NUREG/CR-3467
UCRL-53485

Environmental Assessment of the Use of Radionuclides as Tracers in the Enhanced Recovery of Oil and Gas

Prepared by Y. C. Ng, R. T. Cederwall, L. R. Anspaugh

Lawrence Livermore National Laboratory

Prepared for
U.S. Nuclear Regulatory
Commission

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N W
   Washington, DC 20555
2. The NRC/GPO Sales Program, U S Nuclear Regulatory Commission,
   Washington DC 20555
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda, NRC Office of Inspection and Enforcement bulletins circulars information notices, inspection and investigation notices, Licensee Event Reports vendor reports and correspondence, Commission papers; and applicant and licensee documents and correspondence

The following documents in the NUREG series are available for purchase from the NRC/GPO Sales Program formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations* and *Nuclear Regulatory Commission Issuances*

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission forerunner agency to the Nuclear Regulatory Commission

Documents available from public and special technical libraries include all open literature items, such as books journal and periodical articles, and transactions *Federal Register* notices, federal and state legislation and congressional reports can usually be obtained from these libraries

Documents such as theses dissertations foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited

Single copies of NRC draft reports are available free upon written request to the Division of Technical Information and Document Control U S Nuclear Regulatory Commission, Washington DC 20555

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library 7920 Norfolk Avenue Bethesda, Maryland, and are available there for reference use by the public Codes and standards are usually copyrighted and may be purchased from the originating organization or if they are American National Standards, from the American National Standards Institute 1430 Broadway New York NY 10018

NUREG/CR-3467
UCRL-53485

Environmental Assessment of the Use of Radionuclides as Tracers in the Enhanced Recovery of Oil and Gas

Manuscript Completed: September 1983
Date Published: November 1983

Prepared by
Y. C. Ng, R. T. Cederwall, L. R. Anspaugh

Lawrence Livermore National Laboratory
Livermore, CA 94550

Prepared for
Division of Risk Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NRC FIN A0388

ABSTRACT

An environmental assessment of the use of radioisotopes as interwell tracers in field flooding for the enhanced recovery of oil and natural gas was performed. A typical operation using radioisotopes for interwell tracing was analyzed from the standpoint of three stages of operation: aboveground, subsurface, and recovery and disposal. Doses to workers who handle radioactive tracers and to members of the public were estimated for normal and accidental exposure scenarios. On the basis of estimates of the total quantity of tracer radionuclides injected in a year, the annual number of projects, the average number of injections per project, and assumed values of accident frequency, the collective dose equivalent is estimated to be 1.1 man-rem/y to workers and 15 man-rem/y to members of the public. The national radiological impact of the use of radioisotopes as interwell tracers in EOR projects is estimated to be a total collective dose equivalent of <16 man-rem/y. Accidental exposures are estimated to contribute relatively little to the total.

CONTENTS

Preceding page blank

LIST OF FIGURES

LIST OF TABLES

ACKNOWLEDGMENTS

The authors greatly appreciate the assistance of the many individuals who contributed materially to this study. They are especially grateful to D. Schutz and A. Carmichael, Teledyne Isotopes, Westwood, N.J.; B. J. Williams, Conoco, Inc., Ponca City, Okla.; and B. Zemel, Shell Development Co., Houston, Tex. for their cooperation throughout this project. They provided the wealth of technical information and data on tracer applications in petroleum technology that was essential for this study. The authors are also pleased to thank D. Eagleson of Teledyne Isotopes and J. Netherland of Petroleum Tracers, Inc., Dallas, Tex. for discussions about field operations and radiation protection procedures.

They profited from discussions with A. G. Collins, J. E. Chism, M. Ray, and R. Haemstra of the U.S. Department of Energy's Bartlesville Energy Technology Center, Bartlesville, Okla.; E. D. Bailey, Bureau of Radiation Control, Texas Department of Health, Austin, Tex.; and K. Wong, Radiation Health Branch, California Department of Health Services, Panorama City, Calif.. They are pleased to thank R. Ginn, Railroad Commission of Texas, Austin, Tex. for helpful discussions on underground injection control; and J. Shinn and K. Peterson, Lawrence Livermore National Laboratory (LLNL), for guidance pertaining to atmospheric transport calculations. They appreciate the comments and suggestions pertaining to hydrologic transport from R. B. Codell, U.S. Nuclear Regulatory Commission (NRC), Washington, D.C.; and T. Harvey, L. Edwards, and D. Layton, LLNL.

Comments, suggestions, and constructive criticism by members of the Research Review Group (RRG) and others are gratefully acknowledged. RRG members include R. Wascom, Office of Environmental Affairs, Baton Rouge, La.; E. Bailey, Texas Department of Health, Austin, Tex.; F. Hoch, Department of Health and Rehabilitative Services, Tallahassee, Fla.; J. M. Brown, U.S. NRC; and J. F. Kendig, U.S. NRC. D. Solberg of the U.S. NRC also provided useful comments and suggestions.

The authors are pleased to acknowledge the contributions of colleagues at LLNL including W. Martin for programming support, Y. Ricker for technical assistance and reviewing the manuscript, B. Fox and A. Fountain for word processing, and T. Nelson for technical editing. We deeply appreciate the helpful comments, suggestions and encouragement throughout this project from A. Tse, the technical monitor for this NRC-sponsored project.

ENVIRONMENTAL ASSESSMENT OF THE USE OF RADIONUCLIDES AS TRACERS IN THE ENHANCED RECOVERY OF OIL AND GAS

EXECUTIVE SUMMARY

OBJECTIVE

An environmental assessment was performed to evaluate the radiological impacts of using radioisotopes as interwell tracers in field flooding for the enhanced recovery of oil and gas. This assessment document is intended to furnish the U.S. Nuclear Regulatory Commission (NRC) background material to produce a generic environmental impact statement. The results of the assessment are presented in terms of dose estimates for workers who handle the radioisotopes and for members of the public.

ENHANCED OIL RECOVERY (EOR)

Oil production from a producing well can be increased by injecting fluids (liquids or gases) to maintain reservoir pressure and push the oil toward production wells, a technique called secondary recovery. To increase oil production beyond the levels obtainable by secondary recovery, tertiary methods are used. Tertiary methods include miscible methods aimed at reducing the interfacial tension between the oil and the driving fluid, and thermal methods to reduce the viscosity of the oil by heating. In this report, enhanced oil recovery, or EOR, refers to both secondary and tertiary methods. It has been estimated that about 27 billion barrels of oil remain to be produced by primary and secondary methods in the U.S. and an additional 18-53 billion barrels can be recovered by current and advanced tertiary techniques.

USE OF RADIOISOTOPES IN EOR

Radioactive tracers are injected into underground reservoirs with liquids and gases to monitor the movement of the driving fluids and the efficiency of the recovery process. Isotopes that have been used as gaseous tracers in EOR operations include ^{3}H as HT or tritium-labelled hydrocarbons, $^{14}CO_2$ and ^{85}Kr. Isotopes used as liquid interwell tracers include ^{14}C, ^{22}Na, ^{35}S, ^{45}Ca, ^{57}Co, ^{58}Co, ^{60}Co, ^{63}Ni, ^{65}Zn, ^{85}Sr, ^{90}Sr, ^{110m}Ag, ^{125}I, and ^{131}I. Additional isotopes have been licensed as tracers in underground oil/gas wells and reservoirs. The quantities injected vary widely. For example, quantities of HTO varying from 0.010 Ci to 100 Ci have been injected in a single well in EOR tracer applications conducted in the U.S. It is estimated that about 100 EOR projects using radioactive tracers are conducted annually. Each tracer project involves an average of four or five separate injections of an isotope into a well.

EXPOSURE SCENARIOS

Normal and accidental exposure scenarios are described for three distinct stages of an injection operation: (1) the aboveground stage before, during, and immediately after injection of the tracer, (2) the subsurface stage when the tracer traverses the wellbore and underground reservoir, and (3) the recovery and disposal stage beginning with recovery of the isotope at the production well.

(1) Aboveground Stage. In the aboveground stage licensee workers may be exposed to radiation when the source is unpacked from a shipping container, carried to the

wellhead assembly and [illegible]
other post-injection operations are carried out.

During more than 20 years of experience, incidents or accidents involving the unplanned release of radioisotopes to the environment during EOR operations using tracers have been rare. However, aboveground scenarios that could lead to accidental exposures to workers can readily be postulated as a framework for dose estimates. These include: (1) leakage of a radioisotope solution from a container broken during shipment to the injection site, (2) spillage of a radioisotope solution on the ground or release of a gaseous radioisotope to the atmosphere after accidental dropping and breaking of the source container, (3) leakage of a radioactive liquid or gas through wellhead fittings during injection, and (4) leakage of a radioactive solution from a wellhead through valves inadvertently left open (human error). Exposures that could result from the occurrence of these scenarios include external doses (to workers) and doses from inhalation and submersion (to workers and to members of the public) from airborne radionuclides.

(2) Subsurface stage. Workers and members of the public are not subject to normal exposures during the subsurface stage. Members of the public could be exposed if the radioactive tracer should enter a freshwater aquifer that serves as a water supply. The entry of EOR injection fluids into underground sources of drinking water is unlikely because of the strict underground injection control (UIC) regulations. If the tracer-bearing injection fluid should enter a groundwater system, members of the public could be exposed by drinking the water, eating food crops irrigated with the water, or ingesting milk and meat from animals that had ingested the water or consumed vegetation irrigated with the water.

(3) Recovery and disposal stage. The fluids recovered at production wells contain the tracer at very low concentration. Since these fluids are separated from the oil and either reinjected or stored in an impervious pond, radioactive tracers in solution do not deliver exposures to workers or members of the public. However, members of the public are subject to doses from inhalation and submersion due to the HTO and ^{85}Kr in the combustion products of natural gas bearing residual tritium-labelled hydrocarbons and ^{85}Kr.

DOSE ESTIMATES

(1) Normal exposure. The annual dose to a worker from normal aboveground operations is dependent on the type and amount of radioisotope injected and on the number of injections performed in a year. If a worker were to inject all the radioisotopes projected for the United States in a year, he would receive a dose equivalent in the total body of about 470 mrem. As a worker is unlikely to perform more than 10 injection operations in a year (10% of the estimated injections), the annual dose equivalent in the worker is estimated to be less than 50 mrem. Assuming an average of two workers per project, the collective dose equivalent in workers is estimated to be 0.93 man-rem/y.

Normal exposures do not occur in the subsurface stage of operations.

The dose to members of the public from normal exposures in the recovery and disposal stage results from the residual tracer radioactivity in natural gas. The maximum dose equivalent in an individual is estimated to be 6×10^{-3} mrem/y and is attributed to the HTO in the combustion products of gas. The collective dose equivalent in the general population, using the Los Angeles basin as a frame of reference, was estimated to be about 15 man-rem/y. This dose estimate is much

less than the estimated 40,000 man-rem/y collective dose equivalent in the bronchial epithelium of the same population from naturally occurring ^{222}Rn in gas distribution lines, or the 1×10^6 man-rem/y collective effective dose equivalent in this population from natural sources of radiation.

(2) Accidental exposure. The annual collective dose equivalent in workers from accidents occuring in the aboveground stage is estimated to be 0.13 man-rem/y, which is about one-seventh of that from normal aboveground exposures. Individual doses from accidents would depend on the accident scenarios and the amount of radioisotopes involved. The collective dose equivalent in members of the public from the accidental release of gaseous radioisotopes is estimated to be 0.034 man-rem/y.

In the subsurface stage, the annual dose equivalent in a member of the public from drinking well water from a contaminated aquifer at 1 km from the well was estimated to be about 0.01 mrem/y. The collective dose equivalent from this accident scenario is estimated to be about 1.7×10^{-4} man-rem/y.

Accidental exposures do not occur in the recovery and disposal stage of operations.

(3) Total radiological impact. Summing the 0.93 man-rem/y collective dose equivalent in workers from normal exposures with the 0.13 man-rem/y from accidental exposures leads to a total collective dose equivalent in workers of 1.1 man-rem/y. It bears repeating that accidents are rare occurrences and that conservative values of accident frequency have been assumed in this assessment.

For members of the public, the total collective dose equivalent from normal exposures is estimated to be 15 man-rem/y, derived mainly from HTO in the combustion products of ^3H-labelled gaseous hydrocarbons recovered in natural gas. The collective dose equivalent from the accidental surface release of gaseous radioisotopes is estimated to be 0.034 man-rem/y, which is more than two orders of magnitude lower. The collective dose equivalent from drinking water drawn from a contaminated aquifer is 1.7×10^{-4} man-rem/y, which is less than 1/10,000 of the collective dose to the general public from normal operations. The contamination of the aquifer by salts, acids, alkali, and other substances would be a more pressing concern than the entry of a radioactive tracer.

Summing the collective dose to workers and members of the public yields a total collective dose equivalent of 16 man-rem/y. Because the calculations are conservative, our estimate of the national radiological impact of the use of radioisotopes as interwell tracers in EOR projects is a total collective dose equivalent of <16 man-rem/y.

ALTERNATIVES TO RADIOACTIVE TRACERS

Nonradioactive tracers have also been used to monitor the behavior of liquids and gases in petroleum reservoirs. Chemical tracers include ammonium thiocyanate, potassium iodide, ammonium nitrate, and SF_6 (gas). From the standpoints of cost and ease of handling, radioisotopes are generally preferable to chemicals as interwell tracers for EOR operations. However, chemical tracers are often used to complement radioactive tracers in EOR projects involving multiwell injections of different tracers. There have been situations where a nonradioactive chemical has been preferred over a radioactive tracer.

1. INTRODUCTION

1.1 BACKGROUND

This project has been commissioned by the U.S. Nuclear Regulatory Commission (NRC) to produce an environmental assessment of the use of radionuclides as tracers in field flooding for the enhanced recovery of oil and natural gas, which is commonly referred to as enhanced oil recovery (EOR). The assessment document is intended to furnish the NRC the background material to produce a generic environmental impact statement on the use of radioisotopes as interwell tracers for EOR. The NRC has commissioned this project because a detailed assessment of the use of radioisotopes as tracers in EOR operations has not previously been done.

1.2 SCOPE AND OBJECTIVES

This project is designed to evaluate the consequences of using radioisotopes as interwell tracers when liquids or gases are injected into underground reservoirs to maintain or increase the production of oil or gas. Individual doses and collective doses associated with both normal EOR operations and postulated accidents will be considered.

This study does not address exposures associated with the transportation of radioactive tracers from a packaging site to the injection site or the transportation of residual radioactive waste from the injection site to a disposal site. Regulations governing transportation of radioactive materials, including tracers for EOR projects and other applications, have been promulgated by the NRC and the Department of Transportation (10 CFR and 49 CFR). This study is not intended to address the potential impacts from using radioactive sources in well logging or other single-well tracer applications.

The study focuses on the radiological impacts associated with the use of radioisotope tracers in EOR. The environmental risks associated with EOR methods have been considered (Schumacher, 1978; Campbell, 1981). The radiological impact from the use of radioisotopes as tracers in EOR will be examined in the light of the other associated environmental concerns.

One goal of this project is to produce an environmental assessment of the use of radionuclides as interwell tracers in EOR. This assessment document will furnish the NRC the background material to prepare a generic environmental impact statement. A second goal of the project is to draft a separate guidance document after the environmental assessment has been completed. This guidance document will specify the information that an applicant should furnish the NRC to obtain a license to use radioactive tracers in EOR operations. The guidance document will serve as a model in formulating guidelines to keep occupational doses as low as reasonably achievable and to ensure the health and safety of the public when radioisotopes are used as interwell tracers in EOR operations.

1.3 DESCRIPTION OF ENHANCED OIL RECOVERY (EOR)

1.3.1 Methods for increasing petroleum production*

The oil from a producing well in a new reservoir initially flows because of the pressure exerted by water and gas in the reservoir. As oil production continues the

* Source: Schumacher (1978)

reservoir pressure declines unless fluids are injected into the reservoir to maintain the pressure. The average recovery from primary production, with and without pressure maintenance, is 20 to 30% of the original oil in place.

Oil production can be increased beyond the levels achieved by pressure maintenance through a technique called waterflooding, which is the injection of water through injection wells to push the oil toward production wells. This technique has been called secondary recovery. In practice it is often difficult to distinguish between secondary recovery and pressure maintenance because it is now customary to begin the waterflood before the reduced pressure results in an excessive decline in production.

So-called tertiary recovery methods have been used to increase oil production beyond the levels obtainable by secondary recovery methods, when some 60% or more of the oil may still be in place. The tertiary methods include miscible methods, aimed at reducing interfacial tension between the oil and the driving fluid, and thermal methods to reduce the viscosity of the oil by heating. The miscible methods include the following types of flood: surfactant polymer, polymer, alkaline, carbon dioxide, and miscible hydrocarbon. Thermal recovery methods include in situ combustion and steam flooding, the latter accounting for most of the oil recovered by tertiary methods.

In this document, enhanced oil recovery or EOR refers to any of the methods to increase oil (and gas) production beyond that obtainable from primary production alone.

1.3.2 Use of tracers in EOR

Tracers are used to define the movement of liquids or gases injected into an oil and gas reservoir to enhance recovery and to monitor reservoir performance. The water-soluble or gaseous tracer is introduced into a reservoir with the injected fluid. Both radioactive and nonradioactive tracers may be used. The tracer is placed in the injection well, where it is diluted and swept into the reservoir by injection liquid or gas. The diluted tracer is subsequently recovered at production wells and is monitored by sampling the recovered fluids.

In evaluating reservoir performance, it is desirable to determine the source of the injected fluid being collected at a production well. It is frequently desirable, therefore, to employ several tracers, using a different tracer in each of a number of injection wells.

1.3.3 Underground injection control

The potential contamination of a freshwater aquifer is a recognized hazard in oil production technology. The entry of EOR injection fluids into a freshwater aquifer serving as a water supply could cause serious pollution problems. The injection fluids used for secondary recovery are usually brines compatible with the endogenous fluids in the reservoir. The composition of dissolved solids varies greatly (Collins and Wright, 1982). Chemicals added to tertiary injection fluids could also lead to pollution problems if they entered a freshwater supply. Groundwater contamination is particularly bad because it is long lasting, difficult to trace, and may be far reaching.

Entry of EOR injection fluids into underground sources of drinking water is, however, extremely unlikely because of the strict underground injection control (UIC) regulations (Spears, 1980). Injection wells associated with EOR operations are designated Class II wells and are subject to stringent construction, operating, and monitoring and reporting requirements. Federal UIC regulations have been established by the Environmental Protection Agency (EPA) as required by the Safe Drinking Water Act (20 CFR 122-124, 146). Texas, Louisiana, and New Mexico are among the individual states that have achieved primacy for UIC by drawing up and administering a detailed regulatory program that meets the technical and procedural requirements established by EPA.

1.3.4 Estimates of petroleum recovery by EOR methods

During 1978-1981, crude oil production in the U.S. held level at 8.6 - 8.7 x 10^6 bbl/d (IPAA, 1982). Primary methods accounted for 47% of the production; enhanced recovery methods for 53% (Johnson, 1982). Production by EOR methods in 1980 was about 90% recovery by waterflood, and about 10% by tertiary methods (Matheny, 1980). Steam accounted for 77% of the U.S. tertiary production--up from 67% in 1978--and CO_2 and other gases accounted for 19% (Matheny, 1980).

Of the total of 460 billion barrels of oil discovered in the U.S., 121 billion barrels have been produced and 27 billion barrels remain to be produced by primary and secondary methods. Of the remaining 312 billion barrels that would remain in place, an additional 18 to 53 billion barrels can be recovered by current and advanced tertiary techniques (Johnson, 1982). It has been estimated that current tertiary recovery techniques could produce 1.1 x 10^6 bbl/d by the year 2000, and that more advanced techniques could account for another 2.7 x 10^6 bbl/d of production by then.

Radioactive-tracer projects by service companies that perform interwell tracing for EOR were estimated to number 50 to 60 per year (Bailey, 1982*). If we include projects performed by the oil companies themselves, the total number could be as large as 100 (Bailey, 1982*). The environmental assessment presented in this report is based on the assumption that the total number of EOR operations employing radioactive tracers is 100 per year. Based on the experience of two licensees, each tracer project involves an average of four to five separate injections of an isotope into a well. In this assessment it is assumed that each tracer project on the average involves four separate injections of an isotope.

---

* E.D. Bailey, Division of Licensing, Registration and Standards, Texas Department of Health, Austin, Tex., private communication (1982).

2. RADIOISOTOPES USED AS TRACERS IN EOR

As noted in Sec. 1, tracers are used in EOR operations to trace the movement of the injected fluid through the reservoir, to evaluate sweep efficiency, and thus to monitor the performance of the flood. Radioactive sources for tracing injected liquids are prepared as water-soluble material in aqueous solution; those for tracing injected gases are prepared in gaseous form.

Table 2.1 is a list of radiosiotopes that have been used as interwell tracers in EOR operations. The table was compiled from data furnished by one licensee for July 1, 1979, to November 16, 1982 and data for calendar-year 1982 furnished by another licensee. Table 2.1 shows the range of the amounts typically injected into a single well. As noted in the table, the injected amounts of some radioisotopes vary greatly. The range listed in Table 2.1 reflects the practice of only two licensees in the U.S. over a limited period of time; it is intended only to give a general indication of the quantities of radioactive materials used in individual tracer applications. It does not imply that smaller or larger quantities have not been used or will not be used. Specifically, Table 2.1 excludes the large amounts of some radionuclides that have been injected in foreign EOR operations.

Table 2.2 lists other radioisotopes that have been licensed as tracers in underground oil/gas wells and reservoirs through radioactive material licenses issued by the NRC

Table 2.1 Radioisotopes used as interwell tracers in EOR operations.[a]

Isotope	Half-life	Physical form	Chemical form	Range of amounts injected in a single well, Ci[b]
H-3 (T)	12.3 y	Gas	$HT, CH_3T, C_2H_5T, C_3H_7T$	0.10 - 30
		Liquid	HTO	0.010 - 100
C-14	5730 y	Gas	$^{14}CO_2$	0.030 - 0.80
		Liquid	Anion	0.00025 - 0.0015
Na-22	2.60 y	Liquid	Cation	0.020 - 0.040
S-35	87.2 d	Liquid	Anion	0.010
Ca-45	163 d	Liquid	Cation	0.001 - 0.010
Co-57	271 d	Liquid	Anion (complex)	0.005 - 0.30
Co-58	71.3 d	Liquid	Anion (complex)	0.005 - 0.20
Co-60	5.27 y	Liquid	Anion (complex)	0.001 - 0.30
Ni-63	100 y	Liquid	Anion (complex)	0.020 - 0.35
Zn-65	244 d	Liquid	Anion (complex)	0.020
Kr-85	10.7 y	Gas	Molecular	0.020 - 40
Sr-85	65.2 d	Liquid	Anion (complex)	0.010 - 0.040
Sr-90	29 y	Liquid	Anion (complex)	0.00004 - 0.100
Ag-110m	252 d	Liquid	Anion (complex)	0.020
I-125	59.7 d	Liquid	Anion	0.010 - 0.040
I-131	8.04 d	Liquid	Anion	0.008 - 0.050

[a] Based on data for the period July 1, 1979, to November 16, 1982, from one licensee and for 1982 from another licensee.

[b] The data pertain to EOR projects in the conterminous U.S. and Alaska.

Table 2.2 Other radioisotopes that have been licensed as tracers in underground oil/gas wells and reservoirs.

Isotope	Half-life	Physical form	Chemical form
P-32	14.3 d	Liquid	Anion
Cl-36	3.01×10^5 y	Liquid	Anion
Ar-37	34.7 d	Gas	Molecular
Ar-39	269 y	Gas	Molecular
Cr-51	27.7 d	Liquid	Anion
Fe-55	27 y	Liquid	Anion (complex)
Fe-59	44.6 d	Liquid	Anion (complex)
Sr-89	50.5 d	Liquid	Anion (complex)
Tc-99	2.13×10^5 y	Liquid	Anion
Xe-133	5.29 d	Gas	Molecular

or the state departments of health of California and Texas. These are two of the principal agreement states that account for the major share of U.S. oil production by EOR methods. Radioactive materials licenses sometimes cover a group of unspecified isotopes over a wide range of atomic numbers; they also sometimes cover authorized uses other than interwell tracing in petroleum reservoirs. Table 2.2 may therefore exclude some isotopes that are used as interwell tracers in EOR operations, and it may include isotopes that are not normally used as interwell tracers in EOR.

Certain isotopes that are authorized to be used for tracer studies in oil and gas wells, such as Sc-46, Zr-95, and Ir-192, are not listed in Table 2.2 because they are used in single-well applications rather than as interwell tracers.

The actual amount of radioactivity injected in an operation is determined by the licensee conducting the tracer operation in collaboration with reservoir engineers of the oil company. The reservoir engineers supply the needed information on reservoir characteristics and operating conditions such as rate of fluid injection, reservoir volume, and estimated time to breakthrough. The amount of isotope to be injected is determined taking into account the reservoir characteristics and is established so the expected concentration in the recovered fluid is sufficiently above the detection limits for accurate quantification but below the maximum concentration permissible for release into an unrestricted area. The amount of activity injected can be expected to vary widely from operation to operation.

The radioisotopes used to trace injection liquids in EOR are water-soluble species that do not distribute or exchange with the oil phase. The injection procedure is usually designed so the isotopes are stable anions in solution in the injection and reservoir fluids. As anions, the isotopes are minimally absorbed and unretarded in the reservoir. Soluble cationic species that are not adsorbed in geologic media are also used as tracers. Radioisotopes in the form of gaseous hydrocarbons are miscible with petroleum and are separated into the natural gas fraction. Inert gas isotopes of Ar, Kr, and Xe are thought to remain in the natural gas fraction.

Tritium as HT is separated from the natural gas fraction along with N_2, and tritium as HTO is separated from natural gas (and petroleum) when it is dried. $^{14}CO_2$ is separated from natural gas when it is processed.

The persons who inject tracers are employees of licensees of the NRC or agreement states, who are well trained in the principles and practice of health physics. The radioisotopes for injection are transported to the job site in containers meeting Department of Transportation (DOT) specifications. Typically an injection is carried out as follows: The vial containing the radioactive tracer is carefully removed from the shipping container and checked for damage. It is carried with special tools to the wellhead and placed inside a special assembly connected to the wellhead. The system is closed, the vial is crushed, and valves are opened allowing the injection liquid (or gas) to flush the radioisotope down the wellbore into the reservoir.

3. DESCRIPTION OF EXPOSURE SCENARIOS

This section describes realistic radiation exposure scenarios for EOR operations that use radionuclides for interwell tracing. It is convenient to consider three distinct stages of operation:

(1) The aboveground stage before, during, and immediately after injection of the tracer,

(2) The subsurface stage when the tracer traverses the wellbore and underground reservoir, and

(3) The recovery and disposal stage beginning with the recovery of the isotope at the production well.

Two categories of exposure pathways are considered: normal (Table 3.1) and accidental (Table 3.2). The dose estimates for these scenarios are presented and discussed in detail in Sec. 6.

3.1 ABOVEGROUND STAGE

3.1.1 Normal exposures

As noted in Sec. 2, the persons who inject radioactive tracers in EOR operations are licensee employees who are well trained in health physics and are expected to observe sound practices in the field. They routinely wear personnel monitoring devices such as thermoluminescent dosimeter (TLD) badges and other dosimeters. Unshielded sources are handled at a distance using special tools, and exposure time is kept short. To minimize leakage of radioactivity from the wellhead assembly, a practice run may precede the actual injection as a check for leaks. Survey meter readings are taken periodically to assure that the vial containing the radioisotope has not been damaged in transit, that the isotope is not held up somewhere in the wellhead assembly during flushing, and that leaks that could release activity to the environment are detected. To check for the presence of transferable radioactivity on accessible surfaces, smears may be taken and counted in the field or returned to

Table 3.1 Normal exposure pathways.

Stage	Exposure pathway	Exposed group
Aboveground	External irradiation Possible inhalation of low-level residual HTO, $^{14}CO_2$	Workers
Subsurface	None	None
Recovery and disposal	Inhalation and submersion exposures to HTO, ^{85}Kr, and $^{14}CO_2$ released at production well sites and in combustion products of natural gas	Members of the public

Table 3.2 - Accidental exposure pathways.

Stage	Accident scenarios	Exposure pathways	Exposed group
Aboveground	Leaky source container Source spilled on ground or released to atmosphere Leakage through wellhead assembly Leakage due to human error	External irradiation Inhalation Submersion	Workers Members of the public
Subsurface	Leakage into a freshwater aquifer	Ingestion of the water, produce irrigated with the water, and milk and meat from animals that drank the water and ingested forage irrigated with the water	Members of the public
Recovery and disposal	None	None	None

the laboratory for analysis. Members of the public are not present during injections, and oil company workers not involved in the tracer injections are kept at a distance—particularly when there is a potential for exposure to an unshielded source.

Occupational doses from normal exposures during an EOR radioactive tracer operation are very low. The average external whole-body occupational dose to workers handling radioactive tracers on an EOR project as determined from TLD's was reported by one licensee as 14 mrem with a maximum of 300 mrem. Most doses were below the limits of detection.

A careful reconstruction of the aboveground stage of an interwell radioactive tracer operation for EOR reveals a number of definitive activities when a worker is in close proximity to the source:

A. Unloading the isotope package from the transport vehicle
B. Unpacking the isotope container
C. Removing the shielding and carrying the isotope to the injection port
D. Injecting the isotope and flushing
E. Removing the isotope container from the wellhead and other post-injection operations
F. Handling the tools bearing residual activity during clean up
G. Handling the waste materials bearing traces of radioactivity

Activity C is potentially the most important because the worker may be then exposed to an unshielded source. Activities A and B are potentially important because of the time spent close to the isotope package. In the other activities the worker is exposed to a partially shielded source or small amounts of a diluted radioisotope.

The injection process is subject to minor variations. The research division of one oil company injects tracers into wells by crushing the vials of radioactive material into the feed tank of a small chemical pump connected to the wellhead (D'Hooge et al., 1981). Repeated rinses and monitoring for radioactive content ensure that no contamination remains on the surface. The radioisotope division of another oil company uses an injection method in which the radioactive liquid or gas to be injected is confined in a shielded stainless steel container. The container serves as a fitting that is directly coupled into the wellhead assembly and becomes part of the drive system. The shielding remains in place throughout the whole operation so that external exposure to workers is essentially eliminated.

3.1.2 Accidental exposures

In more than 20 years of experience, incidents or accidents with unplanned release of radioisotopes to the environment during interwell tracing in EOR have been rare. However, a number of scenarios that could lead to accidental exposures can readily be postulated for the aboveground stage:

A. If a vial containing a radioactive liquid were leaking, there could be exposures from handling (repacking the leaky or broken vial for shipment to a disposal site) and a resultant external or internal dose to workers. This scenario has occurred. Confinement of the source in a shielded container similar to that described in Sec. 3.1.1 would virtually eliminate this scenario.

B. If a vial containing a radioactive liquid were accidentally dropped, spilling its contents on the ground or skin, there could be an exposure from the radionuclide on the ground or skin. There could also be an exposure associated with cleaning up the liquid. The exposures could lead to external or internal doses to workers. Only a small fraction of the glass vials dropped can be expected to break and spill its contents.

C. If a glass vial containing a radioactive gas were accidentally dropped releasing its contents, there could be exposures from the airborne radionuclide. The exposures could lead to inhalation and submersion doses to workers and to members of the public downwind. Only a small fraction of the glass vials dropped can be expected to break and release radionuclides to the atmosphere.

D. If a tracer liquid or gas were to leak through wellhead fittings during injection, there could be an exposure from the isotope on the surface of the wellhead assembly, on the ground (or skin), or from contact with the airborne isotope. External doses or inhalation and submersion doses to workers would result. Occurrence of this scenario is unlikely if the wellhead is checked for leaks before the tracer is injected.

E. If a port of an injection wellhead were inadvertently left open by human error, the nuclide solution could readily be transported to the surface of the wellhead, the ground, or the skin with resultant external doses to workers.

In summary, an occurrence of the above scenarios could lead to external doses to workers from radionuclides deposited on the ground or wellhead surfaces, to doses to the skin of workers from radionuclides deposited on the skin, or to doses from inhalation and submersion to workers and members of the public from airborne radionuclides.

It is appropriate to mention the following incident that was reported to the NRC. A portion of the HTO injected as an EOR tracer during winter froze inside a special device that was in place in the aboveground portion of the wellhead assembly of the injection well. The frozen water expanded, cracking the device, which leaked water to the ground when the device was thawed. Additional water backflowed from the well and spilled to the ground when the device was disconnected. Urine samples, from the four workers who were exposed to the spilled water but were not present during the tracer injection ranged from 11 to 164 dpm/cm^3—well within regulatory limits. The corresponding range of dose commitments to the whole body was 0.04 to 0.5 mrem.

3.2 SUBSURFACE STAGE

3.2.1 Normal exposures

Exposures during the subsurface stage of an EOR operation using radioactive tracers would not occur if the tracer moved according to plan and remained underground, far removed from workers or members of the public.

3.2.2 Accidental exposures

As noted in Sec. 1.3.3, entry of EOR injection fluids into underground sources of drinking water is unlikely because of the strict underground injection control (UIC) regulations. The construction requirements for Class II wells include the following:

1. Injections must be into zones that are free of known faults or fractures.
2. Injection wells must be cased and cemented to prevent movement of fluids into or between underground sources of drinking water.
3. Appropriate logs and other tests must be performed during construction.
4. The injection must be described in terms of fluid pressure, temperature, fracture pressure, physical and chemical character of the injection zone and formation fluids.

The operator must limit the injection pressure to ensure that new fractures do not form in the confining strata, and he must monitor the nature of the injected fluids and the pressure, flow rate, and volume. Because of the monitoring requirements the operator would know if mechanical integrity should fail and allow a significant entry of injection fluid into an underground source of drinking water. He would cease injecting, take corrective measures, and report the incident to the appropriate regulatory agencies.

As noted above, the entry of a radioactive-tracer-bearing EOR injection fluid into a freshwater aquifer can be described as an event associated with a very low probability. Furthermore, economic incentives would minimize the amount of leakage that occurs. Leakage from an injection well not only could contaminate underground water supplies with brine and other chemicals, it could well reduce the

effectiveness of the recovery process and result in the loss of chemicals. Nevertheless, it is important in this study to evaluate the radiological consequences of such an event and to place them in perspective.

Unplanned events that would allow entry of tracer-bearing injection fluid into a groundwater system include the following (Schumacher, 1978):

- Leakage through the tubing, the casing, and the cement surrounding the casing into a freshwater aquifer,
- Vertical escape from the injection zone to a freshwater aquifer outside the casing or the cement surrounding the casing,
- Vertical escape from the injection zone through confining beds to freshwater aquifers via newly formed fractures and channels,
- Entry into a freshwater aquifer via badly plugged abandoned wells.

Fig. 3.1, from Braxton et al. (1976), illustrates the contamination of freshwater through bypass of the natural filter system.

The exposure modes associated with contamination of underground freshwater supplies are the doses to members of the public from drinking the water, eating food crops irrigated with the water, and ingesting milk and meat from animals that drank the water and ate vegetation that was irrigated with the water. The potential doses to members of the public will be estimated and the resultant radiological hazard will be examined in the light of the other hazards that would attend entry of injection fluid into the aquifer. The radiological hazard will also be considered in the light of the concentrations of naturally occurring radionuclides in brines and water supplies (see Sec. 6).

3.3 RECOVERY AND DISPOSAL STAGE

3.3.1 Normal exposures

The liquids and gases recovered at production wells contain the radioactive tracer at very low concentration. As noted previously, the amount of injected tracer is designed to yield concentrations that are measurable but below the maximum permissible concentrations for release to an unrestricted area. The very low concentrations necessitate no special requirements regarding handling, containers, and shipping procedures; and external doses to workers who may be involved in sample collection would be trivial. In practice, injection liquids are separated from the oil and, after appropriate chemical treatment, are reinjected into the reservoir. Recovered liquids are sometimes stored in an impervious storage pond. In any event the radioisotopes in the recovered liquids are separated from the petroleum and remain with the aqueous phase without further dispersion in the environment. The radioisotopes used to trace injected liquids in EOR operations are water-soluble species that do not exchange with oil.

Gaseous radioactive tracers may be miscible in the injection fluid or the oil. Thus ^{3}H-labelled methane and other low molecular weight ^{3}H-labelled hydrocarbons are soluble in oil and would appear in the recovered petroleum. They are later separated from the petroleum, however, and become part of the natural gas fraction. $^{14}CO_2$ is miscible in oil but is separated from the oil and removed from natural gas during processing. The CO_2 recovered in CO_2 floods is sometimes reused and sometimes released to the environment. The noble gas isotope, ^{85}Kr,

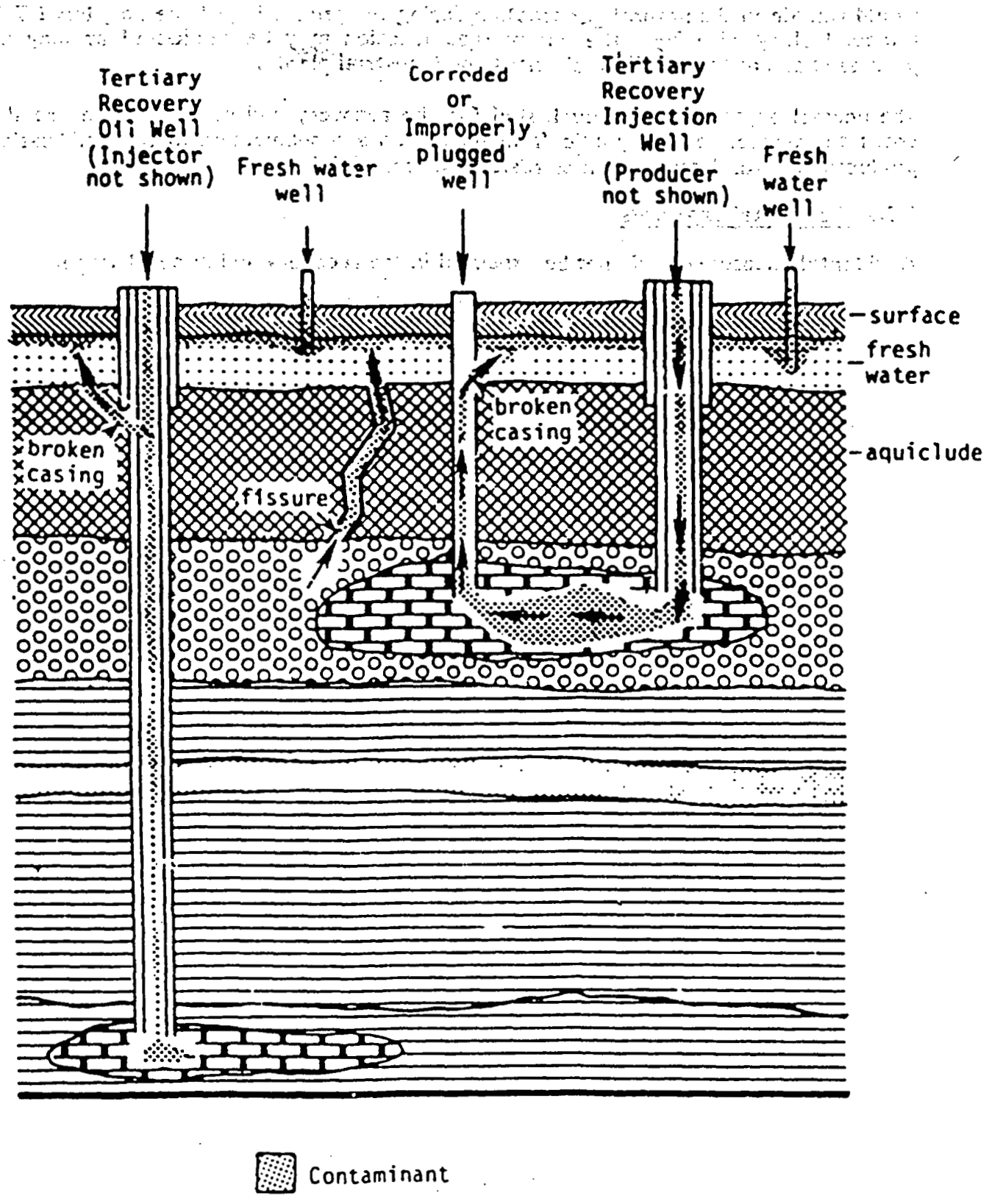

Figure 3.1 Contamination of fresh water through by-pass of the natural filter system. (Source: Braxton et al., 1976.)

would remain in the natural gas fraction during the processing of the gas, but HT is removed along with N_2. The natural gas fraction may be reinjected or may be processed and distributed to residences and industrial plants.

The normal exposures to be evaluated for the recovery and disposal stage are the doses to members of the public from inhalation and submersion in the combustion products of natural gas containing residual tracer activity.

3.3.2 Accidental exposures

Accidental exposures would not be expected in the recovery and disposal stage.

4. DESCRIPTION OF REFERENCE SITES

This section characterizes the reference sites and presents the assumed data needed to assess the dose for two of the accident scenarios:

- Collective dose to the general population from exposures to an accidental release of a gaseous radioisotope
- Dose to members of the public from using water drawn from an aquifer contaminated by a radioactive tracer

Three distinct reference sites are identified: (1) a rural site, (2) an urban site, and (3) a more or less typical site. The collective dose is calculated over a circular region extending 80 km (50 mi) from the reference site. The collective dose estimated for the "typical" site will serve as a reference for calculating the nationwide impact of the use of radioisotopes for interwell tracing in EOR. The collective dose estimates for the urban site are intended to represent the maximum for the scenario involving an accidental release of a gaseous radioisotope.

The characteristics of the freshwater aquifer that the injection well traverses are assumed to be the same for all sites. Oil/gas reservoir characteristics are not specified as they are not needed for dose estimation.

4.1 RURAL SITE

The rural site is located in a rural region characterized by a population density of 13.5/km^2 (35/mi^2) uniformly distributed throughout the region. This corresponds to a total population of 271,000 in the region.

4.2 URBAN SITE

The urban site is in an urban region characterized by a population density of 695/km^2 (1800/mi^2) uniformly distributed throughout the region. This corresponds to a total population of about 14,000,000 in the region.

4.3 "TYPICAL" SITE

The "typical" reference site is located 32 km upwind of the center of a square urban area that is 15 km on a side. The population density in this area is 695/km^2 (1800/mi^2), which corresponds to an urban population of 156,000. The population density elsewhere in the region is 13.5/km^2 (35/mi^2), for an additional 268,000, or a total population in the region of 424,000 (see Fig. 4.1).

4.4 FEATURES COMMON TO ALL SITES

Based on several sources, parameter values considered to be reasonable were selected for the freshwater aquifer underlying each of the sites (Schumacher, 1982; Freeze and Cherry, 1979; McWhorter and Sunada, 1977):

Depth to base of aquifer	130 m
Depth of aquifer	80 m
Groundwater velocity	0.2 m/d
Effective porosity	0.25
Longitudinal dispersivity	40 m
Transverse dispersivity	8.0 m

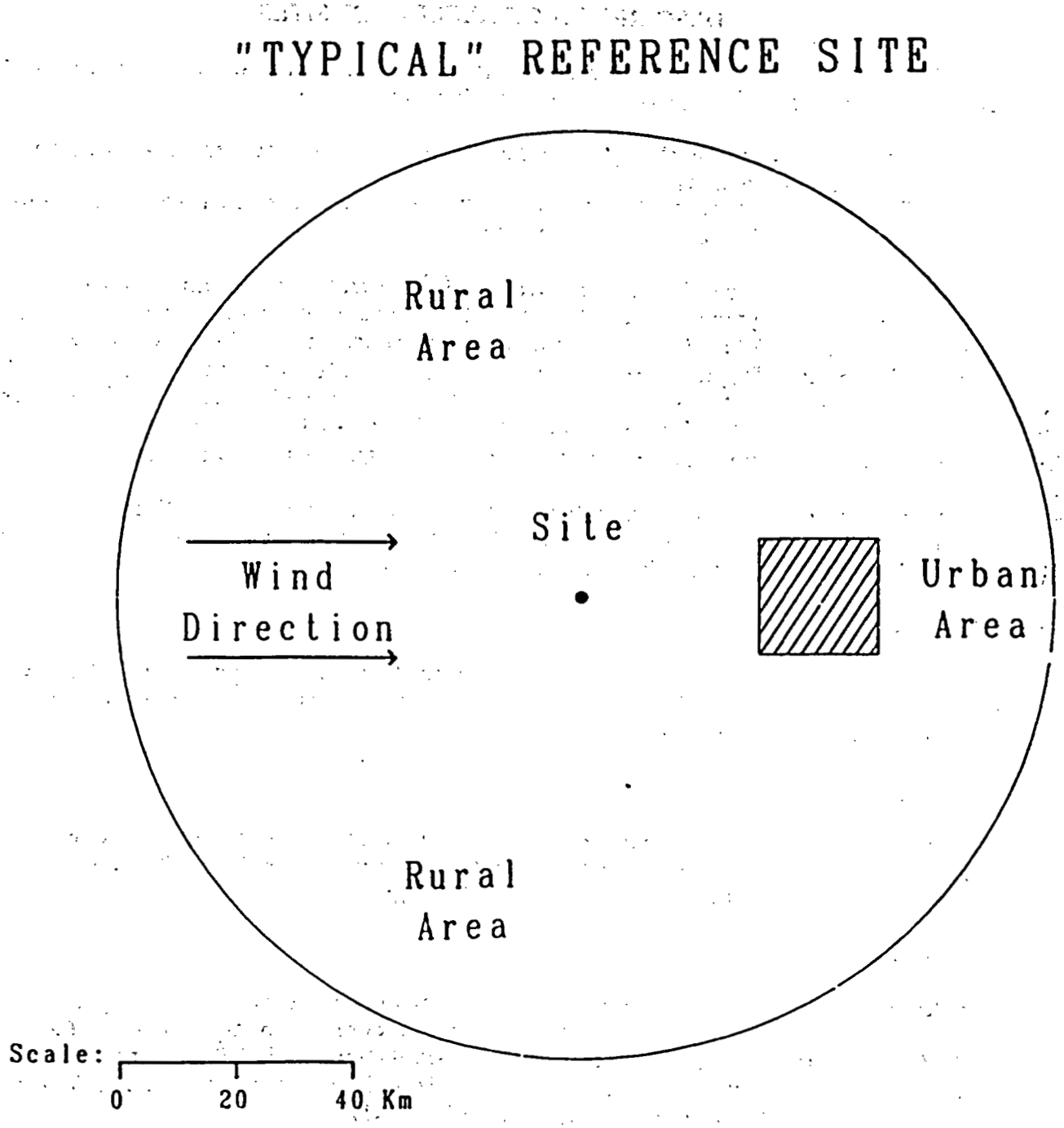

Figure 4.1 Region surrounding the "typical" reference site.

A well supplying drinking water is assumed to intersect the aquifer 1.0 km downgradient from the site on the centerline. The well draws water from the full height of the aquifer.

Three meteorological scenarios were chosen to cover the range of conditions that could occur at any reference site during daytime; separate calculations were made for each (to be discussed in Sec. 6.1.2.3).

5. METHODS FOR ESTIMATING THE DOSE

This section briefly describes the methods used to estimate the doses associated with the exposure scenarios discussed in Sec. 3. More detailed descriptions of the methods are presented in the Appendices.

5.1 EXTERNAL DOSES

External doses to workers from aboveground normal exposures were inferred from personnel and other records obtained in the course of tracer-injection operations (personnel-monitoring records, survey-meter readings, bioassays, etc.). These data are considered in the light of the particular isotopes used and source strength. In addition, exposure rates were calculated for individual isotopes on the basis of source strength and configuration, distance from source to receptor, and the radiological properties of the isotope. Estimation of the exposure rate in air at various distances from a point source and from a distributed source is described in detail in Appendix A. The external exposure rate from a distributed source is of interest for scenarios in which some of the radionuclide is accidentally spilled on the ground or the injection wellhead surface.

5.2 DOSES FROM AIRBORNE RADIONUCLIDES

The potential dose to workers from exposure to a gaseous tracer released from an accidentally broken vial requires an estimate of the integrated air concentration (IAC) for individuals positioned at or near a ground-level release. The IAC was estimated using a Gaussian model. Dose estimates from inhalation and submersion were then estimated using nuclide-specific dose factors from ICRP-30 (ICRP, 1979; 1980; 1982). The collective doses to the general population from a gaseous ground-level release were determined using a particle-in-cell model to estimate the IAC at distances up to 80 km from the release. The atmospheric transport and diffusion calculations are described in Appendix B.

5.3 DOSES FROM INGESTING WATER FROM A CONTAMINATED AQUIFER

The dose to members of the public from using well water drawn from an aquifer accidentally contaminated by a pulse of radionuclides was estimated using a three-dimensional hydrologic advection-dispersion model to calculate the concentration of a nuclide in an aquifer at various distances from the point of entry of the nuclide. The hydrologic transport calculations and dose assessment are described in Appendix C.

5.4 DOSES FROM THE COMBUSTION OF NATURAL-GAS-BEARING TRACER ACTIVITY

The collective dose to the general population from residual tracer activity in natural gas is estimated by scaling to previous estimates of the potential dose from the radionuclides that would be present in natural gas produced from reservoirs stimulated by nuclear explosives. Experiments that were performed to determine the feasibility of stimulating gas reservoirs using nuclear explosives include the Gasbuggy, Rulison, and Rio Blanco projects, which were conducted during 1967-1973 (Nordyke, 1971; Tewes, 1979). The dose estimates associated with the use of gas from wells stimulated by nuclear explosives are reviewed in Appendix D.

6. DOSE ESTIMATES FOR EOR EXPOSURE SCENARIOS

This section presents the dose estimates associated with the use of radioisotopes as interwell tracers in EOR operations. Dose estimates are presented for workers and members of the public according to the normal and accidental exposure scenarios described in Sec. 3.

6.1 ABOVEGROUND STAGE

6.1.1 Normal exposures

Routine exposures during the aboveground stage of an EOR operation using radioactive tracers could result in an external dose to the workers who handle the radioisotopes. The annual dose to workers could be expected to depend on both the number of operations carried out and the total activity injected. To evaluate these situations we first evaluate the potential external dose to workers from a typical EOR waterflood using 20 mCi of ^{60}Co as an interwell tracer. We then calculate the external dose per mCi of ^{60}Co injected. Then using dose rate constants, Γ, and exposure rates listed in Appendix A, we estimate the external dose per mCi injected for the other radioactive tracers. The dose from each radioisotope is estimated from the activities injected and the total dose is estimated as the sum of the doses associated with each radioisotope.

6.1.1.1 Estimation of the external dose to workers from ^{60}Co. The source is assumed to be a solution of 20 mCi of ^{60}Co contained in a glass vial. The dose to workers from normal operations is evaluated by estimating the exposures associated with each of the activities described in Sec. 3.1.1.

1. We first estimate the combined exposure that is associated with operations A and B. Operation A is unloading the isotope package from the transport vehicle; operation B is unpacking the isotope container (see Sec. 3.1.1). The package is assumed to be a 16-in. cube with a Transport Index (TI) of 3, which signifies that the maximum exposure rate at 3 ft (36 in.) from the surface of the package is 3 mR/h. This assumption is based on information from a licensee who advised us that the packaging of gamma-emitting isotopes for vehicular transportation on land commonly results in a TI of 2 to 3. The source is at the center of the package and thus is 8 in. from the surface. The worker is assumed to be at an average distance of 6 in. from the sides of the package for a period of 5 minutes. The exposure rate is

$$ER(A,B) = (44^2/14^2)3 = 30 \text{ mR/h}$$

and the total exposure is

$$E(A,B) = (5/60)30 = 2.5 \text{ mR}$$

2. We next consider the exposure during operation C, removing the shielding and carrying the isotope to the injection port. The unshielded vial containing the radioisotope is assumed to remain at a distance of 2 ft ($\sim$ 0.6 m) by means of a special handling tool and extending the arms. The exposure time, based on direct observation and information from a licensee, is assumed to be 10 s. The exposure rate 0.6 m from a 1-mCi ^{60}Co source is 3.49 mR/h (see Table A.2 in Appendix A). The exposure rate for a 20-mCi ^{60}Co source would be

$$20 \times 3.49 = 70 \text{ mR/h}$$

The total exposure for a 10-s period would be

$$E(C) = (10/3600)70 = 0.19 \text{ mR}$$

3. Operation D, injecting and flushing the radioisotope down the wellbore, involves the presence of a worker close to the wellhead for a short time while the tracer essentially is still in the aboveground portion of the wellbore. The exposure is calculated assuming that the exposure rate at the surface of the wellhead is 3500 mR/h, that the surface is 5 cm from the source, and that the worker is located at a distance of 0.4 m from the source for a period of 30 s. The exposure rate at the surface of the wellhead is based on information from a licensee and is consistent with the 7000 mR/h exposure rate measured at the surface of a glass vial containing 20 mCi of a ^{60}Co solution when attenuation by the source holder and wall of the wellhead assembly is taken into account.

   The exposure rate at a distance of 0.4 m, assuming the inverse square law, is

$$ER(0.4 \text{ m}) = [(0.050)^2/(0.4)^2]3500 = 55 \text{ mR/h}$$

   The total exposure for a 30-second period is

$$E(D) = (30/3600)(55) = 0.46 \text{ mR}$$

4. The combined exposure associated with the post-injection operations at the wellhead, cleanup, and waste disposal (operations E,F,G) is based on measurements indicating that the exposure rate is ≤ 5 mR/h at the surface of waste residues. We also assume that the worker spends 15 min 2-3 ft from the waste residues where the exposure rate is ~ 0.5 mR/h. The total exposure is

$$E(E,F,G) = (15/60)\ 0.5 = 0.13 \text{ mR}$$

5. The total exposure to workers for normal aboveground operations is estimated by summing the four contributions:

$$E = 2.5 + 0.19 + 0.46 + 0.13 = 3.3 \text{ mR}$$

It is instructive to reexamine the four contributions to the total exposure:

$$E(\text{total}) = E(A,B) + E(C) + E(D) + E(E,F,G)$$

1. E(A,B) results from handling the package in which the source is transported. Since in practice the TI is fairly constant for different activities of a radioisotope, E(A,B) could be expected to be constant for each injection operation. The total exposure from operations A and B would be expected to vary directly with the number of separate injections of the isotope and to be independent of the activities injected.

2. E(C) results from proximity to the unshielded source as it is carried to the wellhead assembly of the injection well; it could be expected to vary directly with the activity of the isotope.

3. E(D) results from proximity to the source in the wellhead assembly before and during its release and injection into the wellbore. E(D) could be expected to vary directly with the activity of the isotope.

4. E(E,F,G) results from handling the waste residues during cleanup. E(E,F,G) could be expected to be independent of the activity injected and to be constant as the activity of the isotope varies. The total exposure from operations E, F, and G would be expected to vary directly with the number of separate injections of the isotope and to be independent of the activities injected.

These considerations lead to the conclusion that a portion of the total exposure to workers from the injection of 20 mCi of a ^{60}Co tracer is constant for each injection of the isotope and that a portion varies directly with the activity of the isotope. The portion that is constant for each injection, E_C, is

$$\begin{aligned} E_C &= E(A,B) + E(E,F,G) \\ &= 2.5 + 0.13 \\ &= 2.6 \text{ mR/injection} \end{aligned}$$

The portion that varies directly with the activity, E_P, is

$$\begin{aligned} E_P &= E(C) + E(D) \\ &= 0.19 + 0.46 \\ &= 0.65 \text{ mR} \end{aligned}$$

E_P can be expressed as a normalized exposure, E'_P:

$$E'_P = 0.65 \text{ mR}/20 \text{ mCi} = 0.033 \text{ mR/mCi}$$

It should be noted that this approach for estimating the normalized exposure for a radioisotope used as an interwell tracer in EOR operations is only an approximation. Because characteristics of injection sites, construction of wellhead assemblies, procedures employed by licensees, operating conditions of EOR projects, and the physical, chemical, and radiological properties of radioactive tracers differ widely, each injection operation is unique. Therefore, the dose to a licensee worker from handling a given quantity of an isotope could be expected to vary substantially. It is also reasonable to anticipate that the care and attention exercised by licensee workers in an injection operation will vary with the nature and quantity of the radioisotope that is handled. Thus, greater care may be used in handling large quantities of gamma-emitting isotopes, and special procedures may be followed. In 22 separate injections of ^{60}Co performed by a licensee in which the activity injected into a well varied from 10 to 200 mCi, the whole body dosimeter readings for workers varied from 0 to 40 mR. The mean exposure per operation was 13 ± 16 mR. The mean normalized exposure was 0.12 ± 0.17 mR/mCi injected, but the exposure was not strongly correlated with the quantity injected ($r = 0.56$). The same licensee reported that in one foreign project, in which a total of 2.1 Ci of ^{60}Co was injected and the largest single injection was 700 mCi, the highest exposure to any worker was 40 mR. Another licensee has estimated that the injection of 1 Ci of ^{60}Co is associated with an exposure of 50 to 150 mR to a licensee worker.

Despite the associated uncertainties, this approach allows us to consider differences in the radiological properties and quantities of the isotopes employed and yields estimates that are consistent with exposures that have been recorded in the field.

6.1.1.2 Estimation of external dose to workers from other water-soluble radioisotopes. The external dose to workers from injecting other water-soluble radioactive tracers is estimated in the same manner as illustrated above for ^{60}Co. A TI value of 3.0 was assumed for all gamma-emitting isotopes in evaluating operations A and B. (The gamma-emitting isotopes are those radionuclides in Table A.1 of Appendix A for which the dose-rate constant, Γ, is greater than zero.)

The exposures to workers from liquid gamma-emitters in operations A, B, E, F, and G are assumed to be independent of the activity injected, as in the case of ^{60}Co, and to be the same as that for ^{60}Co, 2.6 mR/injection. The exposure from operations C and D are assumed to vary directly with the activity injected. The normalized exposure for operations C and D was calculated as that for ^{60}Co, 0.033 mR/mCi (from Sec. 6.1.1.1) times the ratio of the dose-rate constants, Γ, of the particular radioisotope and ^{60}Co. (Γ values are listed in Table A.1 in Appendix A.) Thus,

$$E_C = E(A,B) + E(E,F,G) = 2.6 \text{ mR/injection}$$

$$E_{P^-} = E(C) + E(D) = 0.033\,[\Gamma(Z,A)/\Gamma(27,60)]$$

where

$\Gamma(Z,A)$ = dose-rate constant, Γ, for a radioisotope of atomic number Z and mass A.

The exposure from bremsstrahlung resulting from beta emission by ^{90}Sr-^{90}Y was estimated assuming that the bremsstrahlung hazard for a 1-Ci source of ^{90}Sr is approximately equal to that presented by the gamma from 12 mg of radium (BRH, 1970). Since the dose-rate constant, Γ, of ^{226}Ra is 8.25 cm^2-R/mCi-h and the specific activity of ^{226}Ra is 1.0 Ci/g, the exposure to ^{90}Sr from bremsstrahlung was calculated assuming a Γ value of

$$\Gamma = (12/1000)8.25 = 0.1 \text{ cm}^2\text{-R/mCi-h}$$

As noted in Sec. 6.1.1.1, the normal exposure from operations A and B is calculated assuming a 5-min exposure at a location 14 in. from a point source. The exposure rate 14 in. from 1 mCi of ^{90}Sr by the inverse square law is

$$ER(A,B) = (100 \text{ cm}^2\text{-mR/h})/(2.54 \text{ cm/in} \cdot 14 \text{ in})^2 = \cancel{7.9 \times} 10^{-2} \cancel{\text{mR}}\text{/h per mCi}$$

and the normalized exposure from operations A and B is

$$E(A,B) = (5/60)7.9 \times 10^{-2} = 6.6 \times 10^{-3} \text{ mR per mCi}$$

The above calculations are conservative because they assume that the ^{90}Sr source is unshielded. If the quantity of ^{90}Sr in the source were sufficiently high, shielding would have to be taken into account. The normal exposure from operations ~~A and B~~ due to gamma-emitting sources is calculated assuming a TI value of 3.0 for the source package and is estimated to be (see Sec. 6.1.1.1)

$$E(A,B) = 2.5 \text{ mR}$$

The maximum amount of ^{90}Sr that would not exceed a TI value of 3.0 would be

$$2.5/6.6 \times 10^{-3} = 380 \text{ mCi}$$

The exposures from operations C and D due to ^{90}Sr were estimated in the same manner as those due to gamma emitters. The exposure from operations E, F, and G is assumed to be zero because the residual activity would be very low and contribute a negligible exposure rate.

The exposure from bremsstrahlung originating in other beta emitters has not been included because the maximum beta energy for these radioisotopes is low relative to that for ^{90}Sr so that the bremsstrahlung would be less hazardous.

Table 6.1 summarizes the calculations for the normalized exposure of soluble radioisotopes.

Table 6.1 Estimation of external dose to workers from normal aboveground operations associated with injection of liquid tracers.[a]

Normalized exposure E', mR per injection operation or mR per mCi injected				
Radioisotope	Operations A,B mR/injection	Operation C mR/mCi	Operation D mR/mCi	Operations E,F,G mR/injection
H-3	0	0	0	0
C-14	0	0	0	0
Na-22	2.5	0.0087	0.021	0.13
S-35	0	0	0	0
Ca-45	0	0	0	0
Co-57	2.5	0.0007	0.0018	0.13
Co-58	2.5	0.0040	0.0097	0.13
Co-60	2.5	0.0095	0.023	0.13
Ni-63	0	0	0	0
Zn-65	2.5	0.0023	0.0055	0.13
Sr-85	2.5	0.0047	0.011	0.13
Sr-90	0.0066[b]	0.00007	0.0002	0
Ag-110m	2.5	0.011	0.026	0.13
I-125	2.5	0.0013	0.0030	0.13
I-131	2.5	0.0016	0.0039	0.13

[a] The operations, which are discussed in Sec. 3.1.1, include the following:

- A Unloading the isotope package
- B Unpacking the isotope container
- C Removing the shielding and carrying the isotope to the injection port
- D Injecting and flushing
- E Post-injection operations
- F Cleaning up
- G Packing residues for waste disposal

[b] Normalized exposure expressed is mR/mCi.

6.1.1.3 Estimation of external dose to workers from gaseous radioisotopes. The external dose to workers from gaseous radioisotopes is estimated in similar fashion. Only operations A, B, C, and D need to be evaluated because the exposures from operations E, F, and G are zero. Operations A and B combined were evaluated using the same assumptions regarding package geometry and package handling described for ^{60}Co in Sec. 6.1.1.1.:

Of the gaseous radioisotopes used as interwell tracers in EOR operations, only ^{85}Kr emits photons. If a glass vial containing 1 Ci of ^{85}Kr were unshielded, the exposure rate at 1 m, assuming point-source geometry, is 1.22 mR/h (from Table A.2 in Appendix A). The exposure rate at 44 in. (1.12 m) is estimated from the inverse square law to be 0.97 mR/h, which is approximately 1 mR/h. Accordingly, if the ^{85}Kr source is 1 Ci or more in activity, the TI is assumed to be 1.0 and E(A,B) is assumed to be independent of source activity. E(A,B) for ^{85}Kr is estimated in the same manner as for ^{60}Co in Sec. 6.1.1.1:

$$E(A,B) = 0.82 \text{ mR/injection}$$

If the ^{85}Kr source is less than 1 Ci in activity, the source is assumed to be unshielded and E(A,B) is assumed to be directly proportional to the source activity. The normalized exposure, E'(A,B) is then

$$E'(A,B) = 0.82 \text{ mR}/1000 \text{ mCi} = 0.00082 \text{ mR/mCi}$$

To evaluate operation C for gaseous radionuclides, we assume, based on information from a licensee, a 15-s exposure to an unshielded source at a distance of 0.3 m from the body. Operation D is evaluated in the same manner as for liquid tracers. Table 6.2 summarizes the calculations for the normalized exposure of gaseous radioisotopes.

Table 6.2 Estimation of external dose to workers from the normal aboveground operations associated with the injection of gaseous tracers.[a]

Normalized exposure E', mR per injection operation or mR per mCi injected				
	Operations A,B		Operation C	Operation D
Radioisotope	mR/injection	mR/mCi	mR/mCi	mR/mCi
H-3	0		0	0
C-14	0		0	0
Kr-85, activity $\geq$1 Ci	0.82		3×10^{-5}	2×10^{-5}
Kr-85, activity $<$1 Ci		8.2×10^{-4}	3×10^{-5}	2×10^{-5}

[a]The operations include the following (see Sec. 3.1.1)

A Unloading the isotope package
B Unpacking the isotope container
C Removing the shielding (if any) and carrying the isotope to the injection port
D Injecting the isotope and flushing.

6.1.1.4 Internal Dose to Workers. A limited amount of health physics data was acquired and used to evaluate whether or not normal aboveground operations of a radioisotope injection could result in internal doses to workers. Urinary bioassays for tritium of licensee workers who injected tritium were found to vary over a relatively narrow range in response to a very wide range in the amount of HTO injected (from a hundredth of a curie to tens of curies). The urinary tritium level was not correlated with the amount of HTO injected. Most urinary concentrations were from 1×10^{-3} to 1×10^{-2} μCi/L. The highest concentrations were about 3×10^{-2} μCi/L.

A possible explanation for these observations is that tritium release to the environment is not a normal occurrence when HTO is injected as an interwell tracer for EOR. As noted above, most concentrations varied within a factor of 10. Furthermore, the same narrow range of urinary concentrations was measured when widely varying amounts of ^{3}H-labelled elemental hydrogen or hydrocarbon were injected as a gaseous tracer. The measured concentrations may be attributable to the tritium background in the workers' laboratory. At the authors' laboratory, urinary bioassays in the range of 1×10^{-3} to 1×10^{-2} μCi/L are interpreted as evidence for the absence of an enhanced exposure to HTO in the working area (Myers and Dupzyk, 1983*). A concentration range an order of magnitude less would indicate exposure to background environmental levels.

Another explanation is that a fairly constant release and uptake of tritium accompanies each HTO injection project regardless of the amount of ^{3}H injected; the dose would then be independent of the amount of ^{3}H injected. In the incident described in Sec. 3.1.2 in which HTO was spilled on the ground, the urine samples from the four workers who were exposed to a spill of 3 to 5 mCi of HTO but were absent during the injection (5 Ci), measured 5.0×10^{-3} to 7.4×10^{-2} μCi/L. Using the relationship that 1.0×10^{-3} μCi/L in urine is equivalent to a committed dose equivalent of 7.14×10^{-3} mrem to the whole body (ICRP, 1968), the range of committed dose equivalents is 0.04 to 0.5 mrem. In this report we adopt the hypothesis that each injection of HTO is associated with a dose equivalent of 0.02 mrem. We also assume that each injection of tritium as elemental hydrogen or labelled hydrocarbon similarly leads to an internal dose equivalent of 0.02 mrem to workers. A licensee indeed confirms that trace quantities of tritiated water are invariably associated with other forms of tritium.

A few bioassay data were available on ^{14}C in urine from workers who injected ^{14}C-labelled CO_2 in two projects, which involved injections of 0.5 Ci and 0.1 Ci of activity. Urinary bioassays were in the range of 2.9×10^{-3} to 3.2×10^{-2} μCi/L for the former and $\leq 1.9 \times 10^{-3}$ to 3.5×10^{-3} μCi/L for the latter. In the metabolism of ^{14}C as bicarbonate-CO_2, a urinary ^{14}C concentration of 1.0 μCi/L at one day is equivalent to a committed dose equivalent of 0.16 mrem to the whole body (ICRP, 1968). We provisionally adopt the hypothesis that the dose equivalent to workers from the normal aboveground operations of $^{14}CO_2$ injection is proportional to the amount injected and that a 1-Ci injection of $^{14}CO_2$ is associated with a urine concentration of 6.0×10^{-3} μCi/L at one day and a committed dose equivalent of 1.0×10^{-3} mrem to the whole body.

Table 6.3 summarizes the internal dose estimates for the gaseous radioisotopes. The water soluble radioisotopes are considered not to be released in normal aboveground operations and therefore do not yield resultant internal doses to workers.

---

* D.S. Myers and I.A. Dupzyk, Lawrence Livermore National Laboratory, Livermore, Calif., private communication (1983).

Table 6.3 Estimation of internal dose to workers from normal aboveground operations.

Isotope	Physical form	Chemical form	Normalized dose equivalent in the whole body
H-3	Liquid	HTO	0.02 mrem per injection
H-3	Gas	HT, hydrocarbons	0.02 mrem per injection
C-14	Gas	$^{14}CO_2$	0.001 mrem per Ci injected

6.1.1.5 Dose to individual workers from a typical operation. Tables 6.1, 6.2, and 6.3 can be used to estimate the dose to workers from normal aboveground operations. We illustrate the procedure by evaluating a typical EOR operation using radioisotopes as interwell tracers. In this example the project consists of three separate injections: 10 curies of HTO, 2 mCi of ^{90}Sr, and 20 mCi of ^{60}Co. The calculations are straightforward and are summarized in Table 6.4. To simplify the reporting without introducing substantial error, the ratio of the dose equivalent (in rems) or of the absorbed dose (in rad) to the exposure (in R) is assumed to be 1.0 throughout this report except when the dose is delivered by an alpha-emitter, such as ^{222}Rn.

Table 6.4 Example of estimated dose to a worker from a typical EOR project with radioactive tracers.

Isotope	Physical or chemical form	Amount injected mCi	Normalized dose equivalent[a] mrem per injection	mrem per mCi injected	Total dose mrem
H-3	HTO	10,000	0.02[b]		0.02
Co-60	Liquid	20	2.6[c]	0.033[c]	3.3
Sr-90	Liquid	2	0[c]	0.007[c]	0.014
Total					3.3

[a] To simplify the reporting without introducing substantial error, the ratio of the dose equivalent in rem to the whole body and other organs to the exposure in R is assumed to be 1.0, except when the dose is delivered by an alpha-emitter, such as ^{222}Rn.

[b] Value from Table 6.3

[c] Value from Table 6.1.

In this illustration the total dose to a worker is estimated to be 3.3 mrem, which is below the sensitivity of a TLD badge. Assuming that two licensee workers are present at each operation, the collective dose equivalent to workers in this example is estimated to be 6.6×10^{-3} man-rem.

6.1.1.6 Collective dose to workers. Data from Tables 6.1, 6.2, and 6.3 can be used to estimate the collective dose to workers from normal aboveground operations. Table 6.5 summarizes calculations to estimate the annual collective dose equivalent in workers from normal aboveground operations. The results hinge on the total amount of radioisotopes injected (col. 2) and on the number of injections (col. 3) carried out annually. The amounts injected and the number of injections are based on information for the period 1979 to 1983 from two licensees. The estimates of the total amount injected annually were weighted toward the highest values and were scaled upward to account for the amounts injected by all companies that perform tracer injections for EOR projects. The total number of injection operations is 400, which is consistent with our basic assumption that 100 EOR projects using radioisotopes as interwell tracer are carried out annually and that each project on the average involves four separate injections of an isotope into a well (see Sec. 1.3.4).

Table 6.5 also lists the normalized dose equivalent expressed in mrem per injection (col. 4) and the normalized dose equivalent expressed in mrem/mCi injected (col. 5). The sixth and last column lists the total dose equivalent per nuclide, which is calculated as the sum of two products: the product of the number of injections and the dose equivalent per injection, and the product of the total activity injected and the dose equivalent per mCi injected.

The annual dose equivalent in a worker over all the injected radionuclides is 470 mrem/y. This value is multiplied by 2, to account for the two workers engaged per operation, to yield an annual collective dose equivalent in workers of 0.93 man-rem/y.

As noted previously, workers of oil companies remain at a distance of $\sim$10 meters or more from the wellhead and are not exposed. Members of the public would be at a much greater distance.

6.1.2 Accidental exposures

In this section we estimate doses to workers and members of the public that could result from the aboveground accident scenarios described in Sec. 3.1.2. As noted previously, incidents or accidents involving the unplanned release of radioisotopes when they are used as interwell tracers in EOR operations have been extremely rare.

6.1.2.1 Dose from a leaked source (Scenario A, Sec. 3.1.2). When a liquid source leaks from a damaged container during transit, the leak is readily detected upon its arrival at the injection site where the package is surveyed with a detector. The liquid itself does not drip outside the package but is absorbed in the absorbent packing material that surrounds the source. The worker is exposed for a short time to an unshielded source while he is repackaging the source for return shipment. The assumption that the worker is exposed to an unshielded source for 5 minutes at a distance of 1 foot leads to the normalized dose estimates of Table 6.6.

Table 6.5 Annual collective dose to workers from normal aboveground exposures.

Radioisotope	Total amount injected, Ci/y	Number of injections	Normalized dose equivalent[a,b] mrem per injection	Normalized dose equivalent[a,b] mrem per mCi injected	Dose equivalent per nuclide per worker[c], mrem/y
H-3 (gas)	1300	64	0.02		1.3
H-3 (HTO)	1800	128	0.02		2.6
C-14 ($^{14}CO_2$)	1.6	14		1×10^{-6}	0.002
C-14 (liquid)	0.10	8	0	0	0
Na-22	0.12	2	2.6	0.030	8.8
S-35	0.02	2	0	0	0
Ca-45	0.02	2	0	0	0
Co-57	2.0	24	2.6	0.0025	67
Co-58	0.80	20	2.6	0.014	63
Co-60	2.6	43	2.6	0.033	200
Ni-63	1.4	5	0	0	0
Zn-65	0.04	2	2.6	0.0078	5.5
Kr-85, activity <1 Ci	10	10		8.7×10^{-4}	8.7
Kr-85, activity ≥1 Ci	290	32	0.82	5×10^{-5}	41
Sr-85	0.12	5	2.6	0.016	15
Sr-90	0.40	22	0	0.0070	2.8
Ag-110m	0.04	2	2.6	0.037	6.7
I-125	1.0	10	2.6	0.0043	30
I-131	0.12	5	2.6	0.0055	14
Total					470 mrem/y
Collective dose equivalent in workers, assuming two licensee workers per project					0.93 man-rem/y

[a] Values from Table 6.1, Table 6.2, or Table 6.3.

[b] To simplify the reporting without introducing a substantial error, the ratio of the dose equivalent in rem to the whole body and individual organs to the exposure in R is assumed to be 1.0, except in the case of alpha-emitters, such as ^{222}Rn.

[c] A worker would receive these doses if he were to inject the total amount of the radioisotope.

To illustrate the use of Table 6.6, the normalized exposure from a container that leaks ^{60}Co according to the above scenario is 1.2 mR/mCi. The exposure from a 20 mCi ^{60}Co source that leaked would be

$$20 \times 1.2 = 24 \text{ mR}$$

The exposure to the hands would exceed this value.

Table 6.6 Normalized exposure to workers from handling a leaky source (scenario A, Sec. 3.1.2).[a]

Radioisotope	Normalized exposure mR per mCi	Radioisotope	Normalized exposure mR per mCi
H-3	0	Ni-63	0
C-14	0	Zn-65	0.29
Na-22	1.1	Sr-85	0.56
S-35	0	Sr-90	0.01
Ca-45	0	Ag-110m	1.4
Co-57	0.090	I-125	0.16
Co-58	0.50	I-131	0.20
Co-60	1.2		

[a] Assumes a 5-min exposure to an unshielded source; body at a distance of 12 in. (0.3 m).

This scenario could also result in a dose via ingestion. However, the worker would probably be wearing gloves and after noting the leakage would likely take extra precautions. Of the radioisotopes in Table 6.5, ^{90}Sr and ^{125}I are associated with the highest committed dose equivalents per unit ingestion of activity. (Ingestion dose factors are listed in Appendix C.)

The dose to a leaking HTO source will be discussed in Sec. 6.1.2.5.

6.1.2.2 Dose from a source spilled on the ground (Scenario B, Sec. 3.1.2). If a worker accidentally drops a glass vial containing a radioactive tracer in solution and it breaks upon striking the ground, the worker would be subject to exposure from the radioactivity spilled on the ground. The exposure would continue during the time the worker performs cleanup measures on the spilled activity. We estimate the resultant dose by assuming that the worker is exposed for 10 minutes to a source uniformly loading a 15-cm disk on the ground surface. The worker is assumed to be at a distance of 0.5 m from the center of the disk. Table 6.7 presents normalized exposures for this scenario, computed from the tables of exposure rates of Appendix A.2. The exposure rates are calculated for a height of 1 m.

To illustrate the use of Table 6.7, the normalized exposure from ^{60}Co spilled on the ground according to the above scenario is 0.17 mR/mCi. The exposure from a spillage of 20 mCi ^{60}Co would be

$$20 \times 0.17 = 3.4 \text{ mR}$$

This scenario could also result in radioactivity deposited on the skin with a resultant local dose to skin if liquid should splash on the legs. However, it is likely that clothing could intercept any splashed liquid before it would contact the skin.

Table 6.7 Normalized exposure to workers from radioactivity spilled on the ground (scenario B, Sec. 3.1.2).[a]

Radioisotope	Normalized exposure mR per mCi	Radioisotope	Normalized exposure mR per mCi
H-3	0	Ni-63	0
C-14	0	Zn-65	0.041
Na-22	0.16	Sr-85	0.077
S-35	0	Sr-90	0.001
Ca-45	0	Ag-110m	0.20
Co-57	0.013	I-125	0.022
Co-58	0.072	I-131	0.029
Co-60	0.17		

[a] Assumes a 10-min exposure to a uniformly loaded disk on the ground surface. Radius of disk equals 15 cm. Distance from center of disk equals 0.5 m. Exposure is calculated for a height of 1 m.

A glass vial containing a radioactive tracer would be handled with care. Only a small fraction of the vials that are handled would be dropped, and only a small fraction of those dropped would break. Thus, a licensee has estimated that about 1 of 50 glass vials that are dropped from a height of 4 ft onto a hard surface would break. If the radioactive source is confined in a vessel that is not subject to breakage, this scenario would not occur.

The dose due to spilled HTO will be discussed in Sec. 6.1.2.5.

6.1.2.3 Dose from an accidental release of a gaseous source (Scenario C, Sec. 3.1.2). A worker who accidentally drops a glass vial containing a gaseous radioisotope, causing the vial to break and release its contents to the surface air, could experience an exposure from inhalation and submersion in the gas. Similarly, members of the public at locations downwind could be exposed to the expanding cloud of gas. The doses to workers and the collective dose to members of the public were calculated from estimates of the IAC at the exposure locations. The IACs for the workers' dose estimates were calculated by an analytical Gaussian model; those for the collective doses to members of the public were calculated to a distance of 80 km using a particle-in-cell model. The methods are described in detail in Appendix B.

Although the hypothetical accident could be expected to occur during daytime working hours, the range of atmospheric stability and wind speeds is still large since all seasons of the year must be considered for a wide variety of locations. The three adopted scenarios are intended to yield results that reasonably bound the range of likely meteorological conditions (see Table 6.8).

Scenario 1 occurs most frequently during the afternoon in warm seasons. Scenario 2 is most likely in any high wind condition or when the sky is overcast. At low wind speeds, the stability will move away from neutral and be governed by the

Table 6.8 Meteorological scenarios.

Scenario	Stability class			Wind speed (m/s) Measured ht = 6 m	Wind speed (m/s) Extrapolated ht = 1.6 m
1	Unstable	A	low	1.5	1.4
2	Neutral	D	high	8.0	6.6
3	Stable	F	low	1.5	0.7

near-surface temperature gradient, i.e., stable when the surface is cooler than the overlying air, and unstable when the surface is warmer (often in the daytime). Scenario 3 covers most nighttime and early morning low wind-speed conditions and is the scenario that yields the highest concentrations near the surface after a surface release.

Table 6.9 (see Table B.2 in Appendix B) presents IACs near the surface 1 meter downwind from a surface release according to the three scenarios. The IAC values are zero except at a receptor height of 0.2 m. These estimates suggest that if the worker remains in place after dropping and breaking the vial, at most there would be a submersion dose to the lower extremities from ^{85}Kr or a dose from absorption of HTO through skin of the lower extremities. A worker located upwind or who moves immediately upwind of the release would not be exposed.

Table 6.10 (see Table B.1 in Appendix B) presents the maximum IACs versus downwind distance following a unit surface release. These estimates indicate that the worker could be exposed from inhalation and submersion if he moves downwind into the cloud or traverses the cloud. This could occur if the worker instinctively moved to a vehicle for protection. Again, if the worker moves upwind immediately after the accident, the exposure would be minimal or zero.

Table 6.9 Integrated air concentrations near the surface 1 meter downwind from a unit surface release (from Appendix B).[a]

Receptor ht m	1. Unstable IAC, s/m^3	2. Neutral IAC, s/m^3	3. Stable IAC, s/m^3
1.6	0	0	0
1.0	1.9×10^{-11}	0	0
0.2	3.2	0.75	3.4

[a] Thus, for a 1 μCi release, IAC would be expressed in μCi-s/m^3.

Table 6.10 Maximum integrated air concentrations near the surface downwind from a unit surface release (from Appendix B), for three meteorological conditions.

	1. Unstable			2. Neutral			3. Stable		
Receptor ht m	Max IAC s/m^3	Distance[a] m	Time[b] s	Max IAC s/m^3	Distance[a] m	Time[b] s	Max IAC s/m^3	Distance[a] m	Time[b] s
1.6	0.065	8	6	0.014	22	3	0.13	36	51
1.0	0.015	6	5	0.035	14	2	0.33	24	40
0.2	3.2	1	1	0.76	2	0.4	8.11	2	10

[a] Distance refers to the downwind distance along the centerline of cloud travel.

[b] Time refers to the time maximum IAC occurs at that distance and height.

To evaluate the consequence of this accident scenario, doses were estimated assuming that the worker is exposed to the cloud under unstable conditions at a distance of 6 m from the release. The IAC at nostril height (1.6 m) calculated by Eq. B-3 of Appendix B, is 4.0 x 10^{-2} Ci-s/m^3 per curie released. The resulting doses calculated using the dose-rate factors of Table B.3 (Appendix B) are presented in Table 6.11.

Table 6.11 Normalized dose equivalent in a worker from an accidental surface release of a gaseous radioisotope (scenario C, Sec. 3.1.2).[a]

Radioisotope	Exposure mode	Reference organ	Normalized dose equivalent rem/Ci released
H-3 HT, hydrocarbon	Submersion	Lung[b]	4.1 x 10^{-7}
HTO[c]	Inhalation	Whole body	8.4 x 10^{-4}
C-14 $^{14}CO_2$	Inhalation	Whole body	3.2 x 10^{-4}
Kr-85	Submersion	Skin	1.9 x 10^{-3}
	Submersion	Whole body	4.7 x 10^{-7}

[a] The worker is assumed to be exposed to the cloud downwind under unstable conditions at a distance of 6 m from the release where the IAC at nostril height (1.6 m) is 4.0 x 10^{-2} Ci-s/m^3 per Ci released (see Sec. 6.1.2.3).

[b] The dose to the whole body via submersion is zero.

[c] The estimate for HTO gas is used to evaluate the consequences of leaks or spills of liquid HTO.

Tritium as HTO is included in Table 6.11 because we have used this method to estimate the dose from HTO spilled on the ground. Estimation of the doses from a leaky HTO source and from HTO leaked or spilled from a wellhead is described in Sec. 6.1.2.5.

It should reassure the reader who is concerned that a worker may inhale ^{125}I or ^{131}I vapor following the accidental spilling of radioiodide that radioiodide solution sources are strongly alkaline (pH ∿12), which is very unfavorable for oxidation to free iodine. Other water-soluble tracers are also considered not to become airborne following an accidental spill.

The collective dose estimates out to 80 km for a 1-Ci release are shown in Table 6.12 for the "typical" site under the three representative meteorological conditions. (Table 6.12 is abstracted from Tables B.4, B.5, and B.6 in Appendix B; the calculations are described in Sec. B.2). As noted previously, the "typical" site impacts an urban area 15 km x 15 km with its center approximately 32 km (∿ 20 miles) downwind from the site of the release. The urban population density is 695/km^2 (1800/mi^2), and the rural population density is 13.5/km^2 (35/mi^2). The collective doses of Table 6.12 are upper-bound estimates for the "typical" site because the urban area would not be expected to be downwind from the site of release all the time since wind direction is variable. The collective doses for the "typical" site under stable conditions are used to assess the accidental release of gaseous radioisotopes as they are the highest for the three meteorological scenarios.

Table 6.12 Normalized collective dose equivalent in members of the public from an accidental surface release of a radioactive gas for the "typical" reference site.[a]

Radioisotope	Reference organ	Man-rem per Ci released Meteorological conditions Unstable	Neutral	Stable
H-3 HT, hydrocarbons	Lung	1.2×10^{-7}	3.5×10^{-8}	9.4×10^{-7}
HTO	Whole body	2.5×10^{-4}	7.1×10^{-5}	1.9×10^{-3}
C-14 $^{14}CO_2$	Whole body	9.3×10^{-5}	2.7×10^{-5}	7.3×10^{-4}
Kr-85	Skin	5.7×10^{-4}	1.6×10^{-4}	4.5×10^{-3}
	Whole body	7.2×10^{-6}	2.1×10^{-6}	5.6×10^{-5}

[a] The region is assumed to extend 80 km from the site of release. The calculations are described in Sec. B.2 of Appendix B (see Tables B.4, B.5, and B.6 in Appendix B.)

As noted in Sec. 3.1.2, accidents involving the unplanned release of radioisotopes to the environment in the course of interwell tracing with radioisotopes in EOR operations have been rare. One licensee states that a once-a-year occurrence of an accidental release of a gaseous radioisotope would be an overestimate and that the true frequency of this accident scenario is perhaps once in 5 or 10 years. In this report we have assumed conservatively that 0.01 is the probability that a gaseous radioisotope (or HTO) will be released because of an accidentally dropped and broken container. This means that 1% of the operations involving the injection of a gaseous radioisotope or HTO will be accompanied by the accidental release of the radioisotope. Since on an annual basis we have assumed 120 injections of gaseous radioisotopes and 128 injections of HTO (see Table 6.5), the frequency assumed for this accident scenario is about 2.5 per year, which overestimates the expected frequency.

The resultant collective dose estimates for members of the public from accidental releases of gaseous radioisotopes are summarized in Table 6.13. The total collective dose equivalent in the whole body is estimated to be 0.034 man-rem/year.

Table 6.13 Collective dose to members of the public from accidental surface releases of gaseous radioactive tracers for EOR (Scenario C, Sec. 3.1.2).[a]

Radioisotope	Total amount injected, Ci/y	Reference organ	Collective dose equivalent man-rem/y
H-3 HT, hydrocarbons	1300	Lung[b]	1.2×10^{-3}
HTO[c]	1800	Whole body	3.4
C-14 $^{14}CO_2$	1.6	Whole body	1.2×10^{-3}
Kr-85	300	Skin	1.4
		Whole body	0.017
Total		Whole body	3.4 man-rem/y
Total collective dose (P = 0.01)[d]		Whole body	0.034 man-rem/y

[a] Based on the normalized man-rem values of Table 6.12.

[b] The collective dose equivalent in the whole body via submersion is zero.

[c] The dose from the accidental release of HTO is assessed in the same manner as the dose from the accidental release of gaseous radioisotopes (see Sec. 6.1.2.5).

[d] The occurrence of accidents has been very rare. A conservative accident frequency of 0.01 has been used for dose estimation.

6.1.2.4 Dose from activity leaked from the wellhead (Scenarios D, E; Sec. 3.1.2). If a tracer solution were to leak through wellhead fittings (Scenario D of Sec. 3.1.2), through a faulty valve in the wellhead, or through a valve inadvertently left open (Scenario E of Sec. 3.1.2), a worker would be subject to exposure from radioisotopes deposited on the ground, on the surface of the wellhead, or on the skin. As noted previously, Scenario D is unlikely because the wellhead is checked for leaks before the tracer is injected.

For these scenarios, dose estimates to workers are made in the same manner as those for the scenario involving spillage of a source in solution (see Sec. 6.1.2.2). In Scenarios D and E, however, the radioisotope released would be more dilute. The worker is assumed to be exposed for 10 minutes to a source uniformly loading a disk on the ground. The radius of the disk is assumed to be 15 cm, and the distance from the center of the disk is assumed to be 0.5 m. Exposure rates were calculated for a height of 1 m (Appendix A).

Exposures for this combined scenario, normalized to an injection of 1 mCi, are presented in Table 6.14. These values assume that 5% of the source activity leaks from the wellhead to the ground. The normalized exposures of Table 6.14 are therefore 5% of the normalized exposures of Table 6.7, which are associated with the spillage of an undiluted solution.

The dose from HTO leaked from the wellhead is discussed in Sec. 6.1.2.5.

6.1.2.5 Dose from accidental releases of HTO. The dose from accidental releases of HTO was estimated by the method described in Sec. 6.1.2.3 for estimating the dose from an accidental release of a gaseous radioisotope.

Table 6.14 Normalized exposures to workers from radioactivity leaked from a wellhead (Scenarios D, E; Sec. 3.1.2).[a]

Radioisotope	Normalized exposure mR per mCi	Radioisotope	Normalized exposure mR per mCi
H-3	0	Ni-63	8.6×10^{-3}
C-14	0	Zn-65	2.0×10^{-3}
Na-22	7.8×10^{-3}	Sr-85	3.8×10^{-3}
S-35	0	Sr-90	7×10^{-5}
Ca-45	0	Ag-110m	9.8×10^{-3}
Co-57	6.3×10^{-4}	I-125	1.1×10^{-3}
Co-58	3.6×10^{-3}	I-131	1.4×10^{-3}
Co-60	8.6×10^{-3}		

[a] Dose for an injection of 1 mCi. 5% of the activity (i.e., 0.05 mCi) is assumed to leak from the wellhead and uniformly load a circular disk of radius equal to 15 cm on the ground surface.

to estimate the dose from a source that leaks HTO (Scenario A, Sec. 3.1.2), we assume that the worker is exposed to 10% of the source activity. The IAC to which the worker is exposed at nostril height is estimated as the IAC at a distance of 2 m and a receptor height of 1.0 m for unstable conditions. A 1-m height was chosen because it approximates nostril height to a source package on a tailgate. The IAC calculated by Eq. B-3 of Appendix B is 1.3×10^{-3} Ci-s/m^3 per curie released, which leads to a normalized dose equivalent of 0.027 mrem in the whole body from inhalation per Ci released when combined with the dose-rate factor from Table B.3 (Appendix B). Since the worker is assumed to be exposed to 10% of the original source, the normalized dose equivalent is 0.0027 mrem per Ci of the original source.

As noted in Sec. 6.1.2.3, the dose from HTO that spills on the ground from a broken container is estimated using the appropriate normalized dose equivalent from Table 6.11, 8.4×10^{-4} rem per Ci released.

To estimate the dose from HTO that leaks or spills to the ground from the wellhead (Scenarios D and E, Sec. 3.1.2), we assume that 5% of the source activity leaks. We correct the normalized dose in Table 6.11 by a factor of 0.05, which leads to a normalized dose equivalent in the whole body of 4.2×10^{-5} rem per Ci injected.

6.1.2.6 Combined dose to workers. The annual dose to workers from the accident scenarios is estimated as follows: The dose from scenario A is given by

$$D_A = \Sigma P_A A_i E'_{Ai} = P_A \Sigma A_i E'_{Ai}$$

where

D_A is the annual dose to a worker from scenario A (mrem/y),
A_i is the total amount of the ith nuclide that is injected annually (mCi/y),
E_{Ai} is the normalized dose from the ith nuclide by scenario A (mrem/mCi injected)
P_A is the probability that scenario A will occur (unitless, $0 \leq P_A \leq 1.0$)

It is assumed that P_A is independent of the isotope injected.

The dose from each of the other scenarios is estimated in the same manner and the total dose from all the scenarios, D, in mrem/y, is then calculated as the sum of the doses from each of the scenarios.

$$D = P_A \Sigma A_i E'_{Ai} + P_B \Sigma A_i E'_{Bi} + P_C \Sigma A_i E'_{Ci} + P_D \Sigma A_i E'_{Di} + P_E \Sigma A_i E'_{Ei}$$

As noted in Sec. 3.1.2, accidents involving the release of radioisotopes to the environment while radioisotopes are used as interwell tracers in EOR operations have been very rare. The frequency of accidents has been estimated to be one per year or less. In the absence of statistics on the incidence of accidents, we have assumed very conservatively for dose estimation that 0.05 is the probability of an accident releasing radioactivity to the environment during the aboveground stage of operations. Thus, in calculating the total dose from all the scenarios an accident is assumed to occur on 5% of the injections. In addition, the five scenarios are assumed to be equally probable and to occur independently, and the probability of an accident is asumed to be independent of the radioactive tracer. These assumptions greatly simplify the calculations, and the resultant dose estimates for accidents can readily be compared with those for normal exposures and regarded in proper perspective.

Thus,

$$P_A = P_B = P_C = P_D = P_E = 0.01$$

Since it is assumed that 100 EOR projects using radioactive tracers are conducted annually and that each project on the average consists of four tracer injections, an average of 20 aboveground accidents are assumed to occur each year.

Table 6.15 summarizes the calculations for the collective dose to workers from accidental aboveground exposures. If accidental exposures were a common occurrence, the average collective dose equivalent in a worker per year is estimated to be 6.4 man-rem/y. Correcting for the probabilities of an accident and allowing for two workers per project leads to a collective dose equivalent of 0.130 man-rem. The calculated collective dose equivalent for accidents is about 14% of the 0.93 man-rem to workers from normal operations (see Table 6.5). Since we purposely overestimated the dosage from accidents, the average collective dose equivalent in workers is actually much less.

It must be remembered that the dose estimate for accidents is strictly an abstraction that can never actually be realized. Within the framework of our approach, the dose from an actual accident would depend upon the radioisotope tracer, the amount used, and the occurrence of a specific scenario. The collective dose to workers would vary from zero, for the occurrence of any of the scenarios involving ^{35}S, ^{45}Ca, or ^{63}Ni, to 0.72 man-rem for accident scenario A during an attempt to inject 300 mCi of ^{60}Co.

6.2 SUBSURFACE STAGE

6.2.1 Normal exposures

In view of the absence of exposures to workers and members of the public, the dose estimates for normal subsurface operations are zero.

6.2.2 Accidental exposures

Although the accidental contamination of a freshwater aquifer by radioactive tracers used in EOR operations is unlikely, it is essential to evaluate the consequences of such an occurrence. A hydrological advection-dispersion model in three dimensions was used to estimate the concentration at various downgradient locations along the centerline following a pulse input of unit activity into the aquifer underlying the reference site. The maximum annual dose via ingestion of water drawn from the aquifer at these locations was calculated (see Table C.2 in Appendix C). The maximum doses per μCi released for the 1.0-km distance were selected as the normalized dose factors to evaluate the dose. Leakage of the total source activity into the aquifer is clearly unrealistic. We therefore assumed that one-tenth of the injected activity leaks into the aquifer.

Table 6.15 Annual collective dose equivalent in the whole body of workers from accidental aboveground exposures.

Radioisotope	Total amount injected, Ci	Number of injections	Annual dose equivalent, mrem/y[a]					
			Scenario A[b]	Scenario B[c]	Scenario C[d]	Scenario D[e]	Scenario E[e]	Total[f]
H-3 gas	1300	64	0	0	0	0	0	0
H-3 HTO	1800	128	4.9[g]	1500[g]		76[g]	76[g]	1700
C-14 $^{14}CO_2$	1.6	14	0	0	0.51	0	0	0.51
C-14 (liquid)	0.10	8	0	0		0	0	0
Na-22	0.12	2	130	19		0.94	0.94	150
S-35	0.02	2	0	0		0	0	0
Ca-45	0.02	2	0	0		0	0	0
Co-57	2.0	24	180	26		1.3	1.3	210
Co-58	0.80	20	400	58		2.9	2.9	460
Co-60	7.6	43	3100	440		22	22	3600
Ni-63	1.4	5	0	0		0	0	0
Zn-65	0.04	2	12	1.6		0.08	0.08	14
Kr-85	300	42	0	0	0.14	0	0	0.14
Sr-85	0.12	5	67	9.2		0.46	0.46	77
Sr-90	0.40	22	4	0.4		0.03	0.03	4.5
Ag-110m	0.04	2	56	8.0		0.39	0.39	65
I-125	1.0	10	160	22		1.1	1.1	180
I-131	0.12	5	24	3.5		0.17	0.17	28
Total[h]			4100	2100	0.65	110	110	6400 mrem/y
Collective dose equivalent in workers assuming two workers per injection[i]								0.130 man-rem/y

[a] A worker would receive these doses if the accident scenario occurred while he handled the total amount of the radioisotope.

[b] Calculated from the activities in col. 2 and the normalized doses in Table 6.6

[c] Calculated from the activities in col. 2 and the normalized doses in Table 6.7.

[d] Calculated from the activities in col. 2 and the normalized doses in Table 6.11.

[e] Calculated from the activities in col. 2 and the normalized doses in Table 6.14.

[f] Total from all scenarios.

[g] See Sec. 6.1.2.5 for estimation of the dose from HTO.

[h] Assumes $P_A = P_B = P_C = P_D = P_E = 1.0$.

[i] Assumes $P_A = P_B = P_C = P_D = P_E = 0.01$. The occurrence of accidents has been very rare. A conservative accident frequency of 0.01 has been used for dose estimation.

Table 6.16 is a summary of the resultant dose estimates. The estimates are presented for the reference organ that receives the highest dose. If leakage of radioactivity into the aquifer were a regular occurrence, the committed dose equivalent in an individual from drinking of water for one year would be 7.6 mrem in bone surface and 0.43 mrem in the lower large intestine. It should be remembered that the dose estimates in Table 6.16 are strongly biased toward high values because of the unrealistic assumption that the water-supply well is located on the centerline. Furthermore, the assumed release fraction of 0.1 is still conservative. The doses would decrease as the lateral distance of the well from the centerline increases.

Summing all the doses in the fourth column of Table 6.16 is not a valid procedure because the reference organs differ. The fifth column lists dose estimates to the whole body for the radionuclides found to be most important, ^{3}H (HTO), ^{14}C, ^{22}Na, ^{60}Co, ^{63}Ni, and ^{90}Sr. Strontium-90 is singled out as potentially the most important radionuclide. The sum of the doses to the whole body is 1.2 mrem/y.

A rough estimate of the incidence of groundwater pollution via well failure can be obtained from the number of complaints of groundwater contamination filed with the Railroad Commission of Texas (RCT). During 1973, 1974, and the first half of 1975, three complaints of groundwater contamination by brines were filed in District III of the RCT (Braxton et al., 1976). Because brine disposal in Texas since 1973 has been permitted only into impervious subsurface pits, these complaints are attributable to well leaks rather than seeps. From information on regional groundwater hydrology and well locations, it was estimated that fluids from 40 percent of the oil wells that are breached could reach a water well within five years. The incidence rate for well failures in District III is thus

$$\frac{3}{2.5 \text{ yr} \times (0.4)} = 3 \text{ per year}$$

Taking into account the producing wells in the area, which number some 8,000, the incidence of well failure is

$$3 \text{ per year}/8000 = 4 \times 10^{-4}/\text{year}$$

The relatively low incidence of well failure leading to the contamination of an underground water supply is in accord with current observations. It is estimated that in the past five years in Texas no more than 10 documented cases of aquifer contamination by brines are directly attributed to faulty injection or disposal wells; this is out of a total of 46,000 wells (Ginn, 1983*). In this report we have assumed conservatively for dose estimation that the probability associated with this accident scenario is 0.01, i.e., an average of 1% of the injections of a liquid radioisotope tracer will result in the contamination of a freshwater aquifer by radioisotopes and brine.

The collective dose to members of the public was estimated assuming that the well serves a 1-km^2 area about the well and that the population density is 14/km^2. The collective dose is then

* R. Ginn, Oil and Gas Division, Railroad Commission of Texas, Austin, Tex., private communication (1983).

Table 6.16 Annual dose to an individual via ingestion of water from an aquifer contaminated by radioactive tracers for EOR.[a]

Radioisotope	Total amount injected, Ci/y	Reference organ[b]	Dose equivalent[c] mrem/y	Dose equivalent in whole body[c] mrem/y
H-3 (HTO)	1300	Whole body	0.68	0.68
C-14	0.10	Whole body	3.3×10^{-3}	3.3×10^{-3}
Na-22	0.12	Bone surface	1.8×10^{-3}	1.2×10^{-3}[d]
S-35	0.02	Whole body	9×10^{-13}	
Ca-45	0.02	Bone surface	1×10^{-8}	
Co-57	2.0	LLI wall	1.6×10^{-5}	
Co-58	0.80	LLI wall	1×10^{-11}	
Co-60	2.6	LLI wall	0.36	0.039[d]
Ni-63	1.4	LLI wall	0.070	3.7×10^{-3}[d]
Zn-65	0.04	LLI wall	6×10^{-7}	
Sr-85	0.12	LLI wall	1×10^{-13}	
Sr-90	0.40	Bone surface	7.6	0.47[d]
Ag-110m	0.04	Gonads	2×10^{-6}	
I-125	1.0	Thyroid	5×10^{-11}	
I-131	0.12	Thyroid	e	
Total (P = 1.0)				1.2 mrem/y
(P = 0.01)				0.012 mrem/y
Collective dose equivalent in members of the public assuming P = 0.01[f]				1.7×10^{-4} man-rem/y

[a] The accident scenario assumes that 0.1 of the amount injected gains entry into the aquifer. The well is located 1.0 km downgradient from the source along the centerline.

[b] See Table C.1, Appendix C.

[c] Based on the normalized dose equivalent estimates of Table C.2, Appendix C. A person would receive these doses if the accident scenario occurred while the total amount of the radioisotope was being injected.

[d] Based on whole body dose factors (committed dose equivalents) of 1.4×10^{-2} rem/μCi for ^{22}Na (Dunning et al., 1981), 4.4×10^{-3} rem/μCi for ^{60}Co (Dunning et al., 1981), 1.8×10^{-4} rem/μCi for ^{63}Ni (ICRP, 1982), and 9.5×10^{-2} rem/μCi for ^{90}Sr (Dunning et al., 1981).

[e] The dose equivalent from ^{131}I would be very low ($<10^{-20}$ mrem/y).

[f] Assumes the well serves a 1 km^2 rural area where the population density is 14/km^2. The occurrence of accidents has been very rare. A conservative accident frequency of 0.01 has been used for dose estimation.

$$14 \times 0.01 \times 1.2 = 0.17 \text{ man-mrem/y}$$
$$= 1.7 \times 10^{-4} \text{ man-rem/y}$$

The dose from eating vegetables that had been irrigated with the well water or ingesting milk or meat from cattle that had drunk the well water and consumed forage that had been irrigated by the well water would be only a small fraction of that from drinking well water (see Appendix C).

6.2.3 Comparison with dose estimates for naturally occurring radioisotopes in water

The above estimate of dose appears to be small in comparison to that received from the use of groundwater containing naturally occurring radionuclides. As an example, the radioactive noble gas, ^{222}Rn, that results from the decay of the ^{238}U series, frequently occurs at elevated levels in groundwater. Table 6.17 contains data from UNSCEAR (1982) on the occurrence of ^{222}Rn in groundwater.

As a specific example, Prichard and Gesell (1981) have calculated the individual and collective dose equivalents received by residents in Houston and twelve nearby suburban municipalities from ^{222}Rn in their water. The sources of water are about evenly divided between wells and surface sources of much lower ^{222}Rn content.

Table 6.17 Radon concentration in well water from several areas of the world (UNSCEAR, 1982).

Location	Radon concentration (kBq/m^3)[a] Average	Maximum
Austria		
Salzburg	1.5	7
Finland		
Helsinki and Vantaa	1200	
Other areas	280	45,000
Italy	80	
Sweden	19	150
United States		
Aroostock, Maine	48	200
Cumberland, Maine	1000	5800
Hancock, Maine	1400	4600
Lincoln, Maine	560	1600
Penobscut, Maine	540	2400
Waldo, Maine	1100	3100
York, Maine	670	2200
North Carolina	100	1700

[a] 1 kBq/m^3 = 27 pCi/L.

Dose is delivered to the gastric epithelium by ingestion and to the bronchial epithelium by inhalation of radon liberated during water use, such as showering. For the 1,612,000 residents in 1975, the average level of ^{222}Rn in water was measured and calculated to be 437 pCi/L. This results in a calculated average individual exposure to the bronchial epithelium of 0.0024 Working Level Months (WLM) per year and an average individual dose equivalent of between 14 and 34 mrem/y, depending on the conversion factor used. The average individual dose equivalent in gastric epithelium was calculated to be 2.6 mrem/y. For the Houston metropolitan area, the annual collective dose is 4200 man-rem in the gastric epithelium and between 20,000 and 60,000 man-rem in the bronchial epithelium. The individual dose equivalents from natural ^{222}Rn in the water supply (14-34 mrem/y in bronchial epithelium and 2.6 mrem/y in gastric epithelium), greatly exceed the dose equivalent from drinking well water contaminated by radioactive tracers used in EOR operations, estimated to average 0.012 mrem/y in the whole body of an individual.

The main concern, if 0.1 or any other fraction of the injected isotope were to gain entry into a freshwater aquifer, would be the entry of the accompanying salts and other constituents of the injection fluids into the aquifer (Todd and McNulty, 1976).

6.3 RECOVERY AND DISPOSAL STAGE

6.3.1 Normal exposures

Gaseous tritium injected for interwell tracing will be found in the natural gas fraction collected at production wells whether injected as HTO, HT or gaseous hydrocarbons. The HTO and $^{14}CO_2$ are removed when the gas is processed into dry CO_2-free gas. The separated CO_2 is either recovered and reinjected or vented to the atmosphere. The HT may be separated from the hydrocarbons along with N_2. Tritiated ethane and propane are separated into a liquid natural gas fraction and noble gases remain in the gas fraction.

Because the radioisotope concentrations in the fluids collected at production wells are below exempt concentrations, and since there is practically no direct or indirect contact of workers with materials collected, doses to workers are assumed to be negligible.

Doses to members of the public are estimated on the basis of exposures to the combustion products of the tracer-bearing gas after it is processed and distributed to consumers. The background information and data are described in Appendix D.

The annual dose equivalent in the whole body of an individual from tritium per unit concentration in the gas supply varies over a narrow range, and is estimated to be in the range 2×10^{-4} to 4×10^{-4} mrem/y per pCi/L (Tables D.1 and D.2). If the average concentration of ^{3}H in gas were 20 pCi/L, which is 10% of the limiting value for release to an unrestricted area (10 CFR 20 Appendix B Table II. Col. 1), and the dose equivalent rate per unit concentration in gas were 3×10^{-4} mrem/y per pCi/L, the dose equivalent in an individual would be 6×10^{-3} mrem/y. If the ^{3}H concentration in gas were 200 pCi/L, the limiting value, the dose equivalent in an individual would be 6×10^{-2} mrem. The dose estimate of 6×10^{-2} mrem/y is too high because the maximum design concentration would probably be less than the limiting value and the average concentration over a period of time is much less than the maximum. In addition the tracer-bearing gas from one production well in a reservoir would be mixed with non-tracer-bearing gas from other production wells in

the reservoir and from production wells that feed into the distribution system from other reservoirs. Therefore the annual dose to an individual is estimated to be $\leq 6 \times 10^{-3}$ mrem/y.

The dose equivalent rate in skin per unit concentration of ^{85}Kr in gas is estimated to be 4×10^{-4} mrem/y per pCi/L (Table D.1). The ratio of the dose equivalent rate per unit concentration for ^{85}Kr and that for ^{3}H is consistent with the ratio of the dose factors for the two isotopes (see Table B.3). If the concentration of ^{85}Kr in gas were 30 pCi/L, which is 10% of the limiting value for release to an unrestricted area, the dose equivalent in the skin of an exposed individual would be 1.2×10^{-2} mrem. The dose equivalent in the whole body would be about 80 times less.

The Los Angeles basin is used as a frame of reference to estimate the collective dose to members of the public. The annual rate of gas consumption in the Los Angeles basin from data provided by Jacobs et al. (1972b) was

Residential	1.49×10^{10} m^3/y
Industrial	1.38×10^{10} m^3/y
Steam plants	0.73×10^{10} m^3/y
Total	3.60×10^{10} m^3/y

The total quantity of ^{3}H gas injected annually for interwell tracing in EOR is estimated to be 1300 Ci (Table 6.5). We assume that 20% of the ^{3}H activity is recovered at production well sites and is contained in a volume of processed gas equal to the annual gas consumption of the Los Angeles basin. The resulting concentration of ^{3}H in the gas is:

$$(0.20 \times 1300 \text{ Ci})/3.6 \times 10^{10} \text{ m}^3 = 7.2 \times 10^{-9} \text{ Ci/m}^3$$
$$= 7.2 \times 10^{-3} \text{ pCi/cm}^3$$

The average dose equivalent in an individual in the Los Angeles basin would be

$$7.2 \times 10^{-3} \text{ pCi/cm}^3 \times 0.29 \text{ mrem/y per pCi/cm}^3 = 2.1 \times 10^{-3} \text{ mrem/y}$$

(See Table D.2 in Appendix D for the dose equivalent rate per unit concentration in the Los Angeles basin.)

The total population of the city and country of Los Angeles is ∿7,000,000, which leads to a collective dose equivalent of

$$7 \times 10^{6} \times 2.1 \times 10^{-3} \text{ mrem/y} = 1.5 \times 10^{4} \text{ man-mrem/y}$$
$$= 15 \text{ man-rem/y}$$

The resulting collective dose equivalent from ^{3}H per curie in gas for the Los Angeles basin is

$$15 \text{ man-rem/y} \div 260 \text{ Ci} = 5.7 \times 10^{-2} \text{ man-rem/Ci}\cdot\text{y}$$

The estimate of the collective dose from ^{3}H to the regional population that would be served by gas from Rulison is a smaller value, 1.3×10^{-2} man-rem/Ci•y (Table D.1), which considers only the combustion of residential gas. The collective dose estimate for the Los Angeles basin takes into account industrial and steam generating sources as well as residential sources.

The same approach for the Los Angeles basin applied to ^{85}Kr leads to:

Quantity injected annually	300 Ci
Activity in processed gas (20%)	60 Ci
Concentration in gas	1.7×10^{-3} pCi/cm^3
Dose equivalent rate in skin per unit concentration	0.4 mrem/y per pCi/cm^3
Average dose equivalent in skin	6.8×10^{-4} mrem/y
Population of Los Angeles basin	7,000,000
Collective dose (skin)	4.8 man-rem/y
Collective dose (whole body)	0.061 man-rem/y

The dose to the whole body is about 80 times less than that to skin.

The above individual and collective dose estimates can be compared to those arising from the presence of ^{222}Rn in natural gas. According to Johnson et al. (1973), the concentration of ^{222}Rn in natural gas distribution lines varies from 0.5 to 100 pCi/L. For the average concentration of 20 pCi/L, they calculate a yearly dose equivalent of 15 mrem in the bronchial epithelium of an individual occupying a home with an unvented kitchen range. To compare doses in the Los Angeles Basin, we note that the average ^{222}Rn concentration on the West Coast is 15 pCi/L (Johnson et al., 1973), and assume that half of the 7,000,000 people live in homes with unvented kitchen ranges. The average individual dose equivalent in the bronchial epithelium would then be about 10 mrem/y and the collective dose equivalent would be about 40,000 man-rem/y. These estimates contrast with the much lower estimates attributable to residual tritium, which would deliver whole body dose equivalents of 2.1×10^{-3} mrem/y to an individual and 15 man-rem/y to the population.

In this report it is assumed that all injected CO_2 is recovered as a gas and after separation from the natural gas, is vented to the atmosphere. The collective dose equivalent from $^{14}CO_2$ is assumed to be the same collective dose that would result from the accidental release of $^{14}CO_2$, which is listed in Table 6.13 as 1.2×10^{-3} man-rem/y. Since $^{14}CO_2$ is miscible with fluids, this estimate is conservative.

In the case of HTO, we assume that 10% of the HTO recovered in fluids and gas at production well sites will evaporate. From the estimates of the collective dose to members of the public from accidental releases of HTO in Table 6.13, the annual release of 180 curies of HTO would lead to a collective dose equivalent of 0.34 man-rem/y.

The values in Table 6.18 are used in this report as the annual collective dose equivalent in members of the public from residual tracer activities in natural gas and other atmospheric releases of radioactive tracers recovered in EOR operations.

6.3.2 Accidental exposures

Accidental exposures would not be expected in the recovery and disposal stage.

Table 6.18 Collective dose to members of the public from residual tracer activity in natural gas and other atmospheric releases of radioisotopes recovered in EOR operations.

Radioisotope	Reference organ	Annual collective dose equivalent man-rem/y
^{3}H (gas)	Whole body	15
^{3}H (HTO)[a]	Whole body	0.34
^{14}C ($^{14}CO_2$)[b]	Whole body	0.0012
^{85}Kr	Skin	4.8
	Whole body	0.06
Total	Whole body	15 man-rem/y

[a] 10% of the HTO injected is assumed to be recovered at production well sites and evaporate.

[b] All of the $^{14}CO_2$ injected is assumed to be recovered at production well sites and vented to the atmosphere.

7. TOTAL ENVIRONMENTAL IMPACT FROM THE USE OF RADIOISOTOPES AS INTERWELL TRACERS

In this section we combine the dose estimates of Sec. 6, for workers and members of the public, to estimate the total national radiological impact from the use of radioisotopes as interwell tracers in EOR operations.

Table 7.1 is a summary of the dose estimates for workers. Licensee employees are exposed only during the aboveground stage of an operation. As noted in Sec. 6.1.1.6, the collective dose to workers is estimated by applying a factor of 2 to the dose estimates for an individual worker to account for the average of two workers assumed to participate in each project.

The collective dose equivalent in workers from normal exposures is estimated to be 0.93 man-rem/y. The external dose from ^{60}Co and other isotopes of Co accounts for a major fraction of this dose. Taking into account TLD data for licensee workers (reviewed in Sec. 6.1.1.1) and the fact that in some EOR operations the radioactive tracer is continuously shielded, the dose estimate for normal exposures is considered reasonable.

The average collective dose equivalent in workers from accidental exposures is estimated to be 0.13 man-rem/y, which is a factor of 7 lower than the collective dose equivalent to workers from normal exposures. It is important to remember that the actual dose associated with an accident could be expected to vary widely, depending on the radioisotope, the quantity handled and the particular accident scenario. The major contributor to the dose from accidents is ^{60}Co.

As noted previously, accidents leading to unplanned exposures have been extremely rare or nonexistent when radioisotopes were used as interwell tracers in EOR, and we regard the assumed accident frequency of 1% to be conservative. However, the dose estimates for accident scenarios are highly sensitive to parameter values assumed in the calculation. Thus, a very wide range of exposures and dose estimates would be possible for a given accident scenario. The total dose to an individual from an actual accident could be a substantial portion of the values listed in column 5 of Table 7.1 and may be comparable to the dose from normal exposures. The total collective dose equivalent in workers is 1.1 man-rem/y.

The dose to workers' skin has not been discussed. Personnel monitoring records of one licensee indicate that the average external exposure to the hands is two times the average external exposure to the whole body.

Table 7.2 is a summary of the dose estimates for members of the public. The accidental aboveground exposures are due to accidental surface releases of gaseous radionuclides (see Sec. 6.1.2.3). The accidental subsurface exposures are due to drinking water from an aquifer contaminated by radioactive tracers (see Sec. 6.2.2). The normal exposures associated with recovery and disposal are due to combustion products of the ^{3}H-labelled hydrocarbons recovered in the natural gas and to atmospheric releases of gaseous tracers from the fluids recovered at production wells (see Sec. 6.3.1).

The total collective dose equivalent in the whole body of members of the public is estimated to be 15 man-rem/y, derived mainly from HTO formed by the combustion of ^{3}H-labelled gaseous hydrocarbons recovered in natural gas. This dose estimate is exceeded by the collective dose equivalent in members of the public from the ^{222}Rn naturally occurring in natural gas. This is estimated to be 40,000 man-rem

Table 7.1 Estimates of the dose equivalent in workers from the use of radionuclides as interwell tracers in EOR.

Radioisotope	Total amount injected[b] Ci/y	Normal aboveground exposures[a] Number of injections	Dose equivalent to an individual[b] mrem/y	Collective dose equivalent[b] 10^{-3} man-rem/y	Accidental aboveground exposures[a] Dose equivalent to an individual[c] mrem/y	Assumed probability of occurrence[d]	Collective dose equivalent[e] 10^{-3} man-rem/y
H-3 (gas)	1300	64	1.3	2.6	0	0.01	0
(HTO)	1800	128	2.6	5.2	1700	0.01	34
C-14 ($^{14}CO_2$)	1.6	14	0.002	0.004	0.51	0.01	0.01
C-14 (liquid)	0.10	8	0	0	0	0.01	0
Na-22	0.12	2	8.8	18	150	0.01	3.0
S-35	0.02	2	0	0	0	0.01	0
Ca-45	0.02	2	0	0	0	0.01	0
Co-57	2.0	24	67	130	210	0.01	4.2
Co-58	0.80	20	63	130	460	0.01	9.2
Co-60	2.6	43	200	400	3600	0.01	72
Ni-63	1.4	5	0	0	0	0.01	0
Zn-65	0.04	2	5.5	11	14	0.01	0.28
Kr-85	300	42	50	100	0.14	0.01	0.003
Sr-85	0.12	5	15	30	77	0.01	1.5
Sr-90	0.40	22	2.8	5.6	4.5	0.01	0.09
Ag-110m	0.04	2	6.7	13	65	0.01	1.3
I-125	1.0	10	30	60	180	0.01	3.6
I-131	0.12	5	14	28	28	0.01	0.56
Total			470	930	6500		130

Total collective dose equivalent in workers: 0.93 + 0.13 = 1.1 man-rem/y.

[a] Workers are exposed only in the aboveground stage.

[b] From Table 6.5. A worker would receive these individual doses (and two workers these collective doses) if he were to inject the total amount of the radioisotope.

[c] From Table 6.15. A worker would receive these individual doses if accidents occurred while he were handling the total amount of the radioisotope.

[d] The occurrence of accidents has been very rare. A conservative accident frequency of 0.01 has been used for dose estimation.

[e] Collective dose estimate assuming accident frequency of 0.01. It is assumed that each project involves two workers.

Table 7.2 Estimates of dose equivalent in the whole body of members of the public from the use of radionuclides as interwell tracers in EOR.

Radioisotope	Total amount injected Ci/y	Accidental aboveground exposures: Collective dose equivalent[a] (P=1.0) man-rem/y	Probability[b]	Collective dose equivalent 10^{-3} man-rem/y	Accidental subsurface exposures: Dose equivalent in individual[c] mrem/y	Probability[b]	Collective dose equivalent[d] 10^{-3} man-rem/y	Normal recovery Collective dose equivalent[e] 10^{-3} man-rem/y
H-3 (gas)	1300	0	0.01	0	0	0.01	0	15000[f]
(HTO)	1800	3.4	0.01	34	0.68	0.01	0.095	340
C-14 ($^{14}CO_2$)	1.6	1.2×10^{-3}	0.01	0.012	0	0.01	0	1.2
C-14 (liquid)	0.10				3.3×10^{-3}	0.01	4.6×10^{-4}	
Na-22	0.12				1.2×10^{-3}	0.01	1.7×10^{-4}	
S-35	0.02				$<10^{-12}$	0.01	~0	
Ca-45	0.02				$<10^{-8}$	0.01	~0	
Co-57	2.0				$<10^{-5}$	0.01	~0	
Co-58	0.80				$<10^{-11}$	0.01	~0	
Co-60	2.6				0.039	0.01	5.5×10^{-3}	
Ni-63	1.4				3.7×10^{-3}	0.01	5.2×10^{-4}	
Zn-65	0.04				$<10^{-6}$	0.01	~0	
Kr-85	300	0.017	0.01	0.17	0	0.01	0	60
Sr-85	0.12				$<10^{-13}$	0.01	~0	
Sr-90	0.40				0.47	0.01	0.066	
Ag-110m	0.04				$\sim 10^{-6}$	0.01	~0	
I-125	1.0				$<10^{-12}$	0.01	~0	
I-131	0.12				~0	0.01	~0	
Total		3.4		34	1.2		0.17	15000

Total collective dose in members of the public: $3.4 \times 10^{-2} + 1.7 \times 10^{-4} + 15 = 15$ man-rem/y

[a] Collective dose equivalent in the whole body from Table 6.13

[b] The occurrence of accidents has been very rare. A conservative accident frequency of 0.01 has been used for dose estimation.

[c] Dose equivalent in the whole body from Table 6.16. A person would receive these doses if the accident scenario occurred while the total amount of the radioisotope was being injected.

[d] The collective dose equivalent assumes that the well provides drinking water to a 1-km^2 area that has a population density of 14/km^2.

[e] Collective dose equivalent to the whole body from Table 6.18.

[f] The annual dose equivalent in an individual is estimated to be $\leq 6 \times 10^{-3}$ mrem/y.

in the bronchial epithelium of this population. The reader will note that the collective effective dose equivalent in this population from natural sources of radiation is about 10^6 man-rem (UNSCEAR, 1982). (The effective dose equivalent is the dose equivalent in the whole body that is associated with the same total risk as that resulting from the dose equivalents in individual organs.) The collective dose equivalent from the ^3H in HTO evaporated from the fluids recovered in EOR is estimated to be 0.34 man-rem/y.

The collective dose equivalent from the accidental surface release of gaseous radioisotopes is estimated to be 0.034 man-rem/y, which is more than two orders of magnitude less than the 15 man-rem/y collective dose equivalent from the combustion of natural gas.

The accidental subsurface exposures are due to drinking water from an aquifer contaminated by radioactive tracers (see Sec. 6.2.2). Based on our approach, the total dose equivalent in a member of the public from this accident scenario could be as high as 1.2 mrem/y if the accident frequency were 100%. A dose of this magnitude is less than 1% of the hypothetical normal dose to a worker who handles all the radioisotope injected in a year (Table 7.1). The collective dose equivalent of 1.7×10^{-4} man-rem, taking into account an assumed frequency of accidental leaks into the aquifer and the number of persons who would be exposed, is about two orders of magnitude less than the collective dose equivalent from accidental aboveground exposures. It contributes less than 0.01% of the total collective dose to members of the public. The scenario involving the contamination of an aquifer is thus associated with a negligible radiological impact even after allowing for a much higher frequency than the 1% value assumed. A more pressing concern than the entry of a radionuclide into the aquifer would be the entry of salts, acids, alkali, polymer, surfactants, or other substances with the potential to contaminate a water supply. Furthermore, the estimated dose to an individual from the liberation of naturally occurring ^{222}Rn from a water supply derived from groundwater is about 24 mrem/y in bronchial epithelium and 2.6 mrem/y in gastric epithelium (see Sec. 6.2.3).

A summation of the collective doses to workers and members of the public leads to a total collective dose equivalent of 16 man-rem/y (Table 7.3). Because the calculations are conservative our estimate of the average national radiological impact of the use of radioisotopes as interwell tracers in EOR projects is a total collective dose equivalent of <16 man-rem/y.

Table 7.3 Estimate of the total radiological impact of the use of radioisotopes as interwell tracers in EOR.[a]

Population group	Stage of operation, Scenario type	Collective dose equivalent man-rem/y
Workers	Aboveground, normal	0.93
	Aboveground, accidental	0.13
Members of the public	Aboveground, accidental	0.034
	Subsurface, accidental	0.00017
	Recovery and disposal, normal	15
Total		16 man-rem/y

[a] Summarized from Tables 7.1 and 7.2.

8. ALTERNATIVES TO RADIOACTIVE TRACERS

Nonradioactive substances have been considered as tracers to monitor the behavior of liquids and gases in petroleum reservoirs. Substances used to trace fluids should be highly soluble and not be adsorbed as they traverse the reservoir with the injected fluids. Tracer materials that have been considered include (Wagner, 1977):

ammonium thiocyanate
ammonium nitrate
sodium or potassium bromide
sodium or potassium iodide
sodium chloride
fluorescent dyes
water-soluble alcohols

Chemical tracers such as ammonium thiocyanate, potassium iodide, ammonium nitrate, and SF_6 (gas) have been used as interwell tracers. Following are some of the considerations that may limit the applicability of chemical tracers:

a. Water-soluble alcohols such as methyl and ethyl alcohol are biodegradable and flammable.

b. Fluorescent dyes tend to adsorb, and fluorescence is often quenched in the presence of certain ions.

c. Relatively large amounts of chemicals must be used because chemical analysis is usually less sensitive than nuclear measurement and because of compensation for possible adsorption.

d. The large quantities needed lead to handling complications.

e. The large quantities needed could lead to alterations in chemistry with injected fluids becoming incompatible with reservoir fluids.

f. Tracer applications become unfavorable in the presence of high concentrations of carrier in the reservoir fluids.

g. The expense of using relatively large quantities of chemicals as tracers may be great.

h. The potential environmental impacts associated with the release of large quantities of certain chemicals may be unfavorable.

As an example of the quantities of nonradioactive tracers that are injected for interwell tracing, four chemical tracers were employed in a multiple injection project. Each of two wells received 10,000 pounds of ammonium nitrate, another well received 1,400 pounds of potassium iodide and another 2,500 pounds of ammonium thiocyanate (Wagner, 1977). By contrast, another well received 6 Ci of HTO, which amounts to less than 1 g of undiluted HTO.

From the standpoint of cost and ease of handling, radioisotopes are generally preferable to chemicals as interwell tracers for EOR operations. However, there have been situations where nonradioactive chemicals have been preferred as tracers. Nonradioactive chemical tracers are often used to complement radioactive tracers in EOR projects where a different tracer is injected in each of a number of wells.

9. SUMMARY AND CONCLUSIONS

An environmental assessment of the use of radioisotopes in field flooding for the enhanced recovery of oil and natural gas (EOR) was performed. To provide a framework for evaluating the dose, three stages of operation were designated in an EOR operation using tracers: the aboveground stage, the subsurface stage, and the recovery and disposal stage. Scenarios were developed for each of the stages to characterize normal and accidental exposures, and doses to workers and members of the public were estimated. The dose estimates presented in this report are 50-year committed dose equivalents for one year of operation.

A normalized dose, expressed as the dose per Ci or mCi injected or as the dose per injection of any amount of activity, was estimated for each radioisotope and scenario. The normalized dose for an isotope and the amount of isotope injected, or the number of injections involving the isotope, lead directly to a dose estimate for a normal exposure scenario. The dose estimate for an accidental exposure scenario is calculated in the same manner but includes an additional factor to express the probability that the accident will occur. Estimates of the total quantity of a tracer injected in a year, the total number of injections in a year, and accident frequency were then used to assess the total national radiological impact of the use of radioisotopes as interwell tracers in EOR. Accidental exposures in the course of EOR tracer projects have been extremely rare or nonexistent. Therefore, conservative values of accident frequencies (generally 1%) were assumed.

The collective dose equivalent in workers who handle the radioisotopes is delivered in the aboveground stage of operation and is estimated to be 0.93 man-rem/y for normal exposures and 0.13 man-rem/y for accidental exposures. ^{60}Co would be a major contributor to the doses from both normal and accidental aboveground exposures.

Members of the public are subject to accidental aboveground exposures, accidental subsurface exposures, and normal recovery and disposal exposures. Normal exposures associated with recovery and disposal account for essentially the total collective dose equivalent in the public, estimated to be 15 man-rem/y. The bulk of this total is attributable to HTO formed by the combustion of ^{3}H-labelled gaseous hydrocarbons recovered in natural gas. These dose estimates are much less than the collective dose estimates associated with the naturally occurring ^{222}Rn in natural gas.

The collective dose equivalent from the accidental surface release of gaseous radioisotopes is estimated to be 0.034 man-rem/y, which is more than two orders of magnitude lower than the 15 man-rem/y collective dose equivalent from the combustion of natural gas. Accidental subsurface exposures from ingesting water from an aquifer contaminated by radioactive tracers lead to collective doses that are negligible. The doses that could result from this scenario are greatly exceeded by the doses due to the liberation of naturally occurring ^{222}Rn in water supplies derived from groundwater. A more pressing concern than radionuclide entry into an aquifer would be the entry of salts, acids, alkali, polymers, surfactants, or other substances in the injection fluid with the potential to contaminate a water supply.

Nonradioactive chemicals are generally less suited than radioisotopes as interwell tracers for EOR but are used to complement radioisotopes as tracers particularly in multitracer applications. The total national radiological impact of the use of radioisotopes as interwell tracers in EOR projects, estimated by summing the collective dose to workers and members of the public, is a collective dose of <16 man-rem/y.

10. REFERENCES

C. J. Barton and R. E. Moore, "Radiation Doses from Hypothetical Exposures to Rulison Gas," Nucl. Technol. 20, 35-50 (1973).

C. Braxton, R. Stephens, C. Muller, J. White, J. Post, J. Norton, M. Goldberg, and P. Stevenson, Potential Environmental Consequences of Tertiary Oil Recovery, U.S. Environmental Protection Agency, Washington, D.C., PB-260 646 (1976).

Bureau of Radiological Health (BRH), Radiological Health Handbook, U.S. Department of Health, Education, and Welfare, Public Health Service, Rockville, Md., 1970.

D. A. Campbell, "Enhanced Oil-Recovery and its Environmental and Economic Implications in the United States," Environ. Conserv. 8, 5-18 (1981).

Chart of the Nuclides (General Electric Company, San Jose, Calif., 1973), 11th ed.

Code of Federal Regulations (CFR), 10 CFR 20 (1982).

Code of Federal Regulations (CFR), 10 CFR 71 (1982).

Code of Federal Regulations (CFR), 40 CFR 122 (1981).

Code of Federal Regulations (CFR), 40 CFR 123 (1981).

Code of Federal Regulations (CFR), 40 CFR 124 (1981).

Code of Federal Regulations (CFR), 40 CFR 146 (1981).

Code of Federal Regulations (CFR), 49 CFR (1982).

R. B. Codell, K. T. Key, and G. Whelen, A Collection of Mathematical Models for Dispersion in Surface Water and Groundwater, U.S. Nuclear Regulatory Commission, Washington, D.C., NUREG-0868 (1982).

A. G. Collins and C. C. Wright, Enhanced Oil Recovery Injection Waters, U.S. Department of Energy, Washington, D.C., DOE/BETC/RI-82/5 (1982).

C. L. Comar, "Radioactive Materials in Animals -- Entry and Metabolism," Radioactivity and Human Diet, R. S. Russell, Ed. (Pergamon Press, New York, 1966), pp. 127-158.

J. A. D'Hooge, C. Q. Sheely, and B. J. Williams, "Interwell Tracers - An Effective Reservoir Evaluation Tool: West Sumatra Field Results," J. Pet. Technol. 33, 779-782 (1981).

D. E. Dunning, G. G. Killough, S. R. Bernard, J. C. Pleasant, and P. J. Walsh, Estimates of Internal Dose Equivalent to 22 Target Organs for Radionuclides Occurring in Routine Releases from Nuclear Fuel-Cycle Facilities, Vol. III, Oak Ridge National Laboratory, Oak Ridge, Tenn., NUREG/CR-0150 Vol. 3 (1981).

R. A. Freeze and J. A. Cherry, Groundwater (Prentice-Hall, Inc., Englewood Cliffs, N.J., 1979).

S. R. Hanna, G. A. Briggs, and R. P. Hosker, Jr., Handbook on Atmospheric Diffusion, U. S. Department of Energy, Washington, D.C., DOE/TIC-11223 (1982).

G. J. Hine and G. L. Brownell, Eds., Radiation Dosimetry (Academic Press, Inc., New York, 1956).

F. O. Hoffman, R. H. Gardner, and K. F. Eckerman, Variability in Dose Estimates Associated with the Food Chain Transport and Ingestion of Selected Radionuclides, U.S. Nuclear Regulatory Commission, Washington, D.C., NUREG/CR-2612 ORNL/TM-8099 (1982).

Independent Petroleum Association of America (IPAA), The Oil Producing Industry in Your State 1982, Independent Petroleum Association of America, Washington, D.C. (1982).

International Commission on Radiological Protection (ICRP), Radiation Protection: Report of Committee IV on Evaluation of Radiation Doses to Body Tissues from Internal Contamination due to Occupational Exposure, ICRP Publication 10 (Pergamon Press, New York, 1968).

International Commission on Radiological Protection (ICRP), Radiation Protection: Limits for Intakes of Radionuclides by Workers, ICRP Publication 30 Supplement to Part 1 (Pergamon Press, New York, 1979).

International Commission on Radiological Protection (ICRP), Radiation Protection: Limits for Intakes of Radionuclides by Workers, ICRP Publication 30 Supplement to Part 2 (Pergamon Press, New York, 1980).

International Commission on Radiological Protection (ICRP), Radiation Protection: Limits for Intakes of Radionuclides by Workers, ICRP Publication 30 Supplement A to Part 3 (Pergamon Press, New York, 1982).

D. G. Jacobs, E. G. Struxness, M. J. Kelly, and C. R. Bowman, "Consideration of the Radiological Impact from the Hypothetical Use of Contaminated Natural Gas from Nuclearly Stimulated Reservoirs," Health Phys. 22, 429-440 (1972a).

D. G. Jacobs, C. J. Barton, M. J. Kelly, C. R. Bowman, S. R. Hanna, F. A. Gifford, Jr., and W. M. Culkowski, Theoretical Evaluation of Consumer Products from Project Gasbuggy. Final Report, Phase II: Hypothetical Population Exposures Outside San Juan Basin, Oak Ridge National Laboratory, Oak Ridge, Tenn., ORNL-4748 (1972b).

R. H. Johnson, Jr., D. W. Bernhardt, N. S. Nelson, and H. W. Calley, Jr., "Radiological Health Significance of Radon in Natural Gas," in Noble Gases, R. E. Stanley and A. A. Moghissi, Eds., CONF-730915 (1973), (U.S. Govt. Printing Office, Washington, D.C.) pp. 532-539.

H. R. Johnson, Outlook for Enhanced Oil Recovery, Bartlesville Energy Technology Center, Bartlesville, Okla., DOE/BETC/OP-82/4 (1982).

D. C. Kocher, Radioactive Decay Data Tables, Technical Information Center, U.S. Department of Energy, Washington, D.C., DOE/TIC-11026 (1981).

R. Lange, PATRIC, A Three Dimensional Particle-In-Cell Sequential Puff Code for Modeling the Transport and Diffusion of Atmospheric Pollutants, Lawrence Livermore National Laboratory, Livermore, Calif., UCID-17701 (1978).

R. Loevinger, E. M. Japha, and G. L. Brownell, "Discrete Radioisotope Sources," in Radiation Dosimetry, G. J. Hine and G. L. Brownell, Eds. (Academic Press, Inc., New York, 1956), pp. 694-799.

D. B. McWhorter and D. K. Sunada, Ground-Water Hydrology and Hydraulics (Water Resources Publications, Fort Collins, Colo., 1977).

S. L. Matheny, "EOR Methods Help Ultimate Recovery," Oil Gas J. 78(13), 79-124 (1980).

M. Nordyke, "Underground Engineering Applications, Concepts and Experience: Gas Stimulation with PNE," Peaceful Nuclear Explosions II, Their Practical Applications, Proceedings, (International Atomic Energy Agency, Vienna, 1971), pp. 23-59.

H. M. Prichard and T. F. Gesell, "An Estimate of Population Exposures Due to Radon in Public Water Supplies in the Area of Houston, Texas," Health Phys. 41, 599-606 (1981).

Railroad Commission of Texas (RCT), Rules Having Statewide General Application to Oil, Gas, and Geothermal Resource Operations Within the State of Texas (Railroad Commission of Texas Oil and Gas Division, Austin, Tex., 1982).

M. M. Schumacher, Ed., Enhanced Oil Recovery. Secondary and Tertiary Methods (Noyes Data Corp., Park Ridge, N.J., 1978).

D. H. Slade, Meteorology and Atomic Energy 1968, U.S. Atomic Energy Commission, Washington, D.C., TID-24190 (1968).

J. K. Soldat, P. E. Bramson, and H. M. Parker, "Dosimetry for the Noble Gases," in Noble Gases, R. E. Stanley and A. A. Moghissi, Eds. (U.S. Environmental Protection Agency, Washington, D.C., 1973), pp. 432-438.

Spears and Associates, Inc., Environmental Regulations Handbook for Enhanced Oil Recovery, U.S. Department of Energy, Washington, D.C., DOE/BC/00050-15 (1980).

H. A. Tewes, Survey of Gas Quality Results from 3 Gas-Well-Stimulation Experiments by Nuclear Explosives, Lawrence Livermore National Laboratory, Livermore, Calif., UCRL-52656 (1979).

D. K. Todd and D. E. O. McNulty, Polluted Groundwater. A Review of the Significant Literature (Water Information Center, Inc., Port Washington, N.Y., 1976).

United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR). Ionizing Radiation: Sources and Biological Effects (United Nations, New York, 1982).

U. S. Nuclear Regulatory Commission (USNRC), Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, Regulatory Guide 1.109 Rev. 1 (1977).

O. R. Wagner, "The Use of Tracers in Diagnosing Interwell Reservoir Heterogeneities - Field Results," J. Pet. Technol. 29, 1410-1416 (1977).

APPENDIX A. CALCULATION OF EXPOSURE RATES FROM AN EXTERNAL SOURCE

A.1 Exposure rate from a point source

The exposure rate 1 cm from a gamma-emitting point source of 1-millicurie activity is referred to as the dose-rate constant, Γ. The Γ, in cm^2-R/mCi-h, for an isotope that emits one photon is given by (Loevinger et al., 1956)

$$\Gamma = 1.50 \times 10^5 \mu_a E_\gamma \qquad \text{(A-1)}$$

where

μ_a = true linear absorption coefficient in air (cm^{-1}),
E_γ = gamma energy (MeV)

The linear absorption coefficient, μ_a, varies with E_γ. When gamma rays of different energies are emitted, the Γ for the isotope is found by adding the contributions for each energy.

$$\Gamma = \sum_j P_j \Gamma(E_\gamma)_j \qquad \text{(A-2)}$$

where

P_j = the abundance of the jth γ-ray

The exposure rate at any distance from a point source can be determined by

$$ER = q \sum_j P_j \Gamma(E_\gamma)_j \exp(-\mu r)/r^2 \qquad \text{(A-3)}$$

where

ER = exposure rate (R/hr)
q = source activity (mCi)
μ = linear attenuation coefficient in air (cm^{-1})
r = distance (cm)

The linear attenuation coefficient, μ, varies with gamma energy, E_γ.

Because the source used as an interwell tracer in field flooding operations is confined in a vessel or container, Eq. A-3, which applies to an unshielded source, would overestimate the exposure rate. A shielding factor, F, can be used to correct the Γ values for absorption of photons in the vessel walls.

$$F = \exp(-\mu_v d) \qquad \text{(A-4)}$$

where

μ_v = linear attenuation coefficient for vessel wall material (cm^{-1})
d = thickness of vessel wall (cm)

Table A.1 presents Γ values for various EOR radionuclides (see Table 2.1) calculated by Eqs. A-1 and A-2. In the calculations for Γ, $(E_\gamma)_j$ and P_j were obtained from Kocher (1981). The μ_a values for $E_\gamma \geq 0.01$ MeV are interpolated values based on the mass energy-absorption coefficient (μ_a/ρ) for air at several energies as listed in the Radiological Health Handbook (BRH, 1970). The density of air, ρ, was assumed to be 0.001293 g/cm^3. Characteristic L and K X-rays below 0.01 MeV in energy were excluded from the calculations because these low-energy photons would not penetrate clothing and the dead surface layer of skin. In Table A.1 both published

Table A.1 Dose rate constants for radioisotope tracers used in EOR operations.

Radionuclide	Γ, cm²-R/mCi-h		Radionuclide	Γ, cm²-R/mCi-h	
	Calculated	Published[a]		Calculated	Published[a]
H-3	0		Co-60	12.9	13.2
C-14	0		Ni-63	0	
Na-22	11.8	12.0	Zn-65	3.09	2.7
Cl-36	0		Kr-85	0.013	∿0.04
Ar-37	0		Sr-85	6.35	3.0
Ar-39	0		Sr-89	0.001	
Ca-45	0		Sr-90	0	
Cr-51	0.18	0.16	Tc-99	0	
Fe-55	0		Ag-110m	14.8	
Fe-59	6.22	6.4	I-125	1.70	∿0.7
Co-57	1.01	0.9	I-131	2.19	2.2
Co-58	5.45	5.5	Xe-133	0.53	0.1

[a]Source: Radiological Health Handbook (BRH, 1970).

and calculated values of Γ are presented. The two sets of values may differ slightly because the calculated values are based on the recently updated photon abundances and intensities compiled by Kocher (1981).

Exposure rates at various distances from a point source were calculated by Eq. A-3. In the calculation of exposure rates, ER, the linear attenuation coefficients, μ, for $E_\gamma > 0.01$ MeV are interpolated values based on the mass attenuation coefficient (μ/ρ) in air at several energies as listed in the Radiological Health Handbook (BRH, 1970).

To account for gamma absorption in the walls of the vessel containing the tracer, another set of exposure-rate calculations was performed using F from Eq. A-4 as a shielding factor for Eq. A-3 (Table A.2). To simplify the computations without introducing a substantial error, μ/ρ for air and glass were assumed to be equal. The density, ρ, of glass was assumed to be 2.1, and the vessel wall thickness, d, was assumed to be 0.2 cm.

A.2 Exposure rate from a distributed source

To evaluate accidental exposures from liquid tracers accidentally spilled on the ground surface or on the wellhead surface, exposure rates were calculated for a source that is assumed to be uniformly distributed on the surface of a circular disk. The exposure rate resulting from a uniformly loaded circular disk is (Loevinger et al., 1956)

$$R = \frac{q\Gamma\exp(-\mu r)}{a^2} \ln \frac{a^2-b^2+h^2+\sqrt{(a^2-b^2+h^2)^2+4b^2h^2}}{2h^2} \qquad \text{(A-5)}$$

TABLE A 2

EXPOSURE RATE IN AIR AT VARIOUS DISTANCES FROM A POINT SOURCE SHIELDED BY 2MM OF GLASS

MR/H PER MILLICURIE

RANGE (M)	H 3	C 14	NA 22	CL 36	AR 37	AR 39	CA 45	CR 51	FE 55	FE 59	CO 57
0 1	0	0.	1 15E+02	0	0	0	0.	1 68E+00	0.	6 06E+01	7 15E+00
0 2	0	0	2.87E+01	0	0	0	0.	4.20E-01	0.	1 51E+01	1.77E+00
0 3	0	0.	1 27E+01	0.	0.	0	0.	.87E-01	0	6 72E+00	7 82E-01
0 4	0	0.	7 15E+00	0	0	0	0	1.05E-01	0.	3.78E+00	4 36E-01
0 5	0	0	4.57E+00	0	0	0	0.	6 70E-02	0.	2.42E+00	2.77E-01
0 6	0	0	3 17E+00	0	0	0	0.	4.65E-02	0	1 68E+00	1.91E-01
0 7	0	0	2.33E+00	0	0	0	0	3 41E-02	0.	1 23E+00	1.39E-01
0 8	0	0	1 78E+00	0.	0.	0	0	2.61E-02	0.	9.41E-01	1.06E-01
0 9	0.	0	1 41E+00	0	0	0	0.	2.06E-02	0.	7.43E-01	8.29E-02
1 0	0.	0.	1.14E+00	0	0.	0.	0.	1 66E-02	0.	6 02E-01	6.67E-02
2 0	0.	0.	2.82E-01	0.	0.	0	0.	4.10E-03	0.	1 49E-01	1.58E-02
3 0	0	0.	1.24E-01	0.	0.	0	0.	1.80E-03	0.	6 58E-02	6.53E-03
4.0	0	0.	6.93E-02	0.	0.	0.	0.	9.98E-04	0.	3.67E-02	3.49E-03
5 0	0	0.	4.39E-02	0	0.	0.	0.	6 30E-04	0.	2.33E-02	2.13E-03
6 0	0	0	3.02E-02	0	0.	0	0.	4 32E-04	0	1.61E-02	1.42E-03
7.0	0.	0.	2 20E-02	0.	0.	0	0	3 13E-04	0	1 17E-02	1.01E-03
8.0	0	0.	1.67E-02	0	0.	0.	0.	2.36E-04	0.	8.91E-03	7.48E-04
9.0	0	0	1.31E-02	0	0.	0	0.	1.84E-04	0	6 98E-03	5.73E-04
10 0	0	0	1.05E-02	0	0.	0	0.	1 47E-04	0.	5 61E-03	4.52E-04
11.0	0.	0.	8.60E-03	0	0	0.	0.	1.20E-04	0.	4.60E-03	3.64E-04
12 0	0.	0	7 16E-03	0	0	0	0.	9.96E-05	0.	3.84E-03	2.99E-04
13.0	0	0.	6.05E-03	0.	0.	0.	0.	8.37E-05	0.	3.25E-03	2 49E-04
14.0	0.	0.	5.17E-03	0.	0.	0.	0.	7.12E-05	0.	2.78E-03	2.10E-04
15.0	0.	0.	4.46E-03	0.	0	0	0.	6.12E-05	0.	2.40E-03	1.79E-04
16 0	0.	0.	3.89E-03	0.	0.	0.	0.	5.31E-05	0.	2.09E-03	1.54E-04
17.0	0.	0.	3.41E-03	0.	0.	0.	0.	4.64E-05	0.	1.84E-03	1.34E-04
18 0	0.	0.	3.02E-03	0.	0.	0	0.	4.08E-05	0.	1.63E-03	1.17E-04
19.0	0.	0.	2.68E-03	0.	0.	0.	0.	3.61E-05	0.	1.45E-03	1.03E-04
20 0	0.	0.	2.40E-03	0.	0.	0.	0.	3.22E-05	0.	1.30E-03	9.09E-05

TABLE A 2 (CONT)

EXPOSURE RATE IN AIR AT VARIOUS DISTANCES FROM A POINT SOURCE SHIELDED BY 2MM OF GLASS
MR/H PER MILLICURIE

RANGE (M)	CO 58	CO 60	NI 63	ZN 65	KR 85	SR 85	SR 89	SR 90	TC 99	AG110M	I125
0 1	5 28E+01	1.26E+02	0	3 00E+01	1.23E-01	3 93E+01	9.71E-03	0	0	1.44E+02	1.41E+01
0 2	1.32E+01	3.15E+01	0	7 51E+00	3.08E-02	9 74E+00	2.42E-03	0	0	3.59E+01	3.50E+00
0 3	5 85E+00	1 40E+01	0	3.33E+00	1.37E-02	4 29E+00	1.08E-03	0	0.	1.60E+01	1.55E+00
0 4	3 29E+00	7 87E+00	0	1 87E+00	7 69E-03	2 39E+00	6.05E-04	0.	0.	8.97E+00	8.66E-01
0 5	2 10E+00	5.03E+00	0	1 20E+00	4 91E-03	1.51E+00	3.87E-04	0	0.	5.74E+00	5.51E-01
0 6	1 46E+00	3 49E+00	0	8 31E-01	3 41E-03	1 04E+00	2 68E-04	0.	0.	3.98E+00	3.81E-01
0 7	1 07E+00	2 56E+00	0	6 10E-01	2 50E-03	7 58E-01	1.97E-04	0	0.	2.92E+00	2.78E-01
0.8	8.19E-01	1 96E+00	0	4 67E-01	1.91E-03	5 76E-01	1.51E-04	0.	0.	2.23E+00	2.12E-01
0 9	6 47E-01	1 55E+00	0	3 69E-01	1.51E-03	4 51E-01	1 19E-04	0.	0.	1.76E+00	1.66E-01
1.0	5.23E-01	1.25E+00	0.	2 98E-01	1.22E-03	3.62E-01	9.63E-05	0.	0.	1.43E+00	1.34E-01
2 0	1.30E-01	3.11E-01	0.	7 40E-02	3.02E-04	8 40E-02	2.39E-05	0.	0.	3.54E-01	3.17E-02
3.0	5.71E-02	1.37E-01	0	3.26E-02	1.33E-04	3 52E-02	1.05E-05	0.	0.	1.58E-01	1.33E-02
4.0	3.18E-02	7.66E-02	0.	1.82E-02	7 39E-05	1.89E-02	5.87E-06	0.	0.	8.69E-02	7 07E-03
5 0	2 02E-02	4.87E-02	0.	1 16E-02	4.67E-05	1.16E-02	3.72E-06	0.	0.	5.51E-02	4.28E-03
6.0	1 39E-02	3.36E-02	0.	7.96E-03	3.21E-05	7.81E-03	2 56E-06	0.	0.	3.79E-02	2.81E-03
7 0	1.01E-02	2.45E-02	0.	5.80E-03	2.33E-05	5 59E-03	1.87E-06	0	0.	2 76E-02	1.95E-03
8 0	7.66E-03	1 86E-02	0.	4 41E-03	1.77E-05	4 18E-03	1.42E-06	0	0.	2.10E-02	1.41E-03
9 0	5.99E-03	1 46E-02	0.	3.46E-03	1.38E-05	3.24E-03	1 11E-06	0.	0.	1 64E-02	1.05E-03
10 0	4.81E-03	1.17E-02	0.	2.78E-03	1.11E-05	2 58E-03	8 91E-07	0.	0.	1.32E-02	8.07E-04
11.0	3.94E-03	9.62E-03	0.	2.28E-03	9.04E-06	2 10E-03	7.30E-07	0.	0.	1.08E-02	6.30E-04
12 0	3.28E-03	8.02E-03	0.	1.90E-03	7.51E-06	1 74E-03	6.08E-07	0.	0.	9.00E-03	5.01E-04
13.0	2.77E-03	6.79E-03	0.	1.60E-03	6.33E-06	1 46E-03	5.14E-07	0.	0.	7.60E-03	4.03E-04
14.0	2.36E-03	5.81E-03	0	1.37E-03	5.40E-06	1.24E-03	4.39E-07	0.	0.	6.50E-03	3.29E-04
15.0	2.04E-03	5.02E-03	0.	1.19E-03	4.65E-06	1.07E-03	3.79E-07	0.	0.	5.61E-03	2.71E-04
16.0	1.78E-03	4.38E-03	0.	1.03E-03	4.04E-06	9.28E-04	3.31E-07	0.	0.	4.89E-03	2.25E-04
17.0	1.56E-03	3.85E-03	0.	9.09E-04	3.54E-06	8.12E-04	2.90E-07	0.	0.	4.29E-03	1.89E-04
18.0	1.38E-03	3.41E-03	0.	8.04E-04	3.12E-06	7 16E-04	2.57E-07	0	0.	3.79E-03	1.59E-04
19.0	1.22E-03	3.04E-03	0.	7.16E-04	2.77E-06	6.35E-04	2.28E-07	0.	0.	3.38E-03	1.35E-04
20.0	1.09E-03	2.72E-03	0.	6.41E-04	2 47E-06	5 67E-04	2.04E-07	0.	0.	3.02E-03	1.15E-04

TABLE A 2 (CONT)

EXPOSURE RATE IN AIR AT VARIOUS DISTANCES FROM A POINT SOURCE SHIELDED BY 2MM OF GLASS
MR/H PER MILLICURIE

RANGE (M)	I131	XE133
0 1	2 10E+01	4 64E+00
0 2	5 24E+00	1 15E+00
0 3	2 32E+00	5 11E-01
0 4	1 31E+00	2.87E-01
0 5	8.34E-01	1 83E-01
0.6	5 79E-01	1.26E-01
0 7	4 25E-01	9.25E-02
0 8	3.25E-01	7.06E-02
0 9	2 56E-01	5 56E-02
1 0	2.07E-01	4.48E-02
2.0	5.11E-02	1.08E-02
3 0	2 24E-02	4.63E-03
4 0	1 25E-02	2.51E-03
5 0	7.87E-03	1.55E-03
6 0	5.39E-03	1.04E-03
7 C	3.91E-03	7.34E-04
8 0	2.96E-03	5.42E-04
9.0	2.31E-03	4.13E-04
10.0	1.84E-03	3.23E-04
11.0	1.50E-03	2.57E-04
12.0	1.25E-03	2.08E-04
13 0	1.05E-03	1.71E-04
14.0	8.94E-04	1.43E-04
15.0	7.69E-04	1.20E-04
16 0	6.67E-04	1.02E-04
17.0	5.83E-04	8.69E-05
18 0	5.14E-04	7.40E-05
19 0	4.55E-04	6.48E-05
20 0	4.05E-04	5.65E-05

where

a = radius of disk (cm),
b = distance from center of disk to receptor along the plane of the disk (cm),
h = height of receptor above plane of disk (cm),
r = distance from receptor to center of disk (cm),

$$r = \sqrt{b^2 + h^2}$$

and the other symbols are as previously defined.

Table A.3 presents exposure rates from a disk uniformly contaminated with a total of 1 mCi of each of the EOR radionuclides of Table A.1. The estimates cover a range of distances (b) and assume a radius, r, of 15 cm and a receptor height, h, of 1 m (100 cm).

TABLE A 3

EXPOSURE RATE IN AIR (1 METER HT) AT VARIOUS DISTANCES FROM THE CENTER OF A CIRCULAR DISK ON THE GROUND SURFACE UNIFORMLY CONTAMINATED BY 1 MILLICURIE OF A RADIONUCLIDE RADIUS OF DISK EQUALS 15 CM

MR/H PER MILLICURIE

RANGE (M)	H 3	C 14	NA 22	CL 36	AR 37	AR 39	CA 45	CR 51	FE 55	FE 59	CO 57
0.1	0.	0.	1.16E+00	0.	0.	0	0.	1 72E-02	0.	6.08E-01	9.76E-02
0.2	0.	0.	1.12E+00	0.	0.	0.	0.	1.67E-02	0.	5 91E-01	9.37E-02
0 3	0.	0	1.07E+00	0.	0.	0	0.	1 60E-02	0.	5.64E-01	8 84E-02
0 4	0	0	1.01E+00	0.	0.	0	0	1 50E-02	0.	5.30E-01	8.22E-02
0 5	0	0	9.37E-01	0.	0	0	0.	1 39E-02	0	4.93E-01	7.55E-02
0.6	0	0.	8.62E-01	0	0.	0	0.	1 28E-02	0.	4 53E-01	6.87E-02
0.7	0.	0	7.87E-01	0.	0.	0	0.	1 17E-02	0	4.14E-01	6.21E-02
0 8	0.	0.	7.15E-01	0.	0	0	0.	1.06E-02	0.	3.76E-01	5.58E-02
0 9	0.	0.	6 48E-01	0.	0.	0	0.	9.61E-03	0.	3 41E-01	5.00E-02
1 0	0.	0.	5.86E-01	0.	0.	0.	0.	8.69E-03	0.	3.08E-01	4.48E-02
2 0	0.	0.	2.33E-01	0.	0.	0	0.	3.44E-03	0.	1 23E-01	1.82E-02
3 0	0.	0.	1.15E-01	0.	0.	0.	0.	1.69E-03	0.	6.08E-02	7.37E-03
4.0	0.	0.	6.72E-02	0.	0.	0	0	9.83E-04	0.	3 55E-02	4.00E-03
5.0	0.	0.	4.35E-02	0.	0.	0.	0	6.34E-04	0.	2.30E-02	2.44E-03
6.0	0.	0.	3.03E-02	0.	0.	0.	0.	4.39E-04	0.	1.60E-02	1.61E-03
7.0	0.	0	2.22E-02	0.	0.	0.	0	3.21E-04	0.	1.18E-02	1.13E-03
8.0	0.	0.	1.69E-02	0.	0.	0.	0.	2.43E-04	0.	8.99E-03	8.29E-04
9.0	0.	0.	1.33E-02	0.	0.	0.	0.	1.90E-04	0.	7.07E-03	6.31E-04
10.0	0.	0.	1.07E-02	0.	0.	0.	0.	1.52E-04	0.	5.70E-03	4.94E-04
11.0	0.	0.	8.78E-03	0.	0.	0.	0.	1.24E-04	0.	4.68E-03	3.95E-04
12.0	0.	0.	7.32E-03	0.	0.	0.	0.	1.03E-04	0.	3.91E-03	3.23E-04
13.0	0.	0.	6.19E-03	0.	0.	0.	0.	8.69E-05	0.	3.31E-03	2.68E-04
14.0	0.	0.	5.29E-03	0.	0.	0.	0.	7.40E-05	0.	2.83E-03	2.25E-04
15.0	0.	0.	4.57E-03	0.	0.	0.	0.	6 37E-05	0.	2 45E-03	1.91E-04
16.0	0.	0.	3.99E-03	0.	0.	0.	0.	5.52E-05	0.	2 14E-03	1.64E-04
17.0	0.	0.	3.50E-03	0.	0.	0.	0.	4.83E-05	0.	1.88E-03	1.42E-04
18.0	0.	0.	3.10E-03	0.	0.	0.	0.	4.25E-05	0.	1.67E-03	1.24E-04
19.0	0.	0.	2.75E-03	0.	0.	0.	0.	3.76E-05	0.	1 48E-03	1.09E-04
20.0	0.	0.	2.46E-03	0	0	0.	0.	3.35E-05	0.	1.33E-03	9.67E-05

TABL A 3 (CONT)

EXPOSURE RATE IN AIR (1 METER HT) AT VARIOUS DISTANCES FROM THE CENTER OF A CIRCULAR DISK ON THE GROUND SURFACE UNIFORMLY CONTAMINATED BY 1 MILLICURIE OF A RADIONUCLIDE RADIUS OF DISK EQUALS 15 CM

MR/H PER MILLICURIE

RANGE (M)	CO 58	CO 60	NI 63	ZN 65	KR 85	SR 85	SR 89	SR 90	TC 99	AG110M	I125
0 1	5 33E-01	1 27E+00	0.	3.02E-01	1.25E-03	6.10E-01	9.78E-05	0	0.	1.45E+00	1.66E-01
0 2	5.18E-01	1.23E+00	0.	2.93E-01	1.22E-03	5.82E-01	9 49E-05	0	0	1.41E+00	1.60E-01
0.3	4 94E-01	1.17E+00	0.	2.80E-01	1.16E-03	5.46E-01	9.06E-05	0.	0.	1 34E+00	1 52E-01
0.4	4 65E-01	1.10E+00	0.	2.63E-01	1.09E-03	5.05E-01	8.52E-05	0.	0.	1 26E+00	1.43E-01
0.5	4.31E-01	1.03E+00	0.	2.45E-01	1.01E-03	4.61E-01	7.91E-05	0	0.	1 17E+00	1.32E-01
0 6	3.97E-01	9 43E-01	0	2.25E-01	9.32E-04	4 17E-01	7.28E-05	0	0	1 08E+00	1.21E-01
0 7	3 62E-01	8 62E-01	0	2 05E-01	8 51E-04	3.75E-01	6 65E-05	0	0	9.86E-01	1.10E-01
0 8	3 29E-01	7 83E-01	0	1 87E-01	7 73E-04	3.35E-01	6.04E-05	0	0.	8.96E-01	9.92E-02
0 9	2 98E-01	7 10E-01	0.	1.69E-01	7 00E-04	2 99E-01	5.47E-05	0	0.	8.12E-01	8.94E-02
1 0	2 70E-01	6.42E-01	0	1.53E-01	6 33E-04	2.67E-01	4.95E-05	0.	0.	7 35E-01	8.05E-02
2.0	1.07E-01	2.55E-01	0	6.08E-02	2.51E-04	9.23E-02	1.97E-05	0.	0.	2 92E-01	3.05E-02
3 0	5.30E-02	1.27E-01	0.	3 02E-02	1.24E-04	4.09E-02	9 74E-06	0.	0.	1.44E-01	1 44E-02
4 0	3.09E-02	7.39E-02	0	1 76E-02	7.21E-05	2.18E-02	5.68E-06	0.	0	8.42E-02	8.00E-03
5 0	2.00E-02	4.80E-02	0.	1 14E-02	4.66E-05	1 31E-02	3.68E-06	0.	0	5.46E-02	4.94E-03
6.0	1.39E-02	3.35E-02	0.	7.95E-03	3.24E-05	8.67E-03	2.56E-06	0.	0	3.80E-02	3.28E-03
7.0	1.02E-02	2.46E-02	0.	5.84E-03	2.37E-05	6.10E-03	1.88E-06	0.	0.	2.79E-02	2.29E-03
8.0	7.78E-03	1.88E-02	0.	4.46E-03	1.80E-05	4.50E-03	1.43E-06	0.	0	2.13E-02	1.67E-03
9.0	6.11E-03	1.48E-02	0.	3.50E-03	1.41E-05	3.45E-03	1.13E-06	0.	0.	1.67E-02	1.25E-03
10 0	4 91E-03	1.19E-02	0	2.82E-03	1.13E-05	2.73E-03	9.07E-07	0.	0	1.34E-02	9.59E-04
11 0	4.03E-03	9.77E-03	0.	2.32E-03	9 29E-06	2.21E-03	7.45E-07	0.	0.	1.10E-02	7.50E-04
12.0	3.36E-03	8 16E-03	0	1.93E-03	7 73E-06	1 82E-03	6.21E-07	0.	0.	9.20E-03	5.97E-04
13.0	2.84E-03	6.91E-03	0	1.64E-03	6.52E-06	1.52E-03	5.25E-07	0.	0.	7.78E-03	4.81E-04
14 0	2.42E-03	5.92E-03	0.	1.40E-03	5.57E-06	1 29E-03	4.49E-07	0.	0	6.65E-03	3.93E-04
15.0	2.09E-03	5.12E-03	0.	1.21E-03	4.80E-06	1 11E-03	3.88E-07	0.	0.	5.75E-03	3.23E-04
16 0	1 82E-03	4 47E-03	0	1.06E-03	4.17E-06	9.63E-04	3.39E-07	0.	0.	5.01E-03	2.69E-04
17.0	1.60E-03	3.93E-03	0.	9.30E-04	3.66E-06	8 43E-04	2.97E-07	0.	0.	4.40E-03	2.25E-04
18.0	1 42E-03	3.48E-03	0	8.23E-04	3.23E-06	7.42E-04	2.63E-07	0.	0.	3.89E-03	1.90E-04
19.0	1.26E-03	3 10E-03	0.	7 33E-04	2.86E-06	6.58E-04	2.34E-07	0.	0.	3.46E-03	1.61E-04
20 0	1 13E-03	2.78E-03	0.	6.56E-04	2 56E-06	5 87E-04	2.10E-07	0.	0.	3.10E-03	1.38E-04

TABLE 6 3 (CONT)

EXPOSURE RATE IN AIR (1 METER HT) AT VARIOUS DISTANCES FROM THE CENTER OF A CIRCULAR DISK ON THE GROUND SURFACE UNIFORMLY CONTAMINATED BY 1 MILLICURIE OF A RADIONUCLIDE RADIUS OF DISK EQUALS 15 CM

MR/H PER MILLICURIE

RANGE (M)	I131	XE133
0 1	2 14E-01	5 13E-02
0 2	2 08E-01	4 96E-02
0 3	1 99E-01	4 73E-02
0 4	1 87E-01	4.43E-02
0 5	1 73E-01	4.10E-02
0 6	1.59E-01	3 76E-02
0 7	1.45E-01	3 43E-02
0 8	1.32E-01	3 10E-02
0 9	1 20E-01	2 80E-02
1 0	1.08E-01	2 53E-02
2 0	4 27E-02	9.76E-03
3 0	2.11E-02	4.70E-03
4 0	1 22E-02	2 66E-03
5 0	7 90E-03	1 68E-03
6 0	5 48E-03	1 14E-03
7.0	4.00E-03	8 10E-04
8 0	3 04E-03	6 01E-04
9 0	2 38E-03	4.59E-04
10.0	1 90E-03	3.59E-04
11.0	1.56E-03	2.87E-04
12.0	1.29E-03	2.33E-04
13.0	1.09E-03	1.91E-04
14.0	9 27E-04	1.59E-04
15 0	7 98E-04	1.34E-04
16.0	6.93E-04	1.14E-04
17 0	6 06E-04	9 72E-05
18.0	5.34E-04	8.37E-05
19.0	4 73E-04	7.25E-05
20.0	4 22E-04	6.32E-05

APPENDIX B. CALCULATION OF EXPOSURES FROM THE ACCIDENTAL RELEASE OF GASEOUS TRACERS

B.1 Dose to workers from a ground-level release

The time-integrated downwind concentration of a gas within the first 100 meters of cloud travel following a ground-level release to the atmosphere is estimated using a Gaussian model. For an instantaneous release from an elevated point source under constant diffusion conditions (i.e., constant wind direction, wind speed, and atmospheric stability) and allowing for plume reflection at ground level, the concentration is given by (Slade, 1968)

$$\chi(x,y,z) = Q(2\pi)^{-3/2}(\sigma_x\sigma_y\sigma_z)^{-1} \exp[-(x-ut)^2/2\sigma_x^2] \exp(-y^2/2\sigma_y^2) \cdot \exp[-(z-H)^2/2\sigma_z^2 - (z+H)^2/2\sigma_z^2] \quad \text{(B-1)}$$

where the origin of the coordinate system is at ground level directly below the release point, and

$\chi(x,y,z)$	=	air concentration at point with coordinates x,y,z (Ci/m^3),
x	=	downwind distance (m)
y	=	crosswind distance (m)
z	=	height above ground (m)
Q	=	release amount (Ci)
u	=	mean wind speed (m/s)
$\sigma_x, \sigma_y, \sigma_z$	=	standard deviations of the concentration distribution along the wind direction and in the crosswind and vertical directions (m)
H	=	effective release height (m)

The time-integrated concentration or total exposure, ψ, at sensor height h in the downwind direction is given by

$$\psi = \int_{x/u}^{\infty} \chi(x-ut,y,h)dt \quad \text{(B-2)}$$

If the diffusion parameters (σ_y and σ_z), wind velocity (u), and wind direction are considered to be constant over a period of time (Δt), the time integral of the concentration at height (h) in the downwind direction for a ground-level release (H = 0) can be written, from Eqs. B-1 and B-2, as

$$\psi = Q(\pi\sigma_y\sigma_z)^{-1}\exp(-y^2/2\sigma_y^2 - h^2/2\sigma_z^2) \quad \text{(B-3)}$$

A crucial part of this approach is the derivation of appropriate values of σ_y and σ_z to characterize turbulent diffusion. In this treatment we let $\sigma_x = \sigma_y = \sigma_z = \sigma$ and adopt Batchelor's approach for estimating the rate of relative diffusion in the inertial range. The rate of diffusion is given by (Hanna et al., 1982)

$$\frac{d\sigma^2}{dt} \propto t(\epsilon u_o)^{2/3} \quad \text{(B-4)}$$

where

$\frac{d\sigma^2}{dt}$ = rate of relative diffusion (m^2/s)
t = time after release (s)
ε = eddy dissipation rate (m^2/s^3)
σ_o = initial size of puff (m)

Integration of Eq. B-4 yields an expression for the dimensions of the puff (σ) in the inertial range

$$\sigma^2 = \sigma_o^2 + Ct^2(\varepsilon\sigma_o)^{2/3} \qquad \text{(B-5)}$$

Data from which to evaluate the constant C are not abundant. It is of the order of unity and is here assumed to be 1. Eq. B-5 applies only for a few seconds following release, out to 3 seconds for these simulations.

The value of ε can be obtained from boundary layer theory,

$$\varepsilon = (u_*^3/kz)(\phi - z/L) \qquad \text{(B-6)}$$

where u_* = friction velocity (m/s)

k = von Karman's constant, which is assumed to equal 0.4
ϕ = the diabatic influence function $\phi(z/L)$
L = the Monin-Obukhov scaling length (m)

One can readily evaluate ε by considering a typical outdoor surface (roughness length about z_o = 3 cm) and an extreme range of atmospheric stabilities, and letting

$$u_* = 0.1\ u/\phi \qquad \text{(B-7)}$$

Setting z = h, and using the iollowing for the three meteorological scenarios, one has

(a) Very unstable atmosphere ($z/L < 0$)

$$\phi(z/L) = (1 - 15\ z/L)^{-1/4} \qquad \text{(B-8)}$$

let $z/L = -1$
$\phi = 0.5$
$$\varepsilon = 3 \times 10^{-2}\ u^3/h \qquad \text{(B-9)}$$

(b) Neutral atmosphere ($z/L = 0$)
$\phi = 1.0$
$$\varepsilon = 2.5 \times 10^{-3}\ u^3/h \qquad \text{(B-10)}$$

(c) Very stable atmosphere ($z/L > 0$)
$$\phi(z/L) = 1 + 5\ z/L \qquad \text{(B-11)}$$
let $z/L = 0.2$
$\phi = 2$
$$\varepsilon = 5.6 \times 10^{-4}\ u^3/h \qquad \text{(B-12)}$$

For travel times greater than 3 seconds, the diffusion parameter σ^2 becomes proportional to t^3 by 100 to 200 seconds of travel (Hanna et al., 1982). Thus, with the value σ_*^2 obtained from Eq. B-5 for t = 3 seconds, σ_2 for 3 to 150 seconds of travel is

$$\sigma^2 = w\sigma_*^2 + (1-w)\,\varepsilon t^{\alpha}, \qquad \text{(B-13)}$$

where

$$w = 3/(2t-3) \qquad \text{(B-14)}$$

and

$$\alpha = 2 + (t-3)/(150-3) \qquad \text{(B-15)}$$

Eq. B-13 allows a smooth transition from $\sigma^2 \propto t^2$ to $\sigma^2 \propto t^3$, over the time period from 3 to 150 seconds.

For times from 150 seconds to 10^4 seconds, and for values of σ less than about three-tenths of the mixed layer depth,

$$\sigma^2 = \varepsilon t^3 \qquad \text{(B-16)}$$

For longer times and larger values of σ, Eq. B-16 would have to be modified. In this study, interest is confined to the local ($\leq$ 100 m) exposure field for workers.

Integrated air concentration (IAC) was calculated for downwind distances from the source in grid increments of 2 m, with the first cell surrounding the source assigned a distance value of 1 m. Calculated IACs are shown in Figures B.1 to B.3 for the three meteorological scenarios and a receptor height of 1 m. IAC values were calculated for three receptor heights: 1.6 m (nostril height), 1.0 m (waist height), and 0.2 m (boot-top height). Table B.1 shows the maximum IAC encountered for these three receptor heights, along with the distance and cloud travel time. Table B.2 also shows the values at 1 m downwind for the three heights. All downwind exposures and distances are along the centerline of cloud travel. Exposure values decrease as one moves laterally from the centerline.

The dose via inhalation was estimated from the IAC as follows:

$$D = IAC \cdot V \cdot DF \qquad \text{(B-17)}$$

where

D = dose equivalent via inhalation (rem)
IAC = integrated air concentration (μCi-s/m^3)
V = minute volume (expressed in m^3/s)
DF = inhalation dose factor (rem/μCi)

The minute volume for adults under conditions of moderate activity is 20 L/min, which is equivalent to

$$V = 20\ (L/min)\ (1\ m^3/10^3 L)\ (1\ min/60\ s)$$
$$V = 3.33 \times 10^{-4}\ m^3/s \qquad \text{(B-18)}$$

Substitution of Eq. B-18 in Eq. B-17 leads to

$$D = 3.33 \times 10^{-4}\ IAC \cdot DF \qquad \text{(B-19)}$$

The dose via submersion is directly obtainable from the IAC using the appropriate submersion dose factor. Inhalation and submersion dose conversion factors for the gaseous EOR radionuclides of interest are presented in Table B.3.

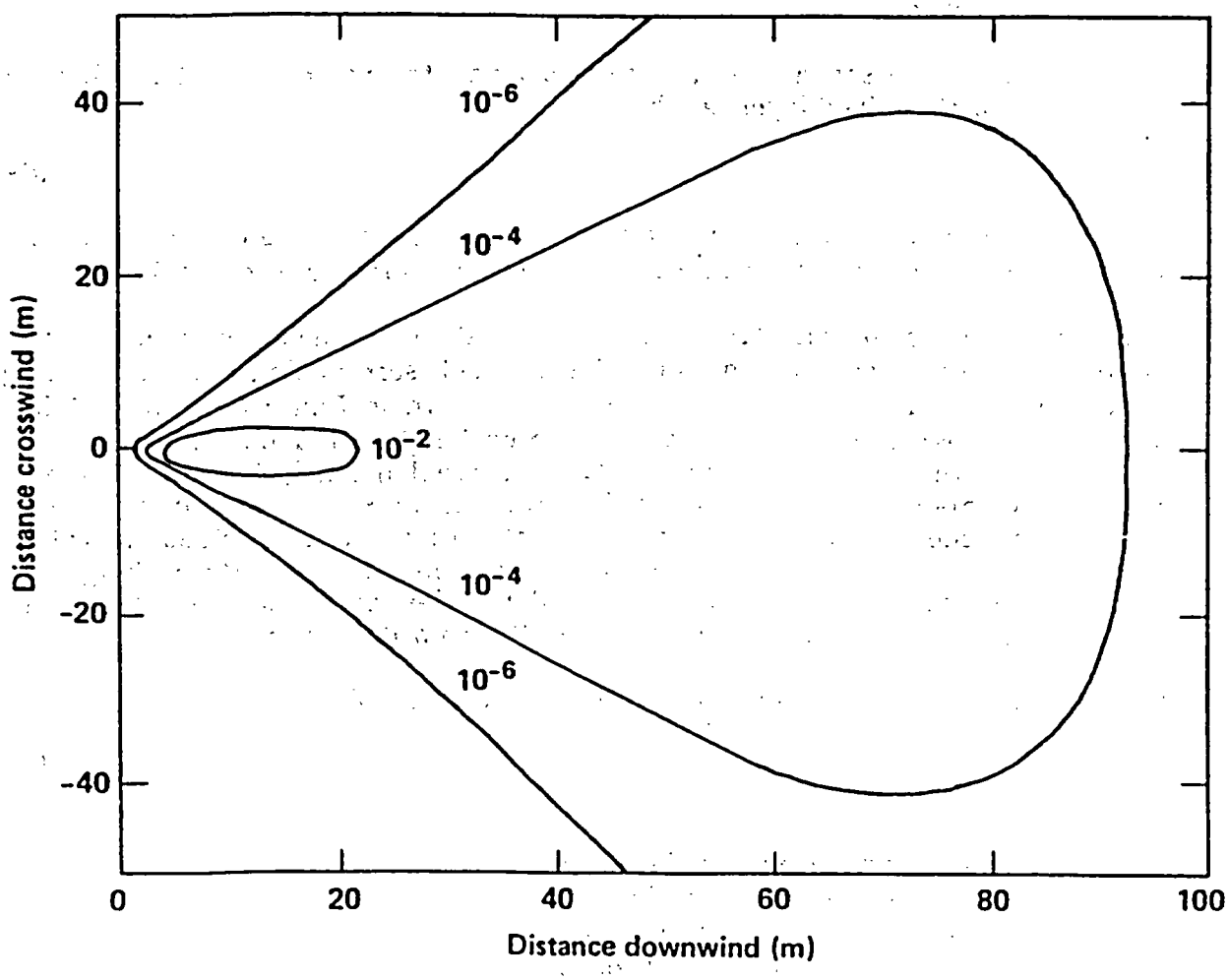

Figure B.1. Integrated air concentrations (Ci-s/m^3) at 1-m height from ground-level release of 1 Ci under meteorological scenario I (unstable; low wind speed).

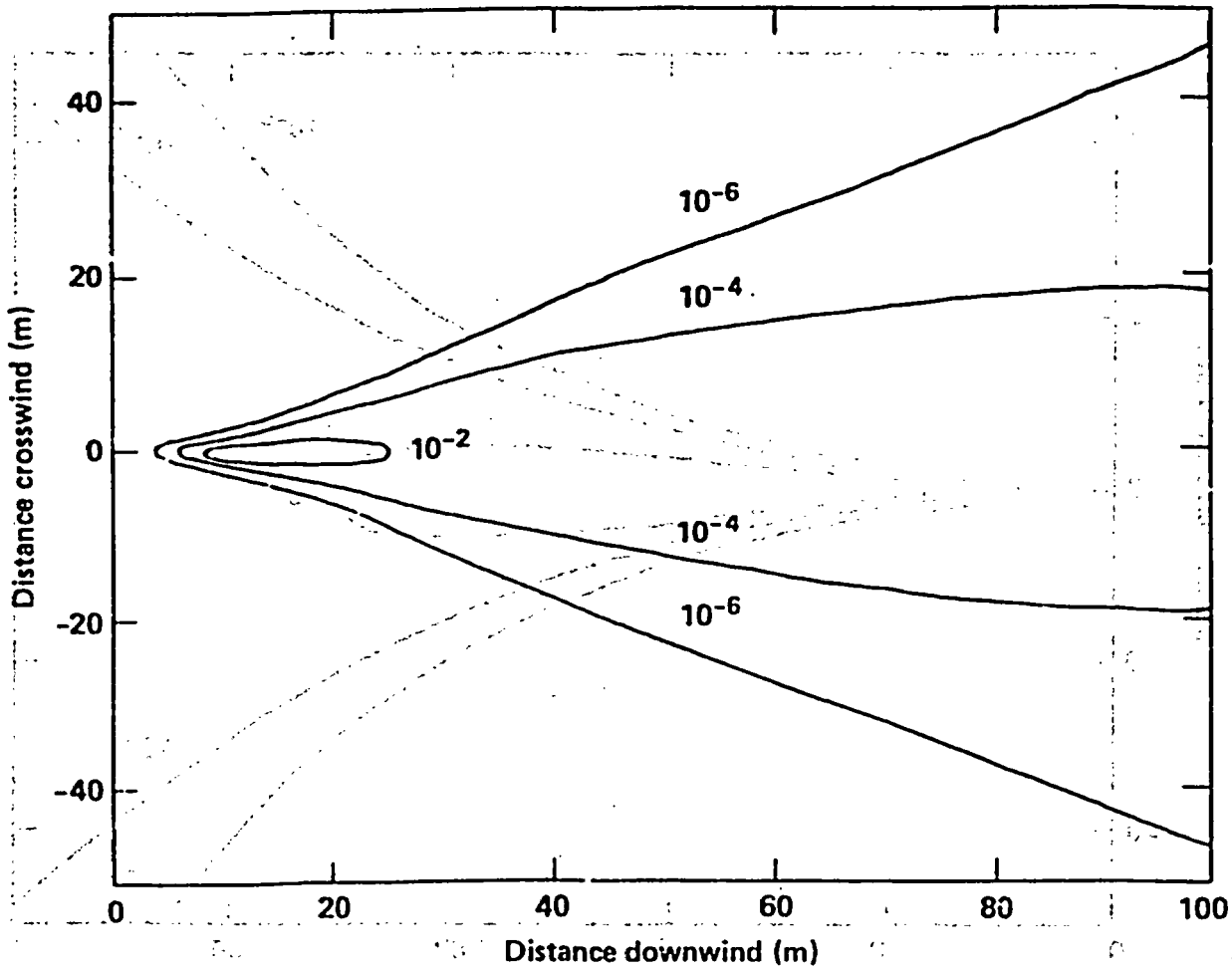

Figure B.2. Integrated air concentrations (Ci-s/m^3) at 1-m height from ground-level release of 1 Ci under meteorological scenario 2 (neutral; high wind speed).

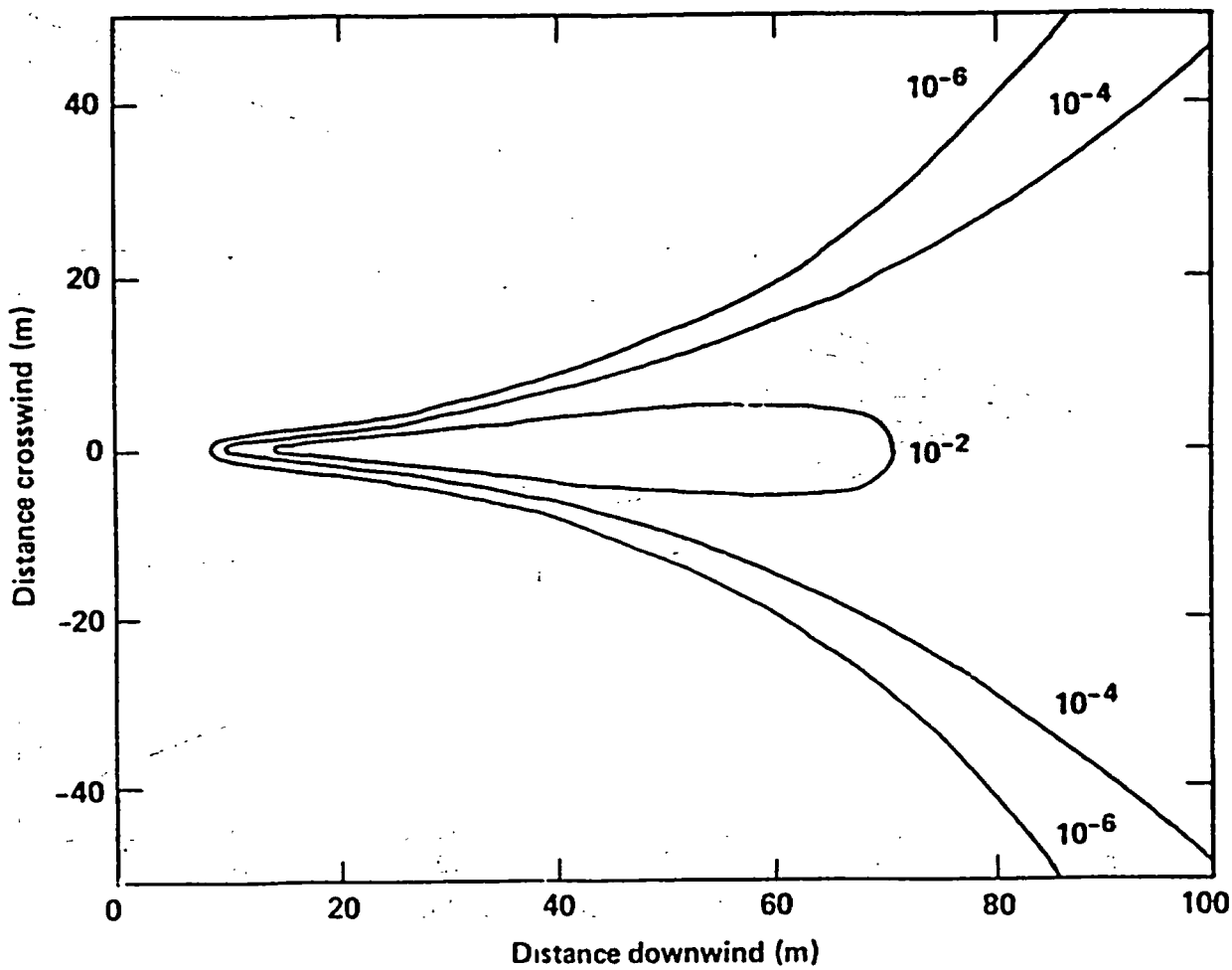

Figure B.3. Integrated air concentrations (Ci-s/m^3) at 1-m height from ground-level release of 1 Ci under meteorological scenario 3 (stable; low wind speed).

Table B.1 Maximum integrated air concentrations near the surface downwind from a unit surface release.

Receptor ht m	1. Unstable Max IAC s/m^3	Distance[a] m	Time[b] s	2. Neutral Max IAC s/m^3	Distance[a] m	Time[b] s	3. Stable Max IAC s/m^3	Distance[a] m	Time[b] s
1.6	0.065	8	6	0.014	22	3	0.13	36	51
1.0	0.015	6	5	0.035	14	2	0.33	24	40
0.2	3.2	1	1	0.76	2	0.4	8.11	2	10

[a] Distance refers to the downwind distance along the centerline of cloud travel.

[b] Time refers to the time maximum IAC occurs at that distance and height.

Table B.2 Integrated air concentrations near the surface 1 meter downwind from a unit surface release.

Receptor ht m	1. Unstable IAC, s/m^3	2. Neutral IAC, s/m^3	3. Stable IAC, s/m^3
1.6	0	0	0
1.0	1.9×10^{-11}	0	0
0.2	3.2	0.75	3.4

B.2 Dose to members of the public

Dose estimates to the public involve consideration of a much greater area than the local field considered previously for workers. The PATRIC code (Lange, 1978), a particle-in-cell model, was used to calculate IACs at distances far downwind (tens of kilometers) from the accident site. PATRIC estimates the transport and diffusion of the released airborne material by following a large number of marker particles. The resultant IAC over any time interval is determined by the cumulative spatial distributions of the particles over that interval.

In our simulations, a horizontal grid cell resolution of 2.5 km was used. In the vertical direction, grid cell resolution was 25 m for unstable and neutral meteorological conditions and 7.5 m for the stable case. These values for the vertical direction were based on estimates of half the depth of the atmospheric surface layer. IAC values were estimated for a receptor height of 1.6 m (nostril height). A short burst of particles of 1-Ci activity was released to simulate a puff, which was tracked until it passed beyond 80 km downwind. A steady wind-direction

Table B.3 Inhalation and submersion dose rate factors for gaseous forms of EOR radionuclides.[a]

Radioisotope	Exposure mode	Reference organ	Dose equivalent rate rem per μCi-s/m^3
H-3 HT, hydrocarbons	Submersion	Lung[b]	1.0×10^{-11}
HTO	Inhalation	Whole body	2.1×10^{-8}[c]
C-14 $^{14}CO_2$	Inhalation	Whole body	7.9×10^{-9}[c]
Ar-37	Submersion	Lung	3.9×10^{-12}
Ar-39	Submersion	Skin	3.9×10^{-8}[d]
	Submersion	Whole body	9.2×10^{-11}[de]
Kr-85	Submersion	Skin	4.8×10^{-8}[d]
	Submersion	Whole body	6.1×10^{-10}[de]
Xe-133	Submersion	Skin	2.0×10^{-8}[d]
	Submersion	Whole body	6.9×10^{-9}[de]

[a] Source: ICRP Publication 30 (ICRP, 1979; ICRP, 1980; ICRP, 1982).

[b] The dose equivalent in the whole body via submersion is zero.

[c] The inhalation dose factor for the whole body is 6.3×10^{-5} rem/μCi for HTO (ICRP, 1974) and 2.4×10^{-5} rem/μCi for $^{14}CO_2$ (ICRP, 1982).

[d] The submersion doses assume submersion in a semi-infinite cloud. A finite-plume correction should be applied at distances close to the source.

[e] Value from Soldat et al. (1973).

was used (270°). A steady wind-speed profile, using given values at a standard measurement height (6 m), was constructed from parameters based on atmospheric stability:

$$u_z = u_6 \left(\frac{z}{6}\right)^p , \qquad \text{(B-20)}$$

where

p = 0.07, 0.15, and 0.55 for unstable, neutral, and stable conditions, respectively.

Plots of the IAC for the three meteorological scenarios are shown in Figures B.4, B.5, and B.6.

The collective dose is the dose summed over all the individuals in the population:

$$S = DN \qquad \text{(B-21)}$$

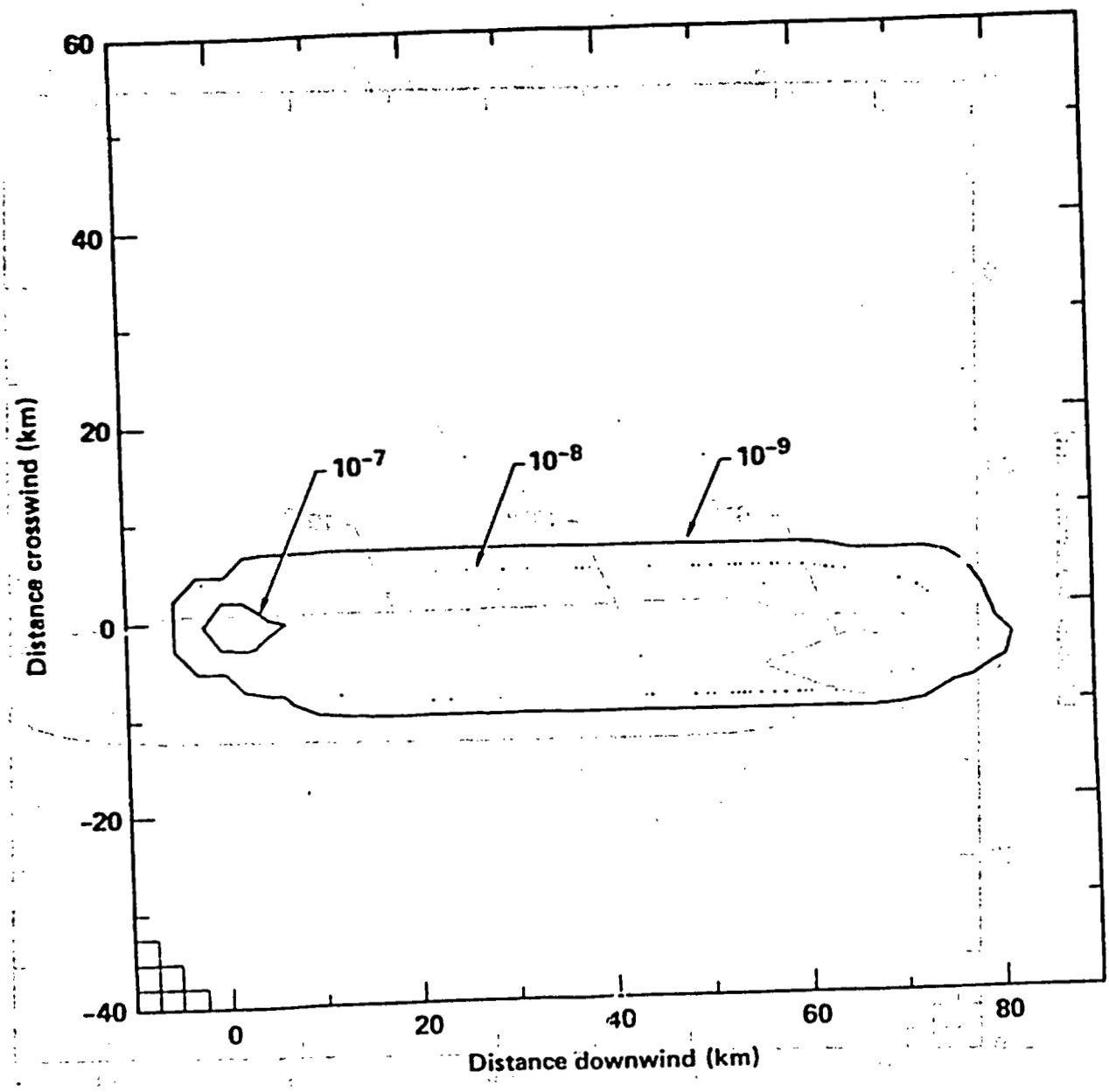

Figure B.4. Integrated air concentrations (Ci-s/m^3) at 1.6-m height from ground-level release of 1 Ci under meteorological scenario 1 (unstable; low wind speed); 2.5-km squares in lower left corner indicate grid resolution.

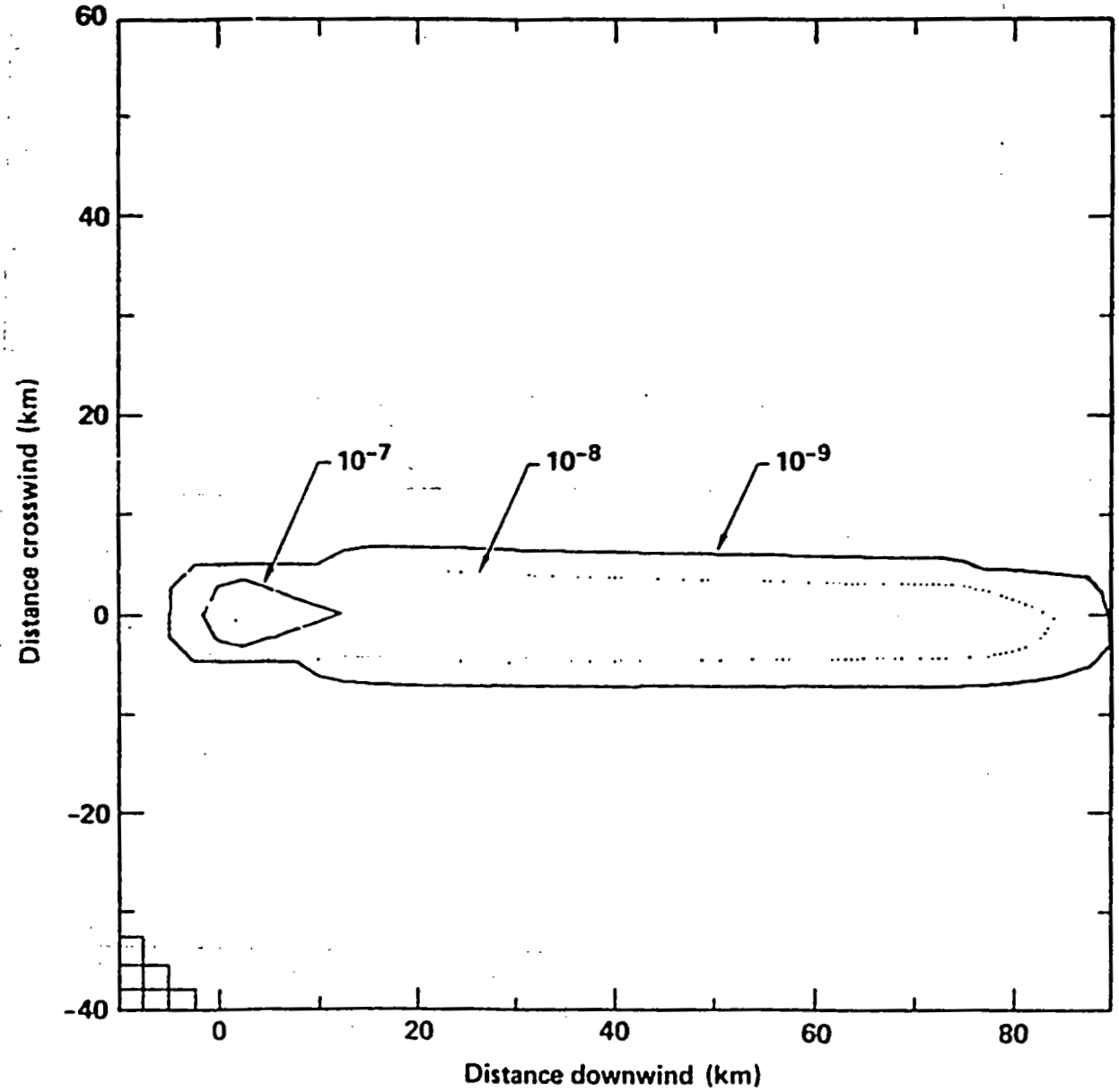

Figure B.5. Integrated air concentrations (Ci-s/m^3) at 1.6-m height from ground-level release of 1 Ci under meteorological scenario 2 (neutral; high wind speed); 2.5-km squares in lower left corner indicate grid resolution.

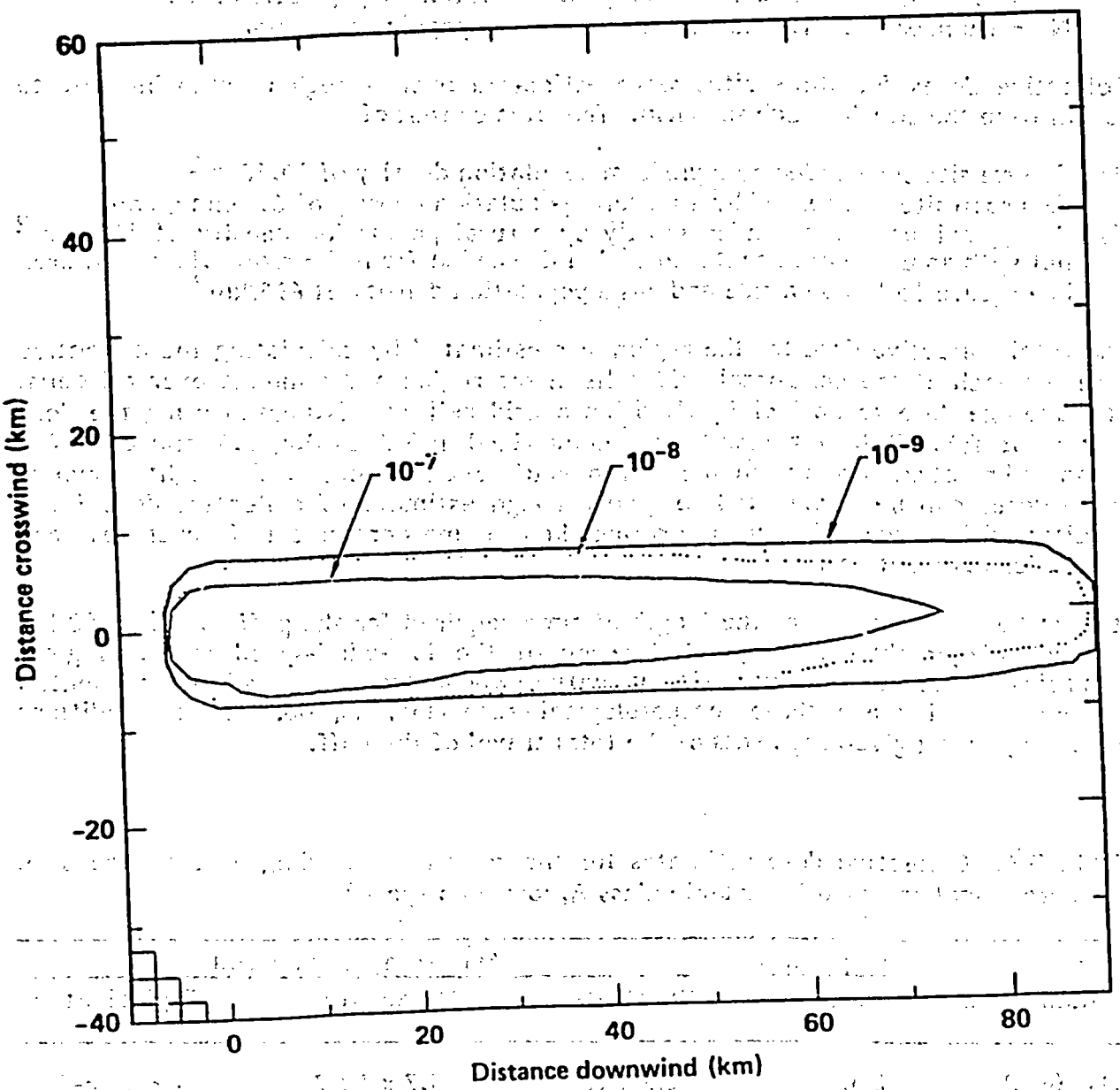

Figure B.6. Integrated air concentrations (Ci-s/m^3) at 1.6-m height from ground-level release of 1 Ci under meteorological scenario 3 (stable; low wind speed); 2.5-km squares in lower left corner indicate grid resolution.

where

S = Collective dose (man-rem)
D = Average dose to an individual in a population or population group
N = Number of individuals in a population or population group.

Collective doses for three sites were estimated over a region extending out to 80 km from the site for each scenario. The sites consist of

(1) A rural site surrounded by a uniform population density of 13.5/km^2,
(2) An urban site surrounded by a uniform population density of 695/km^2, and
(3) A "typical site", surrounded mostly by a rural population density of 13.5/km^2 but with an urban area centered 32 km downwind from the site. The urban area is a square 15 km on a side and has a population density of 695/km^2.

The total collective dose for the region was estimated by calculating the collective dose for each of the horizontal grid cells in the region and summing over all cells. The average dose to an individual within a grid cell was estimated using the dose factors of Table B.3. The results are summarized in Tables B.4, B.5, and B.6. The urban and rural sites represent the extremes of collective doses. A weighted sum of these doses can be formulated to give a rough estimate of collective dose for a specific siting scenario, incorporating known meteorological frequencies and population distributions.

It must be emphasized that the length of time required for the puff to travel 80 km is much longer than the typical duration of the low-wind-speed meteorological scenarios. Thus, a more realistic simulation would probably require a sequential combination of these three meteorological scenarios, appropriate to conditions occurring during given segments of the total travel of the puff.

Table B.4. Collective dose estimates for the region surrounding a reference site, meteorological scenario 1 (unstable class A, low wind speed).

Radionuclide	Reference organ	Man-rem/μCi released		
		Rural site	Urban site	Typical site[a]
H-3 (gas)	Lung	2.9×10^{-8}	5.7×10^{-7}	1.2×10^{-7}
H-3 (HTO)	Whole body	5.8×10^{-5}	1.2×10^{-3}	2.5×10^{-4}
C-14 ($^{14}CO_2$)	Whole body	2.2×10^{-5}	4.4×10^{-4}	9.3×10^{-5}
Ar-37	Lung	1.1×10^{-8}	2.2×10^{-7}	4.6×10^{-8}
Ar-39	Skin	1.1×10^{-4}	2.2×10^{-3}	4.6×10^{-4}
Kr-85	Skin	1.4×10^{-4}	2.7×10^{-3}	5.7×10^{-4}
Xe-133	Skin	5.5×10^{-5}	1.1×10^{-3}	2.3×10^{-4}

[a] Based on a total population exposure of 1.2×10^4 man-μCi-s/m^3.

Table B.5 Collective dose estimates for the region surrounding a reference site, meteorological scenario 2 (neutral class D, high wind speed).

Radionuclide	Reference organ	Man-rem/μCi released		
		Rural site	Urban site	Typical site[a]
H-3 (gas)	Lung	1.0×10^{-8}	2.0×10^{-7}	3.5×10^{-8}
H-3 (HTO)	Whole body	2.1×10^{-5}	4.1×10^{-4}	7.1×10^{-5}
C-14 ($^{14}CO_2$)	Whole body	7.8×10^{-6}	1.5×10^{-4}	2.7×10^{-5}
Ar-37	Lung	3.9×10^{-9}	7.7×10^{-8}	1.3×10^{-8}
Ar-39	Skin	3.9×10^{-5}	7.7×10^{-4}	1.3×10^{-4}
Kr-85	Skin	4.8×10^{-5}	9.5×10^{-4}	1.6×10^{-4}
Xe-133	Skin	1.9×10^{-5}	3.8×10^{-4}	6.6×10^{-5}

[a] Based on a total population exposure of 3.4×10^3 man-μCi-s/m^3.

Table B.6 Collective dose estimates for the region surrounding a reference site, meteorological scenario 3 (stable class F, low wind speed).

Radionuclide	Reference organ	Man-rem/μCi released		
		Rural site	Urban site	Typical site[a]
H-3 (gas)	Lung	2.5×10^{-7}	5.0×10^{-6}	9.4×10^{-7}
H-3 (HTO)	Whole body	5.2×10^{-4}	1.0×10^{-2}	1.9×10^{-3}
C-14 ($^{14}CO_2$)	Whole body	2.0×10^{-4}	3.9×10^{-3}	7.3×10^{-4}
Ar-37	Lung	9.7×10^{-8}	1.9×10^{-6}	3.6×10^{-7}
Ar-39	Skin	9.7×10^{-4}	1.9×10^{-2}	3.6×10^{-3}
Kr-85	Skin	1.2×10^{-3}	2.4×10^{-2}	4.4×10^{-3}
Xe-133	Skin	4.8×10^{-4}	9.6×10^{-3}	1.8×10^{-3}

[a] Based on a total population exposure of 3.4×10^3 man-μCi-s/m^3.

APPENDIX C. ESTIMATION OF THE DOSAGE FROM THE ENTRY OF RADIONUCLIDES INTO AN AQUIFER

The exposures from the entry of radioactive traces into a freshwater aquifer are evaluated by estimating the dose that could result from ingesting water drawn from the aquifer. The aquifer is assumed to be a homogeneous, isotropic, porous medium of infinite lateral (x,y) extent. Solution of the advection dispersion equation in three dimensions leads to an expression for the concentration as a function of time following an instantaneous release in a homogeneous isotropic medium (Codell et al., 1982):

$$C(x,y,z,t) = \frac{1}{n_e B} X(x,t)Y(y,t)Z(z,t) \qquad \text{(C-1)}$$

where

- $C(x,y,z,t)$ = concentration for an instantaneous release of 1 μCi(μCi/m^3)
- t = time after release (d)
- x = distance in direction of flow (m)
- y = transverse distance (m)
- z = vertical distance (m)
- n_e = effective porosity
- B = retardation factor

The water-supply well is assumed to be 1.0 km downgradient from the injection well and is assumed to draw water from the full height of the aquifer. To calculate the concentration in the aquifer at some point downgradient from the release, we assume that the source is a point source at $(0,0,z_s)$. We then calculate the vertically averaged concentration, which is equivalent to a vertical line source of length, h, the height of the aquifer. The resulting concentrations then satisfy the conditions that the water-supply well draws water from the full height of the aquifer.

The solution of Eq. C-1 for these conditions is

$$C = \frac{1}{n_e B} X_1 Y_1 Z_1 \exp(-\lambda t) \qquad \text{(C-2)}$$

where

$$X_1 = (1/\sqrt{4\pi\alpha_x vt/B})\exp[-(x-vt/B)^2/(4\pi\alpha_x vt/B)] \qquad \text{(C-3)}$$

$$Y_1 = (1/\sqrt{4\pi\alpha_y vt/B})\exp[-(y^2/(4\pi\alpha_y vt/B)] \qquad \text{(C-4)}$$

$$Z_1 = 1/h \qquad \text{(C-5)}$$

- λ = radioactive decay constant (d^{-1})
- α_x = longitudinal dispersivity (m)
- v = interstitial water flow velocity (m/d)
- α_y = transverse dispersivity (m)
- h = height of aquifer (m)

The rate of ingestion of activity via ingestion of well water is given by

$$I(x,t) = JC(x,t)/10^3 \qquad \text{(C-6)}$$

where

- I = daily intake of activity (μCi/d)
- J = rate of intake of water (L/d)

The annual intake of activity via ingestion of well water is given by summing the daily intakes over a year.

$$I_A(x,t) = \sum_{i=0}^{364} JC(x,t+i) \qquad \text{(C-7)}$$

where

$I_A(x,t)$ = annual intake of activity (μCi/y)

The dose rate via ingestion of well water is given by

$$D_A = I_A(x,t) \cdot DF \qquad \text{(C-8)}$$

where

D_A = committed dose equivalent from a year's intake (rem/y)
DF = committed dose equivalent per unit activity ingested (rem/μCi)

Doses via ingestion of well water were estimated by the above approach assuming entry of 1 μCi of EOR radionuclides into the aquifer. Parameter values assumed for the calculations were

v = 0.2 m/d (240 ft/y)	B = 1.0
α_x = 40 m	h = 80 m
α_y = 8.0 m	J = 2.0 L/d

The radionuclides, dose factors, and reference organs are shown in Table C.1. Doses were estimated for various distances of the well downgradient from the point of release on the centerline. The results are presented in Table C.2. Detailed results for HTO and ^{90}Sr are presented in Tables C.3 and C.4.

It is also of interest to estimate the dosage from other uses of the contaminated well water. In this report we consider the dosage from ingesting fresh produce irrigated with the water, and milk and meat from cattle that drank the water and ingested forage that was irrigated with the water.

The dosage from ingestion of produce and animal products is compared with that from ingestion of water by comparing the daily intakes via these pathways. The concentrations in produce and animal products were calculated using the terrestrial foodchain model published in Regulatory Guide 1.109 (USNRC, 1977). The basic equation for estimating transfer into vegetation is

$$C_v = \delta \left[\frac{r\{1-\exp-(\lambda_w + \lambda_r)t_e\}}{Y_v} + \frac{B_v\{1-\exp-(\lambda_s + \lambda_r)t_b\}}{P(\lambda_s + \lambda_r)} \right] \qquad \text{(C-9)}$$

where

C_v = concentration of radionuclide in vegetation (μCi/kg),
δ = deposition rate (μCi/m^2 • d),
r = fraction of the depositing radionuclide that is intercepted by standing vegetation (unitless),
Y_v = standing crop biomass (kg/m^2),

Table C.1 Dosimetric data for EOR radionuclides.

Radionuclide	Half life[a] (d)	Reference[b] organ	Committed dose equivalent[b] (rem/μCi ingested)
H-3	4.5×10^{3}	Total body	6.29×10^{-5}
C-14	2.09×10^{6}	Total body	2.07×10^{-3}[c]
Na-22	949	Bone surface	2.04×10^{-2}
P-32	14.3	Bone marrow	3.00×10^{-2}
S-35	87.2	Total body	2.11×10^{-3}
Cl-36	1.1×10^{8}	Gonads	2.96×10^{-3}
Ca-45	163	Bone surface	1.92×10^{-2}
Cr-51	27.7	LLI wall	9.25×10^{-2}
Fe-55	986	Spleen	2.07×10^{-3}
Fe-59	44.6	LLI wall	3.11×10^{-2}
Co-57	271	LLI wall	4.07×10^{-3}
Co-58	71.3	LLI wall	1.48×10^{-2}
Co-60	1920	LLI wall	4.07×10^{-2}
Ni-63	3.65×10^{4}	LLI wall	3.40×10^{-3}
Zn-65	244	LLI wall	1.85×10^{-2}
Sr-85	65.2	LLI wall	5.55×10^{-3}
Sr-89	50.5	LLI wall	7.77×10^{-2}
Sr-90	1.06×10^{4}	Bone surface	1.55×10^{0}
Te-99	7.78×10^{7}	Stomach wall	1.26×10^{-2}
Ag-110 m	252	Gonads	4.07×10^{-2}
I-125	59.7	Thyroid	1.26×10^{0}
I-131	8.04	Thyroid	1.07×10^{0}

[a] Source: Chart of the Nuclides (1973).

[b] Source: ICRP Publication 30 (ICRP, 1979; ICRP, 1980; ICRP, 1982).

[c] Dose factor for labeled organic compounds.

λ_w = rate constant for the removal of surface deposited material from vegetation (d^{-1}),
λ_r = radioactive decay constant (d^{-1}),
t_e = growing season during which vegetation is exposed to depositing radionuclides (d)
B_v = plant/soil concentration ratio
P = density of surface soil layer assuming root zone of 15 cm (kg/m^2)
λ_s = rate constant for migration of radionuclide out of root zone (d^{-1}),
t_b = period over which the depositing radionuclide accumulates in surface layer of soil (d).

In the evaluation of C_v, we distinguish between C_{v1}, the concentration in fresh produce expressed on a wet weight basis and C_{v2}, the concentration in forage crops expressed on a dry weight basis.

Table C.2 Maximum annual dose equivalent via ingestion of water from an aquifer contaminated by a release of 1 μCi.[a]

Radioisotope	Reference organ	mrem/y per μCi released[b] Distance of well from source			
		0.5 km	1.0 km	3.0 km	5.0 km
H-3 (HTO)	Whole body	1.6×10^{-8}	5.2×10^{-9}	3.8×10^{-10}	5.2×10^{-11}
C-14	Whole body	7.2×10^{-7}	3.3×10^{-7}	1.1×10^{-7}	6.7×10^{-8}
Na-22	Bone surface	1.7×10^{-6}	1.5×10^{-7}	8.2×10^{-11}	7.8×10^{-14}
P-32	Bone marrow				
S-35	Whole body	3.6×10^{-11}	4.3×10^{-16}		
Cl-36	Gonads	6.3×10^{-7}	4.7×10^{-7}	1.6×10^{-7}	9.6×10^{-8}
Ca-45	Bone surface	1.3×10^{-8}	6.9×10^{-12}	2.8×10^{-24}	
Cr-51	LLI wall	1.6×10^{-16}	1.7×10^{-26}		
Fe-55	Spleen	1.8×10^{-7}	1.7×10^{-8}	1.2×10^{-11}	1.4×10^{-14}
Fe-59	LLI wall	1.5×10^{-12}	4.2×10^{-20}		
Co-57	LLI wall	2.1×10^{-8}	8.2×10^{-11}	2.7×10^{-19}	
Co-58	LLI wall	5.8×10^{-11}	1.5×10^{-16}		
Co-60	LLI wall	6.5×10^{-6}	1.4×10^{-6}	1.5×10^{-8}	3.2×10^{-10}
Ni-63	LLI wall	1.1×10^{-6}	5.0×10^{-7}	1.4×10^{-7}	7.0×10^{-8}
Zn-65	LLI wall	6.7×10^{-8}	1.5×10^{-10}	1.2×10^{-19}	1.1×10^{-28}
Sr-85	LLI wall	1.1×10^{-11}	1.2×10^{-17}		
Sr-89	LLI wall	1.4×10^{-11}	1.0×10^{-18}		
Sr-90	Bone surface	4.7×10^{-4}	1.9×10^{-4}	3.3×10^{-5}	1.0×10^{-5}
Tc-99	Stomach wall	4.4×10^{-6}	2.0×10^{-6}	6.8×10^{-7}	4.1×10^{-7}
Ag-110m	Gonads	1.7×10^{-7}	4.5×10^{-10}	5.6×10^{-19}	
I-125	Thyroid	1.1×10^{-9}	5.4×10^{-16}		
I-131	Thyroid	8.4×10^{-24}			

[a] Committed dose equivalent from drinking the water for one year.

[b] A blank space signifies that the calculated dose equivalent is $<10^{-30}$ mrem/y.

The expressions used to estimate the transfer of a radionuclide from forage and water to milk and meat are

$$C_m = (C_{v2} Q_m + C_w Q_{w1}) F_m \qquad \text{(C-10)}$$

$$C_f = (C_{v2} Q_f + C_w Q_{w2}) F_f \qquad \text{(C-11)}$$

where

C_m = concentration of radionuclide in milk (μCi/L),
Q_m = daily intake of dry forage by dairy cows (kg/d),
C_w = concentration of radionuclide in water (μCi/L),
Q_{w1} = daily intake of water by dairy cattle (L/d),
F_m = equilibrium transfer coefficient to milk (the fraction of the daily intake by cows that is transferred to a liter of milk, d/L),
Q_f = daily intake of dry forage by beef cattle (kg/d),
Q_{w2} = daily intake of water by beef cattle (L/d),
F_f = equilibrium transfer coefficient to beef (the fraction of the daily intake by cattle that is transferred to a kg of muscle, d/kg).

TABLE C.3

DOSE FROM DRINKING WATER FROM A WELL THAT INTERSECTS AN AQUIFER THAT RECEIVES A PULSE OF 1 MICROCURIE.

RADIONUCLIDE: H3

GROUNDWATER VELOCITY	0.2 M/D
AQUIFER HEIGHT	80.0 M
LONGITUDINAL DISPERSIVITY	40.0 M
TRANSVERSE DISPERSIVITY	8.0 M

DOSE RATE (MILLIRAD/YR PER MICROCURIE RELEASED) TO TOTAL BODY

TIME (D)	DISTANCE (KM)				
	0.5	1.0	2.0	3.0	5.0
200	2.27E-12	2.77E-28	1.39E-95	9.96-211	0.
400	8.83E-11	5.56E-22	2.06E-70	4.21-154	0.
600	7.14E-10	2.56E-18	1.04E-55	5.45-121	0.
800	2.59E-09	6.17E-16	4.43E-46	2.81E-99	2.42-275
1000	5.81E-09	2.83E-14	2.68E-39	5.90E-84	1.10-232
1500	1.41E-08	9.17E-12	8.59E-29	4.07E-60	3.60-166
2000	1.57E-08	1.93E-10	7.90E-23	1.84E-46	6.87-128
3000	8.21E-09	2.68E-09	1.92E-16	1.49E-31	1.08E-85
4000	2.77E-09	5.23E-09	2.98E-13	1.02E-23	4.45E-63
5000	7.89E-10	4.40E-09	1.71E-11	5.16E-19	4.23E-49
6000	2.08E-10	2.41E-09	1.69E-10	5.59E-16	1.09E-39
7000	5.24E-11	1.03E-09	5.91E-10	6.08E-14	5.47E-33
8000	1.29E-11	3.79E-10	1.07E-09	1.51E-12	4.98E-28
9000	3.13E-12	1.26E-10	1.23E-09	1.37E-11	2.94E-24
10000	7.55E-13	3.90E-11	1.04E-09	6.08E-11	2.47E-21
11000	1.81E-13	1.15E-11	6.98E-10	1.60E-10	4.92E-19
12000	4.32E-14	3.26E-12	3.94E-10	2.84E-10	3.29E-17
13000	1.03E-14	8.98E-13	1.95E-10	3.71E-10	9.43E-16
14000	2.45E-15	2.42E-13	8.71E-11	3.82E-10	1.38E-14
15000	5.82E-16	6.43E-14	3.57E-11	3.23E-10	1.18E-13
16000	1.39E-16	1.68E-14	1.37E-11	2.34E-10	6.50E-13
17000	3.30E-17	4.36E-15	4.97E-12	1.49E-10	2.48E-12
18000	7.84E-18	1.12E-15	1.72E-12	8.52E-11	7.01E-12
19000	1.87E-18	2.85E-16	5.74E-13	4.45E-11	1.53E-11
20000	4.45E-19	7.21E-17	1.85E-13	2.15E-11	2.68E-11
21000	1.06E-19	1.81E-17	5.81E-14	9.71E-12	3.90E-11
22000	2.53E-20	4.55E-18	1.78E-14	4.15E-12	4.83E-11
23000	6.03E-21	1.14E-18	5.33E-15	1.68E-12	5.18E-11
24000	1.44E-21	2.83E-19	1.57E-15	6.55E-13	4.92E-11
25000	3.44E-22	7.02E-20	4.54E-16	2.45E-13	4.18E-11
26000	8.22E-23	1.74E-20	1.30E-16	8.88E-14	3.23E-11
27000	1.97E-23	4.30E-21	3.66E-17	3.12E-14	2.29E-11
28000	4.71E-24	1.06E-21	1.02E-17	1.07E-14	1.51E-11
29000	1.13E-24	2.61E-22	2.82E-18	3.57E-15	9.25E-12
30000	2.70E-25	6.43E-23	7.71E-19	1.17E-15	5.33E-12
31000	6.47E-26	1.58E-23	2.09E-19	3.74E-16	2.91E-12
32000	1.55E-26	3.88E-24	5.65E-20	1.18E-16	1.51E-12
33000	3.72E-27	9.52E-25	1.51E-20	3.65E-17	7.47E-13
34000	8.94E-28	2.33E-25	4.03E-21	1.12E-17	3.55E-13
35000	2.15E-28	5.71E-26	1.07E-21	3.36E-18	1.62E-13

TABLE C.4

DOSE FROM DRINKING WATER FROM A WELL THAT INTERSECTS AN AQUIFER THAT RECEIVES A PULSE OF 1 MICROCURIE

RADIONUCLIDE SR90

GROUNDWATER VELOCITY	0.2 M/D
AQUIFER HEIGHT	80.0 M
LONGITUDINAL DISPERSIVITY	40.0 M
TRANSVERSE DISPERSIVITY	8.0 M

DOSE RATE (MILLIRAD/YR PER MICROCURIE RELEASED) TO BONE SURFACE

TIME (D)	DISTANCE (KM)				
	0.5	1.0	2.0	3.0	5.0
200	5.87E-08	7.18E-24	3.60E-91	2.58-206	0.
400	2.32E-06	1.46E-17	5.42E-66	1.11-149	0.
600	1.90E-05	6.86E-14	2.80E-51	1.46-116	0
800	7.01E-05	1.68E-11	1.21E-41	7.67E-95	6.61-271
1000	1.60E-04	7.83E-10	7.44E-35	1.64E-79	3.05-228
1500	4.03E-04	2.64E-07	2.49E-24	1.18E-55	1.04-161
2000	4.68E-04	5.79E-06	2.39E-18	5.58E-42	2.08-123
3000	2.68E-04	8.78E-05	6.31E-12	4.93E-27	3.58E-81
4000	9.88E-05	1.87E-04	1.07E-08	3.67E-19	1.61E-58
5000	3.07E-05	1.71E-04	6.69E-07	2.03E-14	1.67E-44
6000	8.84E-06	1.03E-04	7.21E-06	2.39E-11	4.70E-35
7000	2.43E-06	4.80E-05	2.75E-05	2.84E-09	2.57E-28
8000	6.55E-07	1.93E-05	5.43E-05	7.69E-08	2.55E-23
9000	1.74E-07	6.99E-06	6.85E-05	7.61E-07	1.64E-19
10000	4.58E-08	2.36E-06	6.31E-05	3.70E-06	1.51E-16
11000	1.20E-08	7.61E-07	4.63E-05	1.06E-05	3.28E-14
12000	3.12E-09	2.36E-07	2.86E-05	2.06E-05	2.40E-12
13000	8.13E-10	7.11E-08	1.55E-05	2.94E-05	7.49E-11
14000	2.12E-10	2.10E-08	7.53E-06	3.30E-05	1.20E-09
15000	5.50E-11	6.07E-09	3.37E-06	3.06E-05	1.12E-08
16000	1.43E-11	1.74E-09	1.41E-06	2.42E-05	6.72E-08
17000	3.72E-12	4.91E-10	5.61E-07	1.68E-05	2.81E-07
18000	9.66E-13	1.38E-10	2.12E-07	1.05E-05	8.65E-07
19000	2.51E-13	3.83E-11	7.73E-08	5.99E-06	2.06E-06
20000	6.54E-14	1.06E-11	2.72E-08	3.16E-06	3.95E-06
21000	1.70E-14	2.92E-12	9.33E-09	1.56E-06	6.28E-06
22000	4.44E-15	7.98E-13	3.12E-09	7.28E-07	8.49E-06
23000	1.16E-15	2.18E-13	1.02E-09	3.23E-07	9.95E-06
24000	3.02E-16	5.93E-14	3.29E-10	1.37E-07	1.03E-05
25000	7.88E-17	1.61E-14	1.04E-10	5.62E-08	9.59E-06
26000	2.06E-17	4.35E-15	3.25E-11	2.22E-08	8.19E-06
27000	5.38E-18	1.18E-15	1.00E-11	8.54E-09	6.28E-06
28000	1.41E-18	3.17E-16	3.05E-12	3.19E-09	4.51E-06
29000	3.68E-19	8.53E-17	9.20E-13	1.16E-09	3.02E-06
30000	9.63E-20	2.29E-17	2.75E-13	4.16E-10	1.90E-06
31000	2.52E-20	6.16E-18	8.16E-14	1.46E-10	1.13E-06
32000	6.61E-21	1.65E-18	2.40E-14	5.01E-11	6.43E-07
33000	1.73E-21	4.43E-19	7.03E-15	1.70E-11	3.48E-07
34000	4.54E-22	1.19E-19	2.05E-15	5.67E-12	1.80E-07
35000	1.19E-22	3.17E-20	5.92E-16	1.87E-12	9.01E-08

The quantity of radionuclide ingested daily is calculated as the product of the concentration in food and the appropriate usage factor, U, expressed in kg/d or L/d. Usage factors for leafy vegetables (U_l), milk (U_m), and beef (U_f) are listed in Table C.5. Thus, the intake from ingestion of leafy vegetables, milk and meat are given, respectively, by

$$I_l = U_l C_{vl} \quad \text{(C-12)}$$
$$I_m = U_m C_m \quad \text{(C-13)}$$
$$I_f = U_f C_f \quad \text{(C-14)}$$

where

I_l = intake of radionuclide from ingestion of leafy vegetables (μCi/d),
U_l = intake of leafy vegetables (kg/d),
I_m = intake of radionuclide from ingestion of milk (μCi/d),
U_m = intake of milk (L/d),
I_f = intake of radionulcide from ingestion of beef,
U_f = intake of beef (kg/d).

The radionuclide-independent parameter values for the calculations are listed in Table C.5. The values are mean, median, or midrange values as listed by Hoffman et al. (1982). Because r and Y_v are correlated, they have been combined into a single parameter, r/Y_v, the normalized interception fraction.

Strontium-90 was selected as the most suitable reference radionuclide for comparing the intakes through water and foods contaminated by radionuclides originating in well water. Sr-90 was singled out because it was identified as the radionuclide that could contribute most to the dose from drinking contaminated well water. In addition, it is sufficiently long lived that transfer from water to foods takes place without substantial losses from radioactive decay; and the transfer factors, i.e., plant/soil concentration ratio, transfer coefficient to milk, and transfer coefficient to meat are relatively high among the radioisotope tracers of Table C.1. Transfer factors and other radionuclide-dependent parameter values for ^{90}Sr are presented in Table C.6.

The concentrations in foods were calculated after setting C_w = 1.0 μCi/m^3. The soil buildup time, t_b, was set equal to the time that radionuclides are depositing on vegetation, t_e. The deposition rate, δ, corresponding to a concentration of 1 μCi/m^3 in irrigation water was calculated assuming a water usage of 180 L/m^2 • mo, an irrigation rate associated with crop cultivation in Eastern Washington (Baker, 1983*). It is implicit in the calculations that irrigation water is applied by sprinkler and that the same values of r/Y_v and λ_w apply to the deposition of radionuclides on vegetation whether the radionuclides are in solution or in or on particulates. Since

$$180 \text{ L/m}^2 \cdot \text{mo} = 6 \times 10^{-3} \text{m}^3/\text{m}^2 \cdot \text{d},$$

the deposition rate, δ, associated with a C_w of 1 μCi/m^3 is

$$\delta = 6 \times 10^{-3} \text{m}^3/\text{m}^2 \cdot \text{d} \times 1 \ \mu\text{Ci/m}^3$$
$$= 6 \times 10^{-3} \ \mu\text{Ci/m}^2 \cdot \text{d}$$

---

* D.A. Baker, Personal communication, Battelle Pacific Northwest Laboratory, Richland, Wash. (1983).

Table C.5 Radionuclide-independent parameter values for estimating the dose from agricultural products.[a]

	Parameter	Value
r/Y_v	Normalized interception fraction for leafy vegetables	0.1 m^2/kg wet wt
	Normalized interception fraction for forage	1.8 m^2/kg dry wt
λ_w	Weathering rate for particulates	0.057 d^{-1}
t_e	Growing season, leafy vegetables	75 d
	Growing season, dairy forage	30 d
	Growing season, forage for beef cattle	40 d
P	Density of surface soil	213 kg/m^2
Q_m	Dry forage intake, dairy cows	11.0 kg/d
Q_{w1}	Water intake, dairy cattle	75 L/d[b]
Q_f	Dry forage intake, beef cattle	8.3 kg/d
Q_{w2}	Water intake, beef cattle	45 L/d[b]
U_l	Ingestion rate of leafy vegetables	0.049 kg/d
U_m	Ingestion rate of milk	0.26 L/d
U_f	Ingestion rate of beef	0.26 kg/d

[a] Source: Hoffman et al. (1982).

[b] Source: Comar (1966).

Table C.6 Strontium-90 parameter values for estimating the dose from agricultural products.[a]

	Parameter	Value
λ_r	Radioactive decay rate	6.64×10^{-5} d^{-1}
B_{v1}	Plant/soil concentration ratio, leafy vegetables	8.5×10^{-2} (wet vegetation/dry soil)
B_{v2}	Plant/soil concentration ratio pasture vegetation	1.4 (dry vegetation/dry soil)
λ_s	Migration rate from root zone	6.7×10^{-5} d^{-1}
F_m	Transfer coefficient to milk	1.2×10^{-3} d/L
F_f	Transfer coefficient to beef	5.8×10^{-4} d/kg

[a] Source: Hoffman et al. (1982).

Table C.7 compares the daily intake of ^{90}Sr from drinking well water, eating leafy vegetables irrigated with the well water, and ingesting milk and beef from cattle that drank the well water and consumed forage that was irrigated with the well

water. The intake from drinking water is three to four times greater than that from ingesting leafy vegetables and milk and about 10 times greater than that from eating beef. In the case of the other radionuclides of Table C.1, the intake from drinking water could be expected to exceed that from ingesting leafy vegetables and milk by a comparable factor or more. This is attributable to the greater importance of the deposition of radionuclides on plant surfaces compared with plant uptake from soil and to the relatively high values of the transfer factors for ^{90}Sr.

In the case of HTO, comparing the daily intake of water from ingestion of drinking water and from ingestion of the water content of vegetables, milk, and meat will readily lead to the conclusion that the intake of HTO from drinking water will similarly exceed that via vegetables, milk, and meat.

Table C.7 Intake of Strontium-90 originating in well water drawn from a contaminated aquifer.

Pathway	Intake $\mu Ci/d$ per $\mu Ci/m^3$
Drinking water	2.0×10^{-3}
Leafy vegetables	5.1×10^{-4}
Milk	5.6×10^{-4}
Beef	2.2×10^{-4}

APPENDIX D. ESTIMATION OF THE DOSAGE FROM RADIONUCLIDES IN NATURAL GAS

Dose estimates for hypothetical exposures to natural gas from reservoirs developed with nuclear explosives are used in this report to estimate the dose from the radioactive tracers recovered in natural gas during an EOR operation. Barton et al. (1973) estimated doses to members of the public served by gas distribution systems fed by the Rulison well, which was developed by use of nuclear explosives in 1971. Estimating 1040 Ci of tritium in the gas delivered in the first year and an average tritium concentration of ~10 pCi/cm^3 in the local distribution system, the dose from inhalation and skin absorption of HTO in ground level releases of Rulison combustion products was estimated to be ~0.6 mrem in various communities served by the system. Unvented gas ranges were considered to be the source of the most likely home exposures to combustion products from gas appliances. The dose estimate for exposure to the HTO in unvented combustion products from a gas range was 1.3 mrem for the same conditions above. The collective dose to the population served by the local distribution system, assuming total combustion of the tritium produced (1040 Ci), was estimated to be '4 man-rem.

Jacobs et al., (1972a) assessed the potential dose to workers and members of the public from processing, distribution, and combustion of the gas produced by the Project Gasbuggy nuclear device. It was estimated that the total gas that could be produced in the well would yield 2350 Ci of tritium, 368 Ci of ^{85}Kr, and 10.5 Ci of ^{14}C. Most of these activities (i.e., 2330 Ci of tritium, 360 Ci of ^{85}Kr, and 10 Ci of ^{14}C) were removed from the cavity in a total gas volume of 251 million ft^3 during the major testing and flaring program lasting about 14 months. The total gas produced would be processed at the Blanco processing plant where a small fraction would be used at the plant. The dilution factor for Gasbuggy gas at the Blanco plant was estimated to be 576.

A small fraction of the gas entering the Blanco plant would be used as fuel at the plant and the resultant average concentrations in Blanco gas would be 5.69 x 10^2 pCi/l. for ^{3}H, and 88 pCi/L for ^{85}Kr. Residents of the camp would be subject to an exposure from the combustion products of the fuel used to operate the plant and to a household exposure which we assume results only from unvented gas ranges. Estimates of the dose from unvented gas ranges are based on those of Barton et al. (1973). A summary of the dose estimates for Rulison and Gasbuggy is shown in Table D.1.

The activities in the dry gas produced at the Blanco plant would be 2180 Ci of ^{3}H, and 360 Ci of ^{85}Kr. This gas would enter a gas distribution system and be sent to population centers in California. En route to California, Blanco gas would be further diluted by a factor of 3 to 16. (The total dilution factor for Gasbuggy gas at points of consumption would vary from 1840 to 9220.) Dose estimates per unit concentration of tritium in natural gas are summarized in Table D.2 for the Los Angeles Basin and San Francisco Bay area. The dose estimates are shown for two sources of exposure: HTO in combustion products dispersed in the atmosphere, and HTO in combustion products from unvented kitchen ranges in residences.

Table D.1 Estimates of the dose from radionuclides in natural gas from wells developed by nuclear explosives.[a]

Radionuclide	Source	Total amount Ci	Dilution factor	Concentration pCi/L	Dose estimate
^{3}H	Rulison	1040	∿ 10	∿ 10,000	∿ 0.6 mrem/y via inhalation and skin absorption of the HTO in the ground level release of combustion products 1.3 mrem/y from household exposures to unvented kitchen ranges
^{3}H	Rulison	1040			14 man-rem to population (22,000) served by regional distribution system
^{3}H	Gasbuggy		576	570	0.15 mrem/y to residents of Blanco camp from plant combustion products 0.07 mrem/y from household exposures to unvented kitchen range
^{85}Kr	Gasbuggy		576	88	0.022 mrem/y to residents' skin from combustion products of Blanco plant 0.013 mrem to residents' skin from unvented kitchen range

[a] Sources: Barton et al. (1973), Jacobs et al. (1972a).

Table D.2 Estimates of the dose from the hypothetical use of natural gas containing 1 pCi/cm^3 of tritium in the Los Angeles Basin and San Francisco Bay area.[a]

Source	Annual dose, mrem/y	
	Los Angeles Basin	San Francisco Bay Area
Atmosphere		
Population weighted average	0.024	0.007
Domestic use		
All appliances vented except gas range	0.27[b]	0.27[b]
Total	0.29	0.28

[a] Source: Jacobs et al. (1972a).

[b] If gas heaters and appliances are all unvented, the annual dose would be 2.0 mrem/y in the Los Angeles Basin and 2.5 mrem/y in the San Francisco Bay Area.

NRC FORM 335 (11-81)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1 REPORT NUMBER (Assigned by DDC)
NUREG/CR-3467
UCRL-53485

4 TITLE AND SUBTITLE (Add Volume No., if appropriate)

Environmental Assessment of the Use of Radionuclides as Tracers in the Enhanced Recovery of Oil and Gas

2 (Leave blank)

3 RECIPIENT'S ACCESSION NO

7 AUTHOR(S)

Y. C. Ng, R. T. Cederwall, and L. R. Anspaugh

5 DATE REPORT COMPLETED

MONTH	YEAR
September	1983

9 PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Lawrence Livermore National Laboratory
P.O. Box 5507
Livermore, Ca. 94550

DATE REPORT ISSUED

MONTH	YEAR
November	1983

6 (Leave blank)

8 (Leave blank)

12 SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Division of Risk Analysis
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

10 PROJECT TASK WORK UNIT NO

11 FIN NO

A 0388

13 TYPE OF REPORT

Topical Report

PERIOD COVERED (Inclusive dates)

15 SUPPLEMENTARY NOTES

14 (Leave blank)

16 ABSTRACT (200 words or less)

An environmental assessment of the use of radioisotopes as interwell tracers in field flooding for the enhanced recovery of oil and natural gas was performed. Doses to workers who handle radioactive tracers and to members of the public were estimated for normal and accidental exposure scenarios. On the basis of estimates of the total quantity of tracer radionuclides injected in a year and assumed values of accident frequency, the collective dose equivalent is estimated to be 1 man-rem/y to workers and 15 man-rem/y to members of the public. Accidental exposures are estimated to contribute relatively little to the total.

17 KEY WORDS AND DOCUMENT ANALYSIS

17a DESCRIPTORS

Environmental Assessment,
Radioactive Tracers,
Field Flood,
Recovery of Oil and Gas.

17b IDENTIFIERS/OPEN-ENDED TERMS

18 AVAILABILITY STATEMENT

Unlimited

19 SECURITY CLASS (This report)

Unclassified

20 SECURITY CLASS (This page)

21 NO. OF PAGES

22 PRICE

$

NRC FORM 335 (11-81)