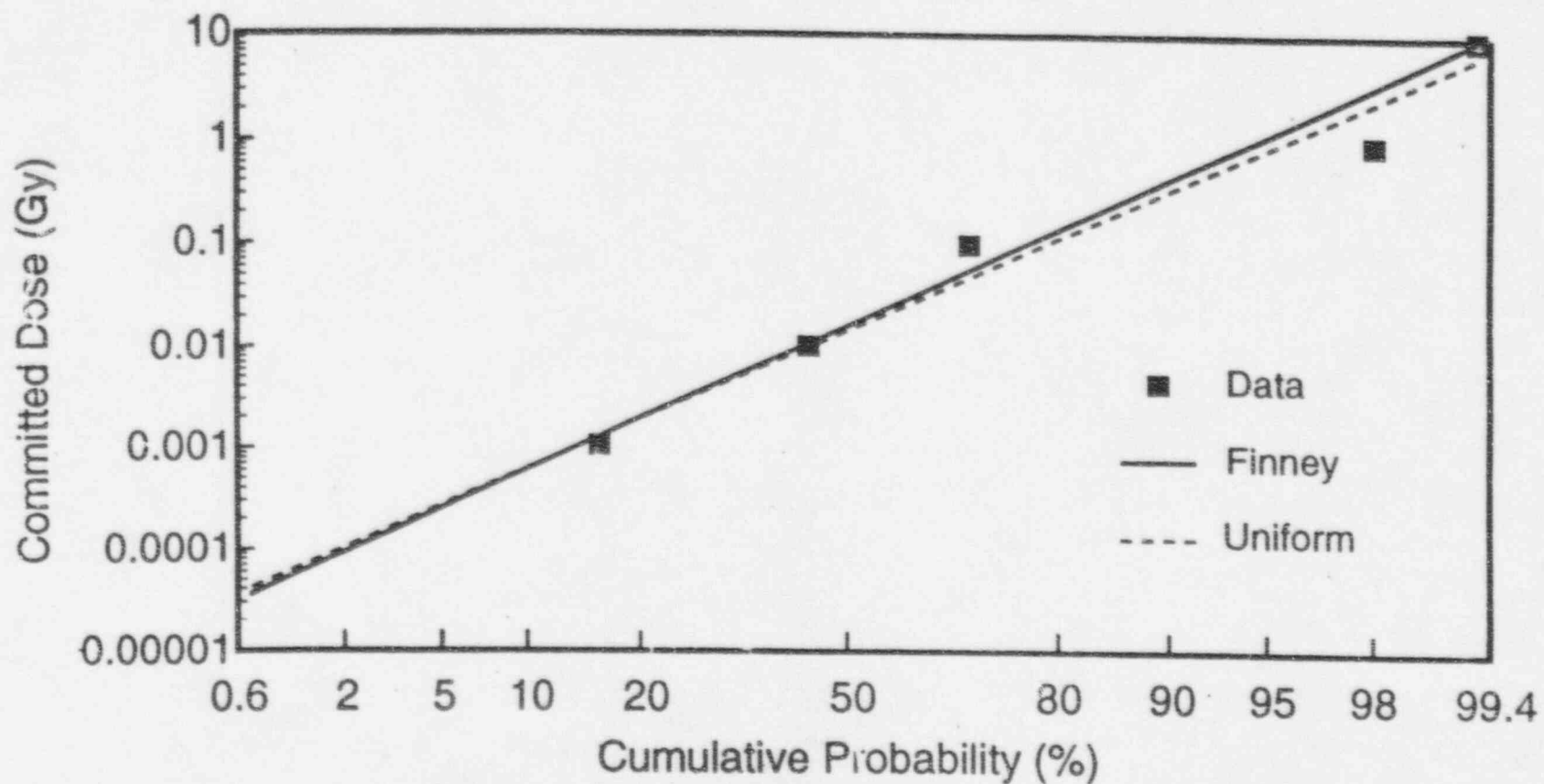


FIGURE 3.3.3. Log Probability Analysis of Committed Dose for 77 Individuals with Internal Contamination from the 1987 Goiânia ^{137}Cs Accident. Regression lines are uniform- or Finney-weighted. (IAEA, 1988a, from Figure 14)

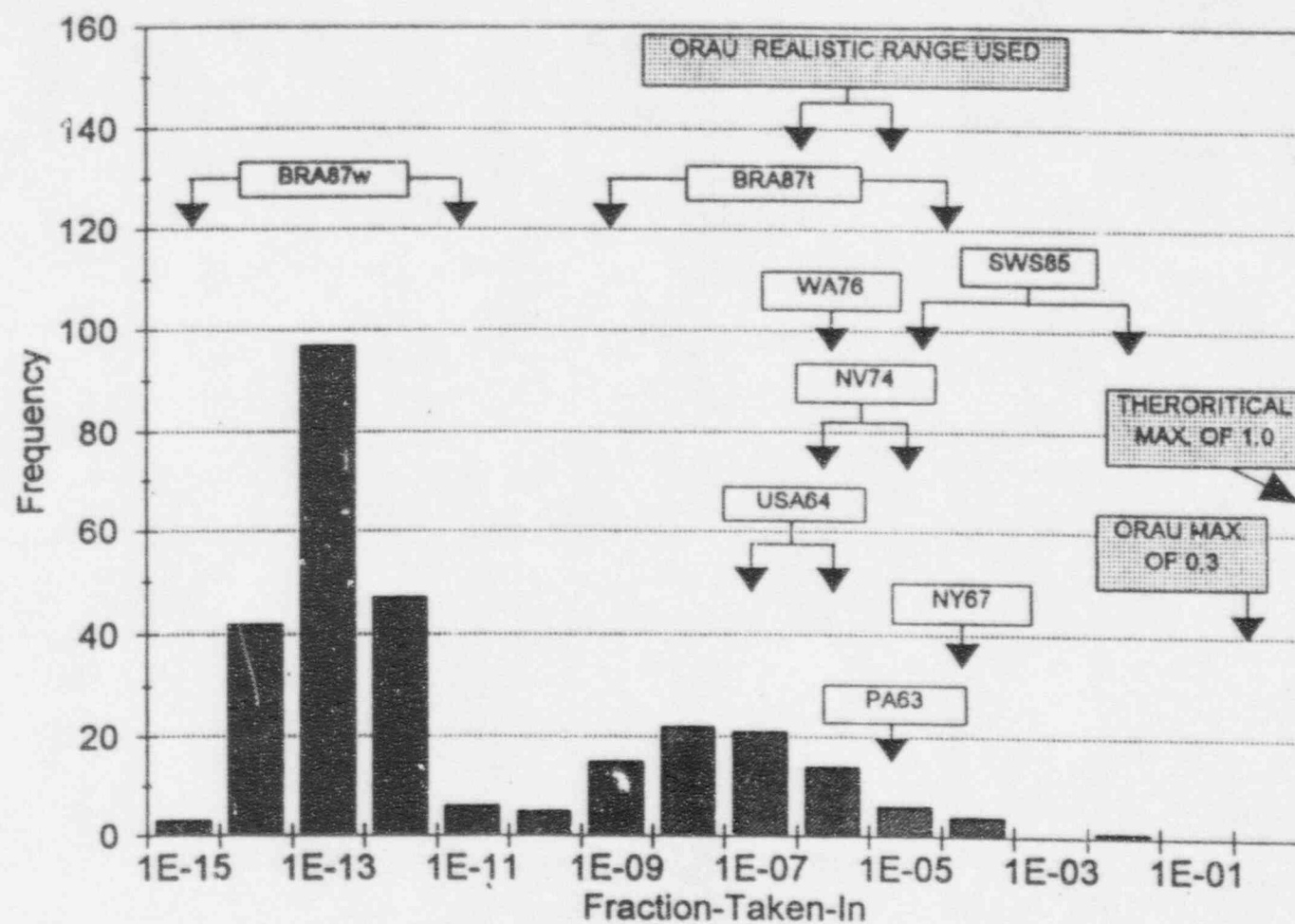


FIGURE 3.3.4. Frequency of Fraction-Taken-In For 60 Radiation Accidents

4.0 CONCLUSIONS

Work should be done in concert with NMSS staff to develop a risk framework that meets the NRC's regulatory decision making needs. Such a framework is proposed in this report, but should be fine-tuned to meet the staff's requirements.

Such a framework should include the probabilities of accidents occurring. The probabilities should be expressed on a per-source, per-year basis, and include summation over accident types and multiplication by the number of sources in use. Potential harm from accidents should be assessed using distributions of coefficients, not point estimates, derived from historical accidents, including fractions of source activity taken in, and probable external doses (both local and whole-body) based on analyses such as those presented above.

Time-and-proximity factors for 231 individuals involved in 40 accidents reviewed by PNL (excluding the large population accidents in Goiânia Brazil and Juarez Mexico) averaged 30 hours at a meter, with a standard deviation of 100 hours at a meter, a geometric mean of 0.37 hours at a meter with a geometric standard deviation of 43. Collective time-and-proximity factors were 12, 113, and 31 for the Goiânia, Indiana PA, and Korea-USA accidents, respectively.

Fractions-taken-in were found to be in the range of 2×10^{-8} to 2×10^{-4} for many accidents reviewed by PNL, many involving unsealed sources. Cleanup workers in Goiânia had fractions-taken-in in the range of 2×10^{-15} to 7×10^{-11} , while townspeople, including those directly involved in the incident, had fractions-taken-in averaging 6×10^{-6} . Roughly 98% of individuals in the 60 accidents surveyed had fractions-taken-in of zero.

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6.0 TASK 6 APPENDIX

Table Task-6-A1. Individual dose equivalent values for the Indiana, PA accident.

at Indiana Regional Cancer Center Indiana, Pennsylvania, on November 16, 1993, NUREG-1480 1.37E11 Bq (3.7 Ci) Ir-192						Gamma (Sv/h [m ² /Bq])	Gamma*Ac t (Sv/h @ 1m)
		mSv	mSv	mSv	Sv	F_p	
		Low	High	Avg.	Avg. (h @ 1 m)		
1 Table 6.3	Physician A	1.82	7.21	4.515	0.004515	0.206127	
2 Table 6.3	TRTT-A			8.2	0.0082	0.374361	
3 Table 6.3	RTT-B			1.1	0.0011	0.050219	
4 Table 6.3	RTR			1.4	0.0014	0.063915	
5 Table 6.3	Medical Physicist A			1.2	0.0012	0.054785	
6 Table 6.3	Nurse A			6.3	0.0063	0.287619	
7 Table 6.4	Patient C	0.09	0.1	0.095	0.000095	0.004337	
8 Table 6.4	Patient D	0.09	0.1	0.095	0.000095	0.004337	
9 Table 6.4	Patient E	0.09	0.1	0.095	0.000095	0.004337	
10 Table 6.4	Patient F	0.09	0.1	0.095	0.000095	0.004337	
11 Table 6.4	Patient G	0.09	0.1	0.095	0.000095	0.004337	
12 Table 6.4	Patient H	0.09	0.1	0.095	0.000095	0.004337	
13 Table 6.4	Patient I	0.09	0.1	0.095	0.000095	0.004337	
14 Table 6.4	Patient J	0.17	0.99	0.58	0.00058	0.026479	
15 Table 6.4	Phlebotomist	3.5	13.9	8.7	0.0087	0.397188	
16 Table 6.4	Office Manager	0.04	0.09	0.065	0.000065	0.002967	
17 Table 6.4	Medical Secretary	0.04	0.09	0.065	0.000065	0.002967	
18 Table 6.4	Tumor Registrar	0.04	0.09	0.065	0.000065	0.002967	
19 Table 6.5	Patient K	2.4	4.3	3.35	0.00335	0.15294	
20 Table 6.5	Patient L	2.4	4.3	3.35	0.00335	0.15294	
21 Table 6.5	Administrative Aide A	0.09	0.58	0.335	0.000335	0.015294	
22 Table 6.5	Administrative Aide B	0.09	0.58	0.335	0.000335	0.015294	
23 Table 6.5	Laboratory Employee	0.08	0.23	0.155	0.000155	0.007076	
24 Table 6.6	Ambulance Driver	4.8	8.4	6.6	0.0066	0.301315	
25 Table 6.6	Ambulance Aide	13.3	25.7	19.5	0.0195	0.890248	
26 Table 6.8	RN A	25	42	33.5	0.0335	1.529401	
27 Table 6.8	RN B	89	137	113	0.113	5.158875	
28 Table 6.8	RN C	38	63	50.5	0.0505	2.305515	
29 Table 6.8	RN D	3.6	5.5	4.55	0.00455	0.207725	
30 Table 6.8	GPN A	88	136	112	0.112	5.113221	
31 Table 6.8	LPN A	56	87	71.5	0.0715	3.264244	
32 Table 6.8	LPN B	112	174	143	0.143	6.528488	
33 Table 6.8	LPN C	56	87	71.5	0.0715	3.264244	
34 Table 6.8	LPN D	28	43	35.5	0.0355	1.620709	
35 Table 6.8	CNA A	103	148	125.5	0.1255	5.729547	
36 Table 6.8	CNA B	52	74	63	0.063	2.876187	
37 Table 6.8	CNA C	155	223	189	0.189	8.628561	
38 Table 6.8	CNA D	52	74	63	0.063	2.876187	
39 Table 6.8	CNA E	103	148	125.5	0.1255	5.729547	
40 Table 6.8	CNA F	52	74	63	0.063	2.876187	
41 Table 6.8	CNA G	52	74	63	0.063	2.876187	
42 Table 6.8	CNA H	3.6	5.5	4.55	0.00455	0.207725	
43 Table 6.8	CNA I	4.3	6.2	5.25	0.00525	0.239682	
44 Table 6.8	Maintenance Man A	19	38	28.5	0.0285	1.301132	
45 Table 6.8	Dietician	3.6	6.3	4.95	0.00495	0.225986	

Table Task-6-A1, continued. Individual dose equivalent values for the Indiana, PA accident.

46 Table 6.8	Activities Director	4.8	22	13.4	0.0134	0.61176
47 Table 6.10	Relative A	54.4	166	110.2	0.1102	5.031045
48 Table 6.10	Relative B	29.2	42	35.6	0.0356	1.625274
49 Table 6.10	Relative C	36.5	64.5	50.5	0.0505	2.305515
50 Table 6.10	Relative D	23.5	36.5	30	0.03	1.369613
51 Table 6.10	Relative E	21.9	31.5	26.7	0.0267	1.218955
52 Table 6.10	Relative F	21.9	31.5	26.7	0.0267	1.218955
53 Table 6.10	Friend A	23.5	92.8	58.15	0.05815	2.654766
54 Table 6.11	Resident B	37.6	128	82.8	0.0828	3.780131
55 Table 6.11	Resident C	63.9	197	130.45	0.13045	5.955533
56 Table 6.11	Resident D	23.1	84	53.55	0.05355	2.444759
57 Table 6.11	Resident E	11.2	15.4	13.3	0.0133	0.607195
58 Table 6.11	Resident F	27.5	31.7	29.6	0.0296	1.351351
59 Table 6.11	Resident G	18.3	22	20.15	0.02015	0.919923
60 Table 6.11	Resident H	34.2	39.5	36.85	0.03685	1.682341
61 Table 6.11	Resident I	21.6	25.9	23.75	0.02375	1.084277
62 Table 6.11	Resident J	14.4	24.3	19.35	0.01935	0.8834
63 Table 6.11	Resident K	17.3	22.8	20.05	0.02005	0.915358
64 Table 6.11	Resident L	12.2	27.5	19.85	0.01985	0.906227
65 Table 6.11	Resident M	57.3	90.9	74.1	0.0741	3.382944
66 Table 6.11	Resident N	13.8	23.4	18.6	0.0186	0.84916
67 Table 6.13	Driver A			1.7	0.0017	0.077611
68 Table 6.13	Driver B	2.5	5.2	3.85	0.00385	0.175757
69 Table 6.13	Driver C	0.34	0.36	0.35	0.00035	0.015979
70 Table 6.13	Supervisor A	13	51.3	32.15	0.03215	1.467768
71 Table 6.13	Safety Technician A	34.3	89.5	61.9	0.0619	2.825968
72 Table 6.13	Safety Technician B	26.9	68.4	47.65	0.04765	2.175402
73 Table 6.13	Other BFI 1	0.013	0.017	0.0148	1.48E-05	0.000676
74 Table 6.13	Other BFI 2	0.013	0.017	0.0148	1.48E-05	0.000676
75 Table 6.13	Other BFI 3	0.013	0.017	0.0148	1.48E-05	0.000676
76 Table 6.13	Other BFI 4	0.013	0.017	0.0148	1.48E-05	0.000676
77 Table 6.13	Other BFI 5	0.013	0.017	0.0148	1.48E-05	0.000676
78 Table 6.13	Other BFI 6	0.013	0.017	0.0148	1.48E-05	0.000676
79 Table 6.13	Other BFI 7	0.013	0.017	0.0148	1.48E-05	0.000676
80 Table 6.13	Other BFI 8	0.013	0.017	0.0148	1.48E-05	0.000676
81 Table 6.13	Other BFI 9	0.013	0.017	0.0148	1.48E-05	0.000676
82 Table 6.13	Other BFI 10	0.013	0.017	0.0148	1.48E-05	0.000676
83 Table 6.13	Other BFI 11	0.013	0.017	0.0148	1.48E-05	0.000676
84 Table 6.13	Other BFI 12 11/29	0.004	0.007	0.00564	5.64E-06	0.000257
85 Table 6.13	Other BFI 13 11/29	0.004	0.007	0.00564	5.64E-06	0.000257
Collective				2478	2.478	113.139
Average				29.16	0.02916	1.33105
Standard Deviation				40.75	0.041	1.860
Minimum				0.00564	5.64E-06	0.000257
Maximum				189.00	0.189	8.628561
No. Individuals				85	85	85
Geometric Mean				2.91	0.00291	0.13304
Geometric Standard Deviation				25.3	25.3	25.3

Table Task-6-A2. Individual dose equivalents and Time-and-Proximity Factors for the 1990 Korea-USA ¹⁹²Ir Shipment Accident.

Inadvertent Shipment of a Radiographic Source from Korea to Amersham Corporation, Burlington, Massachusetts, NUREG-1405, 1990

Ir-192 Korea: 1/18/90 - 2/11/90

INC S/N 1062 USA: 2/11/90 - 3/8/90

date	74.6 days		Gamma (Sv/h @ 1m)	Gamma*Ac (Sv/h @ 1m)
	days	Act. (Ci)	[m ² /Bq]	
1/18/90	55	4.00	1.60E-13	0.02368
3/8/90	6	2.54		
3/14/90	0	2.40	2.49	

Source of data	Who	mrem	rem	F _p , hours Sv at a meter
Table 5.2	Nova Truck Driver	40	0.04	0.0004 0.016892
Table 5.2	Nova Cargo Unloader 1	330	0.33	0.0033 0.139358
Table 5.2	Nova Cargo Unloader 2	330	0.33	0.0033 0.139358
Table 5.2	Nova Unloading Forklift Operator	50	0.05	0.0005 0.021115
Table 5.2	Nova Unloading Checker	70	0.07	0.0007 0.029561
Table 5.2	Nova Shipping clerk	230	0.23	0.0023 0.097128
Table 5.2	Nova Shipping Supervisor	230	0.23	0.0023 0.097128
Table 5.2	Nova Asst. Shipping Clerk	150	0.15	0.0015 0.063345
Table 5.2	Nova Receiving Clerk	400	0.4	0.004 0.168919
Table 5.2	Nova Loading Forklift Operator	200	0.2	0.002 0.084459
Table 5.2	Nova Cargo Loader 1	470	0.47	0.0047 0.19848
Table 5.2	Nova Cargo Loader 2	470	0.47	0.0047 0.19848
Table 5.2	Nova Loading Checker	220	0.22	0.0022 0.092905
Tables 5.3 & 5.4	Covenant Senior Driver	34850	34.85	0.3485 14.71706
Tables 5.3 & 5.4	Covenant Driver Trainee	27560	27.56	0.2756 11.63851
Table 5.4	Patriot Operator	5600	5.6	0.056 2.364865
Table 5.4	Patriot Warehouseman	1080	1.08	0.0108 0.456081
Table 5.4	USCS Inspector	810	0.81	0.0081 0.342061
Table 5.4	Patriot Truck Driver	550	0.55	0.0055 0.232264
Table 5.5	Amersham Rad Tech A	20	0.02	0.0002 0.008446
Table 5.5	Amersham Rad Safety Specialist	30	0.03	0.0003 0.012669
Table 5.5	Amersham Rad Tech B	20	0.02	0.0002 0.008446
Table 5.5	Amersham Rad Safety Officer	15	0.015	0.00015 0.006334
Table 5.5	Amersham Hot Lab Supervisor	40	0.04	0.0004 0.016892
Collective		73765	73.765	0.73765 31.15
Average		3073.5	3.07	0.0307 1.30
Standard Deviation		8801.9	8.80	0.0880 3.72
Minimum		15	0.015	0.00015 0.00633
Maximum		34850	34.85	0.3485 14.72
No. Individuals		24	24	24 24
Geometric Mean		259.58	0.260	0.00260 0.110
Geometric Standard Deviation		7.87	7.87	7.87 7.87

Table Task-6-A3. Summary information for 40 accidents involving sealed sources and 231 individuals with known or estimated doses.

Code	Year	Nuclide	Source Type	Activity, A (Ci)	Activity, A (Bq)	Gamma (Sv/h (m ² /Bq))	Gamma*Act (Sv/h @ 1m)	Whole Body Time- and- Proximity Factor (hours at a meter)
ALG78	78	Ir-192	IndRad	25 9.25E+11	1.60E-13	0.148	87.837838	
ALG78	78	Ir-192	IndRad	25 9.25E+11	1.60E-13	0.148	89.527027	
ALG78	78	Ir-192	IndRad	25 9.25E+11	1.60E-13	0.148	81.08108	
ALG78	78	Ir-192	IndRad	25 9.25E+11	1.60E-13	0.148	74.324324	
ALG78	78	Ir-192	IndRad	25 9.25E+11	1.60E-13	0.148	270.27027	
ALG78	78	Ir-192	IndRad	25 9.25E+11	1.60E-13	0.148	6.7567568	
ALG78	78	Ir-192	IndRad	25 9.25E+11	1.60E-13	0.148	6.7567568	
AUS70	70	Ir-192	IndRad	22 8.14E+11	1.6E-13	0.13024	1.23	
AUS70	70	Ir-192	IndRad	22 8.14E+11	1.6E-13	0.13024	3.53	
BAN85	85	Ir-192	IndRad	50 1.85E+12	1.6E-13	0.296	8.45	
CA79	79	Ir-192	IndRad	28 1.04E+12	1.6E-13	0.16576	0.0120656	
CA79	79	Ir-192	IndRad	28 1.04E+12	1.6E-13	0.16576	0.0241313	
CA79	79	Ir-192	IndRad	28 1.04E+12	1.6E-13	0.16576	0.0422297	
CA79	79	Ir-192	IndRad	28 1.04E+12	1.6E-13	0.16576	0.0844595	
CA79	79	Ir-192	IndRad	28 1.04E+12	1.6E-13	0.16576	0.1025579	
CA79	79	Ir-192	IndRad	28 1.04E+12	1.6E-13	0.16576	0.11764	
CA79	79	Ir-192	IndRad	28 1.04E+12	1.6E-13	0.16576	0.1870174	
CA79	79	Ir-192	IndRad	28 1.04E+12	1.6E-13	0.16576	0.3619691	
CA79	79	Ir-192	IndRad	28 1.04E+12	1.6E-13	0.16576	0.3921332	
CA79	79	Ir-192	IndRad	28 1.04E+12	1.6E-13	0.16576	1.3875483	
CA79	79	Ir-192	IndRad	28 1.04E+12	1.6E-13	0.16576	5.2787162	
CZE66	66	I-131	Medical	2.25 8.33E+10	7.63E-14	0.006351975	0.002	
CZE66	66	I-131	Medical	2.25 8.33E+10	7.63E-14	0.006351975	0.004	
CZE66	66	I-131	Medical	2.25 8.33E+10	7.63E-14	0.006351975	0.006	
CZE66	66	I-131	Medical	2.25 8.33E+10	7.63E-14	0.006351975	0.008	
CZE66	66	I-131	Medical	2.25 8.33E+10	7.63E-14	0.006351975	0.01	
CZE66	66	I-131	Medical	2.25 8.33E+10	7.63E-14	0.006351975	0.012	
CZE66	66	I-131	Medical	2.25 8.33E+10	7.63E-14	0.006351975	0.014	
CZE66	66	I-131	Medical	2.25 8.33E+10	7.63E-14	0.006351975	0.016	
CZE66	66	I-131	Medical	2.25 8.33E+10	7.63E-14	0.006351975	0.018	
CZE66	66	I-131	Medical	2.25 8.33E+10	7.63E-14	0.006351975	0.02	
CZE66	66	I-131	Medical	2.25 8.33E+10	7.63E-14	0.006351975	0.022	
CZE66	66	I-131	Medical	2.25 8.33E+10	7.63E-14	0.006351975	0.024	
CZE66	66	I-131	Medical	2.25 8.33E+10	7.63E-14	0.006351975	0.026	
CZE66	66	I-131	Medical	2.25 8.33E+10	7.63E-14	0.006351975	0.028	
CZE66	66	I-131	Medical	2.25 8.33E+10	7.63E-14	0.006351975	0.03	
CZE66	66	I-131	Medical	2.25 8.33E+10	7.63E-14	0.006351975	0.032	
CZE73	73	Co-60	Tele	2973 1.10E+14	3.7E-13	40.7	0.0025	
CZE73	73	Co-60	Tele	2973 1.10E+14	3.7E-13	40.7	0.0344	
FRG68	68	Ir-192	IndRad	7.8 2.89E+11	1.6E-13	0.046176	32.4	
FRG72	72	Ir-192	IndRad	29.73 1.10E+12	1.6E-13	0.176	1.7	
FRG81	81	Co-60	Tele	2595 9.60E+13	3.7E-13	35.52	0.0113	
FRG81	81	Co-60	Tele	2595 9.60E+13	3.7E-13	35.52	0.0056	
IND68	68	Ir-192	IndRad	1.4 5.18E+10	1.6E-13	0.008268	156.85	
ISR90	90	Co-60	Steril	340541 1.26E+16	3.7E-13	4662	0.0032	

Table Task-6-A3 continued. Summary information for 40 accidents involving sealed sources and 231 individuals with known or estimated doses.

Code	Year	Nuclide	Source Type	Activity, A (Ci)	Activity, A (Bq)	Gamma (Sv/h [m ² /Bq])	Gamma*Act. (Sv/h @ 1m)	Whole Body Time-and-Proximity Factor (hours at 1 meter)
ITA75	75	Co-60	Steril	36000	1.33E+15	3.7E-13	492.84	0.0284
JOH59	59	Co-60	IndRad	1.75	6.48E+10	3.7E-13	0.0239575	1.04
JPN71	71	Ir-192	IndRad	5.26	1.95E+11	1.6E-13	0.0311392	3.2113863
JPN71	71	Ir-192	IndRad	5.26	1.95E+11	1.6E-13	0.0311392	4.1748022
JPN71	71	Ir-192	IndRad	5.26	1.95E+11	1.6E-13	0.0311392	4.8170794
JPN71	71	Ir-192	IndRad	5.26	1.95E+11	1.6E-13	0.0311392	8.0284657
JPN71	71	Ir-192	IndRad	5.26	1.95E+11	1.6E-13	0.0311392	16.056931
JPN71	71	Ir-192	IndRad	5.26	1.95E+11	1.6E-13	0.0311392	42.711438
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0063345
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0084459
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0084459
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0126689
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0168919
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0168919
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0211149
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0295608
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0633446
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0844595
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0929054
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0971284
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.0971284
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.1393581
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.1393581
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.1689189
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.1984797
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.1984797
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.2322635
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.3420808
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	0.4560811
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	2.3648649
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	11.638514
KOR90	90	Ir-192	IndRad	4	1.48E+11	1.60E-13	0.02368	14.717061
KY76	76	Ir-192	IndRad	78	2.89E+12	1.60E-13	0.46178	2.06
LA78	78	Ir-192	IndRad	100	3.70E+12	1.60E-13	0.592	0.0845
MEX82	82	Co-60	IndRad	5	1.85E+11	3.70E-13	0.06845	175.31045
MEX82	82	Co-60	IndRad	5	1.85E+11	3.70E-13	0.06845	886.63258
MEX82	82	Co-60	IndRad	5	1.85E+11	3.70E-13	0.06845	511.32213
MEX82	82	Co-60	IndRad	5	1.85E+11	3.70E-13	0.06845	419.28415
MEX82	82	Co-60	IndRad	5	1.85E+11	3.70E-13	0.06845	438.27611
MOR84	84	Ir-192	IndRad	16.2	6.00E+11	1.60E-13	0.096	10.416667
MOR84	84	Ir-192	IndRad	16.2	6.00E+11	1.60E-13	0.096	26.041667
MOR84	84	Ir-192	IndRad	16.2	6.00E+11	1.60E-13	0.096	67.706333
MOR84	84	Ir-192	IndRad	16.2	6.00E+11	1.60E-13	0.096	83.88
MOR84	84	Ir-192	IndRad	16.2	6.00E+11	1.60E-13	0.096	106.5138
MOR84	84	Ir-192	IndRad	16.2	6.00E+11	1.60E-13	0.096	123.3923
MOR84	84	Ir-192	IndRad	16.2	6.00E+11	1.60E-13	0.096	139.457
MOR84	84	Ir-192	IndRad	16.2	6.00E+11	1.60E-13	0.096	156.6363

Table Task-6-A3 continued. Summary information for 40 accidents involving sealed sources and 231 individuals with known or estimated doses.

[illegible]

Table Task-6-A3 continued. Summary information for 40 accidents involving sealed sources and 231 individuals with known or estimated doses.

Code	Year	Nuclide	Source Type	Activity, A (Ci)	Activity, A (Bq)	Gamma (Sv/h (m ² /Bq))	Gamma*Act. (Sv/h @ 1m)	Whole Body Time-and-Proximity Factor (hours at a meter)
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.0639153
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.0776114
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.1529401
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.1529401
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.175767
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.2061267
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.2077246
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.2077246
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.2259861
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.2396822
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.2876187
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.3013148
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.3743608
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.3971877
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.607195
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.6117604
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.84916
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.8834003
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.8902484
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.9062272
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.9153579
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	0.9199233
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.0642768
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.2189554
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.2189554
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.3011322
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.3513514
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.3696129
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.4677684
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.529401
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.6207085
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.6252739
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	1.6823411
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	2.1754018
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	2.305515
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	2.305515
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	2.4447589
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	2.6547863
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	2.8259679
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	2.876187
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	2.876187
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	2.876187
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	3.264244
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	3.264244
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	3.3829438
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	3.7801315

Table Task-6-A3 continued. Summary information for 40 accidents involving sealed sources and 231 individuals with known or estimated doses.

Code	Year	Nuclide	Source Type	Activity, A (Ci)	Activity, A (Bq)	Gamma (Sv/h (m ² /Bq))	Gamma*Act (Sv/h @ 1m)	Whole Body Time- and-Proximity Factor (hours at 1 meter)
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	5.0310446
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	5.1132213
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	5.1568751
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	5.7295471
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	5.7295471
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	5.9555332
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	6.5284879
PA92	92	Ir-192	Brachy	3.7	1.37E+11	1.60E-13	0.021904	8.628561
PRC63	63	Co-60	IndRad	10	3.70E+11	3.70E-13	0.1369	14.609204
PRC63	63	Co-60	IndRad	10	3.70E+11	3.70E-13	0.1369	29.218408
PRC63	63	Co-60	IndRad	10	3.70E+11	3.70E-13	0.1369	43.827611
PRC63	63	Co-60	IndRad	10	3.70E+11	3.70E-13	0.1369	58.436815
PRC63	63	Co-60	IndRad	10	3.70E+11	3.70E-13	0.1369	292.18408
PRC63	63	Co-60	IndRad	10	3.70E+11	3.70E-13	0.1369	584.36815
PRC80	80	Co-60	Steril	53000	1.96E+15	3.70E-13	725.57	1.89
PRC85	85	Cs-137	IndRad?	10	3.70E+11	9.25E-14	0.034225	3.35
PRC86	86	Co-60	(1)	6888	2.55E+14	3.70E-13	94.29672	3.83
PRC87	87	Co-60	Steril	89000	3.29E+15	3.70E-13	1218.41	7.3
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	8.143138
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	8.677649
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	9.118739
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	9.521521
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	9.912457
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	10.31026
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	10.73358
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	11.20769
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	11.77738
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	12.55044
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	14
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	60.87168
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	91.30752
PRC92	92	Co-60	Expt	12	4.44E+11	3.70E-13	0.16428	121.7434
SAF77	77	Ir-192	IndRad	6.76	2.50E+11	1.60E-13	0.04	2.5
SAF77	77	Ir-192	IndRad	6.76	2.50E+11	1.60E-13	0.04	4.25
SAF77	77	Ir-192	IndRad	6.76	2.50E+11	1.60E-13	0.04	29
SAL89	89	Co-60	Steril	18000	6.66E+14	3.70E-13	246.42	0.013
SAL89	89	Co-60	Steril	18000	6.66E+14	3.70E-13	246.42	0.018
SAL89	89	Co-60	Steril	18000	6.66E+14	3.70E-13	246.42	0.034
SCO69	69	Ir-192	IndRad	25	9.25E+11	1.60E-13	0.148	4.05
TN71	71	Co-60	Expt	7700	2.85E+14	3.70E-13	105.413	0.0247
TRK76	76	Co-60	Tele	2260	8.36E+13	3.70E-13	30.9394	0.0001118
TX72	72	Cs-137	IndRad	4	1.48E+11	9.25E-14	0.01369	730
UK77	77	Ir-192	IndRad	21.6	8.00E+11	1.60E-13	0.128	0.78
UK81	81	Cs-137	Brachy	0.12	4.44E+09	9.25E-14	0.0004107	0.487
WA76	76	Am-241	Defuse	343	1.27E+13	7.57E-15	0.096165142	0.052
WI61	61	Co-60	Expt	200	7.40E+12	3.70E-13	2.738	0.913

Table Task-6-A3 continued. Summary information for 40 accidents involving sealed sources and 231 individuals with known or estimated doses.

Code	Year	Nuclide	Source Type	Activity, A (Ci)	Activity, A (Bq)	Gamma		Whole Body Time- and- Proximity Factor	
						(Sv/h [m^2/Bq])	Gamma*Act (Sv/h @ 1m)		
								Minimum	0.0001118
								Maximum	730
								Average	30
								Std Dev	100
								GeoMean	0.37
								GSD	42.53
								Median	0.46
								Mode	6.76E-04
								Number	231

APPENDIX G: PRELIMINARY RISK ANALYSIS FOR SELECTED SOURCES

A full implementation of the risk analysis described in Section 2 of the Task 7 Final Report is beyond the scope of work of the current project. However, a sample risk analysis is given below.

G1.0 RADIOLOGICAL PROPERTIES OF RADIONUCLIDES REGISTERED BETWEEN 1987 AND 1992

Data provided to PNL by S. L. Baggett (Baggett 1993) included 24 valid radionuclide names. Table G-1 lists those radionuclides as row labels in column 1, in order of increasing atomic number. The specific dose equivalent rate constant¹ for each nuclide (or chain, in the case of ²²⁶Ra) is given in the next column in units of rems per hour at 1 m from 1 Ci (Unger and Trubey, 1981). The next two columns list the most restrictive ALI values for inhalation and ingestion, respectively. There is no ALI for ⁸⁵Kr. The half-life for each nuclide is given in the center column.

Since distributions of source strengths and numbers of sources in service *with activities below 20 millicuries (the limit of the ORAU report)* are not available, a source strength of 20 mCi was arbitrarily chosen as a reference value in Table G-1. Also, for simplicity, the "worst plausible case" reference value for the time-and-proximity factors, F_p , of 1000 hours at one meter was chosen (730 was the highest observed) for use in Table G-1. Similarly, a "worst plausible case" reference value of the fraction-taken-in, F_i , of 10^{-4} was chosen for use in Table G-1.

Column 7 shows the dose equivalent in rems for external exposure for 1000 hours at 1 meter away from an unshielded 20 mCi source. Columns 8 and 9 show the committed effective dose equivalent in rems for intakes of 10^{-4} of a 20 mCi source, i.e., 2 μ Ci, for inhalation and ingestion, respectively. Note that these columns can be interpreted as doses in millirems for 1 hour at a meter and 10^{-7} fraction-taken-in, both more central values from the PNL analysis presented in Appendix F, Task 6.

The sixth column of Table G-1 gives the ratio of the value in column 7 divided by the value in column 9. The column 6 values are independent of source strength. Column 6 values are a ratio of external exposure hazard (dose) to intake hazard (dose) under the 1000 hours at a meter and 10^{-4} scenarios. Figure G-1 is a bar plot of Column 6 values. For ⁸⁵Kr, there is no ALI even though it poses an external hazard; for ⁶³Ni, ⁵⁵Fe, ³H, ²¹⁰Bi, ¹⁰⁶Ru, ¹⁴C, and

¹Although it would be desirable to use effective dose equivalent rate constants, none have been published in the peer-reviewed literature. Calculations of such constants, based on the methods of the ICRP and ICRU, as shown in Appendix A of the Task 7 Final Report, lead to unrealistically high numbers for low-energy photon emitters if "bare" sources are assumed (e.g., ²⁴¹Am). Further research is needed in this area. Thus, the work of Unger and Trubey (1981), based on ANSI/ANS-6.1.1-1977, has been used for dose equivalent rate constants.

Table G-1. Summary of radiological data needed for risk analysis for radionuclides listed in the NMSS General License Database System from 1987 through 1992.

Nuclide	rem/h at 1 m from 1 Ci	Most Restrictive Inhalation ALI (uCi)	Most Restrictive Ingestion ALI (uCi)	Half-Life (years)	Ingestion of 1u ⁻⁴ of Source	Dose from 1000 h at 1 m / Dose hours at 1 m from 20 mCi of 10 ⁻⁴ of Source (rems)	Dose from 1000 hours at 1 m from 20 mCi of 10 ⁻⁴ of Source (rems)	Dose from Inhalation of 10 ⁻⁴ of 20 mCi of Source (rems)	Dose from Ingestion of 10 ⁻⁴ of 20 mCi of Source (rems)
H-3	0	3000	3000	12.4	0	-	0.0033	0.0033	
C-14	0	90	90	5,730	0	-	0.11	0.11	
Sc-46	1.17	200	900	0.230	2101	23	0.05	0.011	
Ti-44	0.145	6	300	47.3	86.8	2.9	1.7	0.033	
Fe-55	0	2000	9000	2.70	0	-	0.0050	0.0011	
Co-60	1.37	200	500	5.27	1370	27	0.05	0.020	
Ni-63	0	2000	9000	96.0	0	-	0.0050	0.0011	
Kr-85	0.00157	-	-	10.7	-	0.031	-	-	
Sr-90	0	4	30	29.1	0	-	2.5	0.33	
Ru-106	0	10	200	1.01	0	-	1.0	0.050	
Cd-109	0.184	40	300	1.30	111	3.7	0.25	0.033	
I-129	0.126	9	5	1.57E+7	1.26	2.5	1.1	2.0	
Cs-137	0.382	200	100	30.0	76.4	7.6	0.05	0.10	
Ba-133	0.455	700	2000	10.7	1822	9.1	0.014	0.0050	
Pm-147	2.676E-06	100	4000	2.62	0.0214	5.35E-05	0.10	0.0025	
Eu-152	0.744	20	800	13.3	1191	15	0.50	0.013	
Ti-204	0.00112	2000	2000	3.78	4.46	0.022	0.0050	0.0050	
Po-210	5.269E-06	0.6	3	0.379	3.16E-5	1.05E-04	17	3.3	
Bi-210	0	200	800	0.0137	0	-	0.05	0.013	
Ra-226	1.43	0.6	2	1,600	5.71	29	17	5.0	
Pu-238	0.0790	0.007	0.9	87.7	0.142	1.6	1,429	11	
Am-241	0.314	0.006	0.8	432	0.502	6.3	1,667	13	
Cm-244	0.0644	0.01	1	18.1	0.129	1.3	1,000	10	
Cf-252	0.0418	0.002	2	2.64	0.167	0.84	5,000	5.0	

⁹⁰Sr, the specific dose equivalent rate constant is 0, even though, in the case of ⁹⁰Sr-⁹⁰Y, there is an external irradiation hazard. For sources with Column 6 ratios greater than 1, external exposure dominates risk; for sources whose values are less than 1, intake dominates risk. Note that these are based on ingestion ALIs; values based on inhalation ALIs, in general, are less than or equal to the Column 6 values.

Figure G-2 shows Column 9 values plotted as a function of Column 7 values, illustrating the very wide range of doses associated with 20 mCi sources under the exposure scenarios described above.

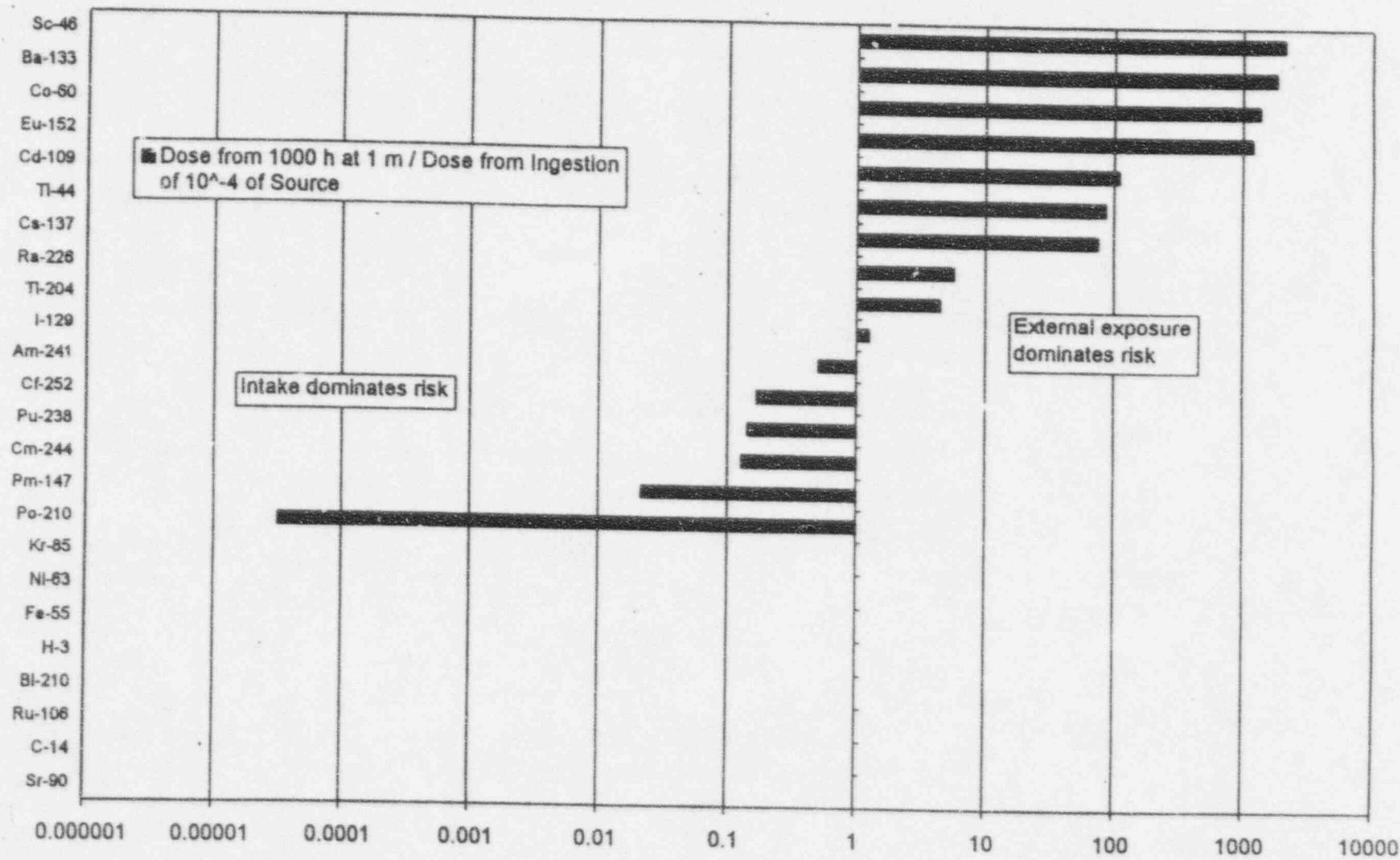


Figure G-1. Ratios of (Dose Equivalent from 1000 h at 1 m) divided by (Committed Effective Dose Equivalent from intake of 10^{-4} of a source). These dimensionless ratios are independent of source strength.

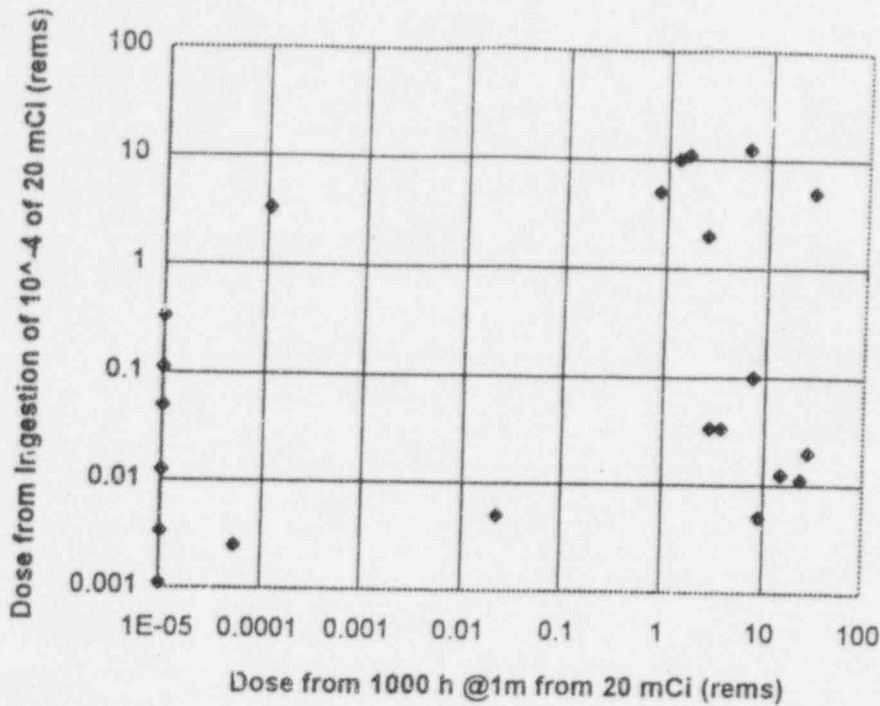


Figure G-2. Committed effective dose equivalent from ingestion of 10^{-4} of a 20 mCi source (vertical axis) as a function dose equivalent from external exposure for 1000 hours at 1 meter from an unshielded 20 mCi source. The relative doses are seen to be widely variable.

G2.0 NUMBER OF DEVICES

Numbers of sources are given in Table Task-3-1B for by Device Code. For this analysis, Device Code D, Gamma Gauges, is chosen for radionuclide ^{137}Cs , which includes 2493 sources with average activity 883 mCi. Table G-2 shows a possible (hypothetical) distribution of numbers of sources and activities to give the correct total and average. It would be best to have exact numbers from a full database.

2493
- 15000
71000
G2

Table G-2. Possible (hypothetical) distribution of numbers of sources and activities for ^{137}Cs gamma gauges for a total of 2493 sources and an average of 883 mCi per source, as shown in Table Task-3-1B.

Activity (mCi)	Number
1	500
10	500
100	559
1000	800
10,000	134
TOTAL	2493

G3.0 RATES OF DEVICE INVOLVEMENT

G3.1 RATES OF INCIDENTS

Table Task-3-8 shows that there were 6 incidents in ORAU Category B (Regulatory Guide 10.10 Device Code D) over a 10 year period, 5 incidents involving ^{137}Cs . Assuming that one out of two incidents is reported, this gives a rate of $(6 \text{ incidents reported}) \times (2 \text{ incidents occurring per incident reported}) \div (10 \text{ years}) = 1.2 \text{ incidents per year}$ for all Device Code D devices. Using the total number of Device Code D sources in Table Task-3-2, 24,679, and correcting for the fraction of Device Code D sources that are ^{137}Cs , one calculates $(1.2 \text{ incidents per year}) \times (2493 / 24,679) = 0.121 \text{ incident per year for } ^{137}\text{Cs Device Code D devices}$.

One could argue from Table Task-3-8 that the rate for cesium sources is predominant, that is, cesium sources account for 5/6 of all incidents. Using this logic would yield a rate of $(1.2 \text{ incidents per year}) \times (5/6) = 1.0 \text{ incident per year for } ^{137}\text{Cs Device Code D devices}$. This analysis uses the latter value.

G3.2 NUMBERS OF DEVICES PER INCIDENT

Table Task-3-7 shows 9 devices were involved in the 6 incidents listed in Table Task-3-6, for a rate of 1.5 device per incident.

G3.3 RATE OF DEVICE INVOLVEMENT IN INCIDENTS

The rate of device involvement in incidents is the product of the number of incidents per year and the number of devices per incident, $(1.0 \text{ incident per year}) \times (1.5 \text{ devices per}$

incident) = 1.5 devices involved in incidents per year.

G4.0 SEVERITY OF INCIDENTS

There are several aspects to severity.

G4.1 FRACTION OF INCIDENTS RESULTING IN LEAKING SOURCES

In Table Task-3-6, 19 out of 114 incidents resulted in leaking sources, although none of these involved Device Code D (ORAU Category B) gamma gauge sources. One could argue that such sources are made to withstand leaking. Alternatively, Table Task-3-7 shows that 75 of 300 devices involved in incidents leaked, primarily static eliminators and ORAU Category L sources. For this value, 0.125 of incidents result in a leaking source.

G4.2 NUMBER OF PERSONS INVOLVED IN INCIDENT

Most incidents involve 1 person, but more may be involved. For intakes, Table 3.1.1 for Task 6 showed the distribution of numbers of persons involved in accidents, as summarized in Table G-3. There are 4364 persons involved in the 60 incidents, using 77 persons (instead of the value of 20 in Table 3.3.1) for Goiânia.

Due to the difficulty of incorporating different numbers of persons with different fractions-taken-in and different time-and-proximity factors, the accompanying analysis considers only 1-person incidents.

Table G-3. The distribution of numbers of persons involved in incidents.

Number of Persons Involved in Incident	Frequency of Occurrence
1	34
2	9
3	3
4	1
5	3
6	2
11	1
16	1
22	1
24	1
28	1
77	1
94	1
4000	1

G4.3 DISTRIBUTION OF FRACTIONS-TAKEN-IN

These data are also taken from Table 3.3.1 of the Task 6 Report. The fractions-taken-in are 4355 values, expanded from Table 3.3.1, with the omission of the Swiss accident of 1985 (which involved an unsealed ^3H source and a, very high fraction-taken-in, namely, 0.02). The average fraction-taken-in was 1.34×10^{-7} , with a standard deviation of 4.17×10^{-6} . The latter two parameters were used to define a lognormal distribution.

G4.4 DISTRIBUTION OF TIME-AND-PROXIMITY FACTORS

These factors were modeled using a lognormal distribution with mean of 30 hours at 1 m and a standard deviation of 100 hours at 1 m, based on the results presented in Table Task-6-A3.

G4.5 DISTRIBUTION OF SOURCE REMOVAL FROM SHIELD

A scenario of the source removal from the shield was assumed to occur in 0.5 of incidents. This is an arbitrary figure that is probably an overestimate.

G5.0 PRELIMINARY PROBABILISTIC RISK ANALYSIS RESULTS

The output of a 1000-trial Crystal Ball simulation is attached at the end of this appendix.

External dose equivalent per year was modeled as $(\text{Incident Rate [devices/year]}) \times (\text{Source Activity}) \times (\text{Probability of Removal from Shield}) \times (\text{Time-and-Proximity Factor}) \times (\text{Specific Dose Equivalent Rate Constant})$, where the factors in italics were sampled from distributions described above. The mean dose was 6.81 rems per year from incidents (recall that many of these sources have activities of 10,000 millicuries, as described in Table G-2) with a maximum of 1056 rems. Had the source activities been limited to 20 mCi, the highest dose would have been 20/883 as large, or 24 rems.

Committed effective dose equivalent (CEDE) from ingestion intakes was modeled as $(\text{Incident Rate [devices/year]}) \times (\text{Source Activity}) \times (\text{Probability of Leakage}) \times (\text{Fraction-Taken-In}) \times (5 \text{ rems CEDE/ALI}) \times (1 \text{ ALI}/0.1 \text{ mCi})$, where the factors in italics were sampled from distributions described above. The mean dose was 0.4 mrem with a maximum of 154 mrem. Had the source activities been limited to 20 mCi, the highest dose would have been 20/883 as large, or 3.5 millirems.

G6.0 DISCUSSION

This analysis is preliminary and was made to demonstrate the proof-of-principle. The methods should be reviewed and refined, and calculations for extremity doses, non-stochastic effects, and collective doses should be made for each nuclide and each Device Code listed in Table Task-3-1. It should be determined whether there is a need to restrict the data to source strengths less than or equal to 20 mCi as was done in the scope of work for the ORAU Report. There is a need for detailed data on numbers of devices by source type (the Regulatory Guide 10.10 Device Code is not adequate), isotope(s), dates placed in service, activities, and design. These data should be available for each of the 500,000 or so sources now in use.

It was discovered that incorporating realistic scenarios (e.g., different fractions-taken-in and different time-and-proximity factors for each individual in multiple-person incidents) is more difficult than expected, the remaining funds did allow us to complete this extra task.

G7.0 ADDITIONAL REFERENCES FOR APPENDIX G

Baggett, S. L. 1993. Letter to D. J. Strom of Pacific Northwest Laboratory dated November 2, 1993. Washington, DC: Sealed Source Safety Section, U.S. Nuclear Regulatory Commission.

Unger, L. M., and D. K. Trubey. 1981. *Specific Gamma-Ray Dose Constants for Nuclides Important to Dosimetry and Radiological Assessment*. ORNL RSIC-45. United States

G8.0 SAMPLE PROBABILISTIC RISK ANALYSIS: CRYSTAL BALL REPORT

Forecast: External Dose

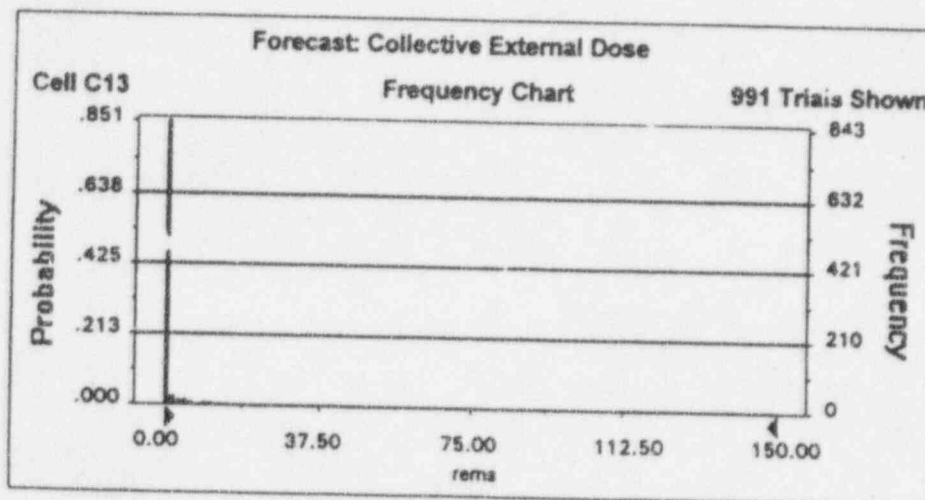
Cell: C13

Summary:

Display Range is from 0.00 to 150.00 rem
 Entire Range is from 0.00 to 1,056.46 rem
 After 1,000 Trials, the Std. Error of the Mean is 1.53

Statistics:

	Value
Trial	1000
Mean	6.81E+00
Median	0.00E+00
Mode	0.00E+00
Standard Deviation	4.84E+01
Variance	2.34E+03
Skewness	1.44E+01
Kurtosis	2.62E+02
Coeff. of Variability	7.10E+00
Range Minimum	0.00E+00
Range Maximum	1.06E+03
Range Width	1.06E+03
Mean Std. Error	1.53E+00



Appendix G

Forecast: Collective External Dose (cont'd)

Cell: C13

Percentiles:

<u>Percentile</u>	<u>rams</u>
0.0%	0.00
2.5%	0.00
5.0%	0.00
50.0%	0.00
95.0%	21.48
97.5%	50.46
100.0%	1,056.46

End of Forecast

Appendix G

Forecast: Intake Dose

Cell: C14

Summary:

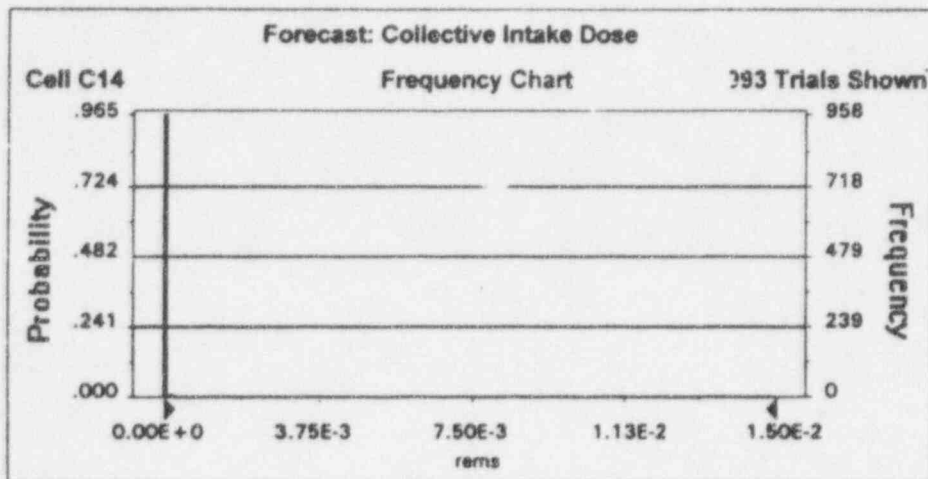
Display Range is from 0.00E+0 to 1.50E-2 rems

Entire Range is from 0.00E+0 to 1.54E-1 rems

After 1,000 Trials, the Std. Error of the Mean is 1.74E-4

Statistics:

	Value
Trials	1000
Mean	3.99E-04
Median	0.00E+00
Mode	0.00E+00
Standard Deviation	5.51E-03
Variance	3.04E-05
Skewness	2.33E+01
Kurtosis	6.22E+02
Coeff. of Variability	1.38E+01
Range Minimum	0.00E+00
Range Maximum	1.54E-01
Range Width	1.54E-01
Mean Std. Error	1.74E-04



Appendix G

Forecast: Collective Intake Dose (cont'd)

Cell: C14

Percentiles:

<u>Percentile</u>	<u>rams</u>
0.0%	0.00E+00
2.5%	0.00E+00
5.0%	0.00E+00
50.0%	0.00E+00
95.0%	4.95E-05
97.5%	6.15E-04
100.0%	1.54E-01

End of Forecast

Appendix G

Assumptions

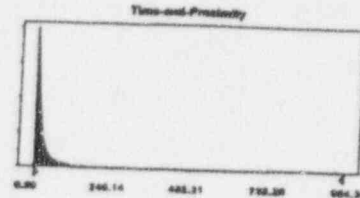
Assumption: Time-and-Proximity

Cell: C9

Lognormal distribution with parameters:

Mean 30.00
Standard Dev. 100.00

Selected range is from 0.00 to +Infinity
Mean value in simulation was 33.57



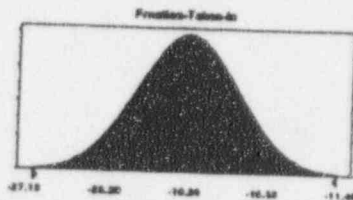
Assumption: Fraction-Taken-In

Cell: C8

Lognormal distribution with parameters:

Mean -19.26 (log space)
Standard Dev. 2.62 (log space)

Selected range is from -Infinity to +Infinity
Mean value in simulation was 0.00



Assumption: Probability of Leakage

Cell: C7

Custom distribution with parameters:

Single point 0.00
Single point 1.00
Total Relative Probability

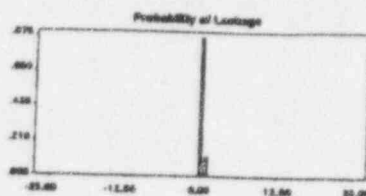
Relative Prob.
0.875000
0.125000
1.000000

Mean value in simulation was 0.14

Appendix G

Assumption: Probability of Leakage (cont'd)

Cell: C7



Assumption: Device Activity

Cell: C6

Custom distribution with parameters:

Single point	1.00
--------------	------

Single point	10.00
--------------	-------

Single point	100.00
--------------	--------

Single point	1,000.00
--------------	----------

Single point	10,000.00
--------------	-----------

Total Relative Probability

Relative Prob.

0.200562

0.200562

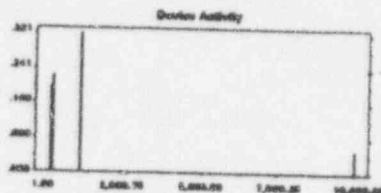
0.224228

0.320899

0.053751

1.000000

Mean value in simulation was 764.28



Assumption: Probability of Removal from Shield

Cell: C11

Custom distribution with parameters:

Single point	0.00
--------------	------

Single point	1.00
--------------	------

Total Relative Probability

Relative Prob.

0.500000

0.500000

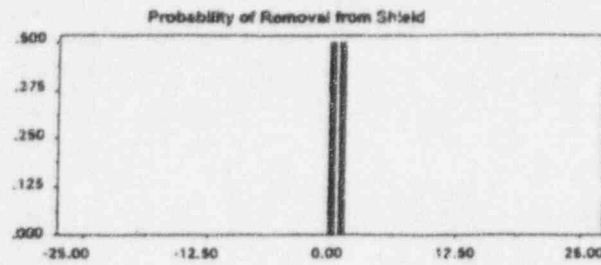
1.000000

Mean value in simulation was 0.47

Appendix G

Assumption: Probability of Removal from Shield (cont'd)

Cell: C11



Assumption: Number of Persons

Cell: C10

Custom distribution with parameters:

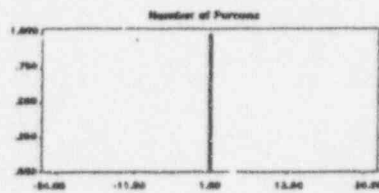
Single point 1.00
Total Relative Probability

Relative Prob.

1.000000

1.000000

Mean value in simulation was 1.00



End of Assumptions