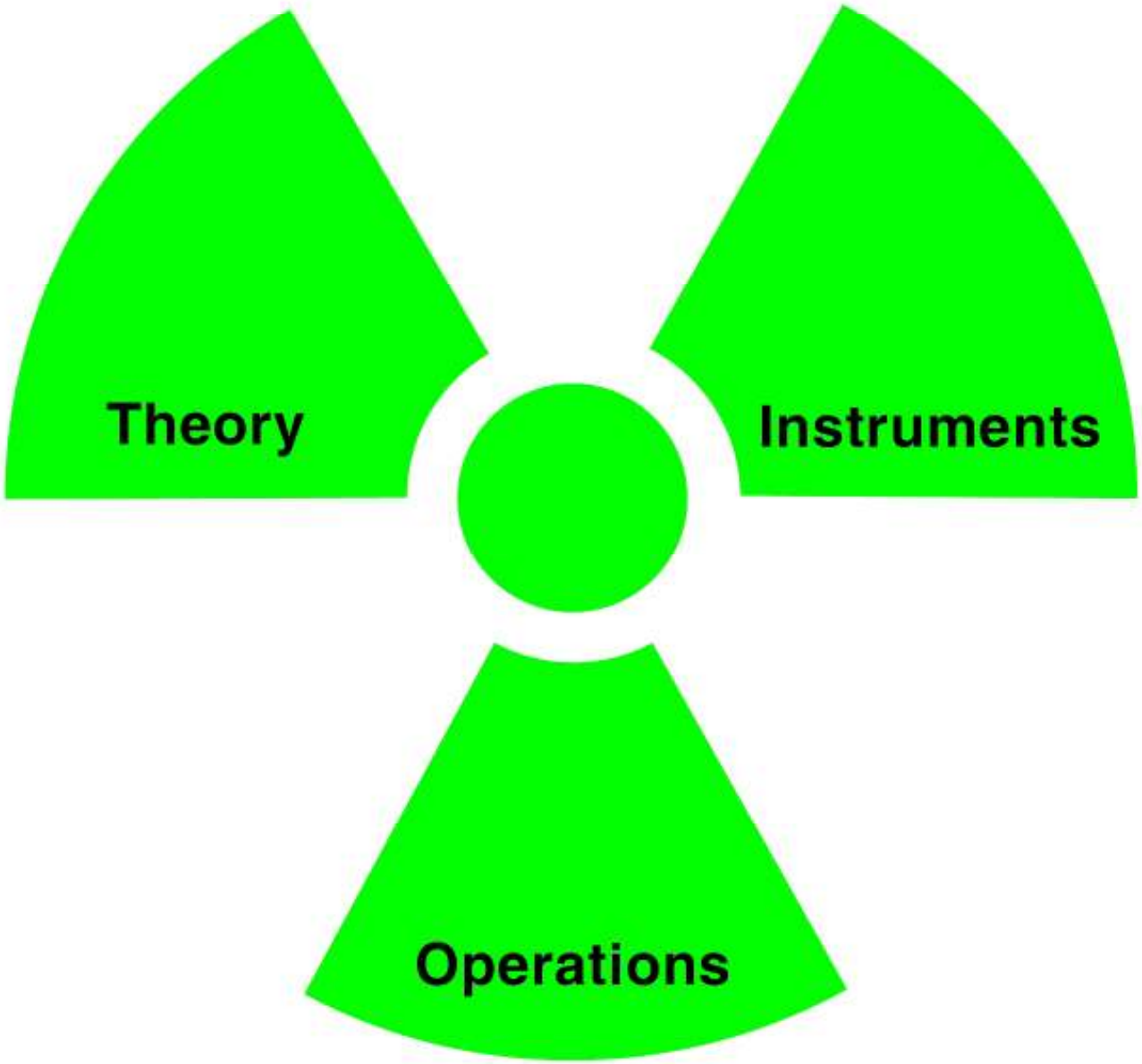
BASIC RADIATION PROTECTION TECHNOLOGY

6th Edition



Daniel A. Gollnick

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Pacific Radiation
2945 Stonehill Drive
Altadena, CA 91001
(626) 798-8100
www.pacificrad.com

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This textbook is dedicated to Don W. Marshall, founding Chairman of the National Registry of Radiation Protection Technologists. Without Don's patience and encouragement in the early phases of this project, this text would have remained just another unfulfilled good idea.

Prologue for Pedestrians (i.e., Beginners)

The ancient question about the appearance first of the chicken or the egg is appropriate in this setting – the beginning exploration of the field of radiation protection technology. The earliest editions of this text assumed that the reader had some basic knowledge of the subject. For example, quantities for measuring radiation were used loosely in the earlier chapters but not formally defined until Chapter 5. Some of the basic regulations about allowed radiation doses to workers were also assumed to be known but only finally treated fully in Chapter 15.

The purpose of this prologue is to give some introductory knowledge about both radiation units and radiation regulations "up front", knowing that these subjects will be dealt with in great detail later in the book. It is also a good place to briefly discuss the overall organization of the book into three Units, three Supplemental Chapters and three Appendices.

The book is divided into three main sections or Units, consistent with the division of the field by the National Registry of Radiation Protection Technologists, the NRRPT®. The first unit is Radiation Protection Theory which encompasses the "fundamentals" and basic principles upon which the field rests. Next, is the unit on Radiation Protection Instrumentation. This section deals with the hardware that we rely on for measurements. Finally, the Radiation Protection Operations unit covers the methods and principles of day-to-day health physics practice (the "software" of radiation protection technology).

After the main material is covered, the book concludes with some supplementary subjects. The topics of "Reactor Health Physics," "Working Safely with Radioisotopes," and "The MARSSIM Decommissioning Process" constitute the supplemental chapters. For technicians interested in these specialized areas, additional subject material is presented which goes beyond the introduction to these topics in the main body of the book. The supplemental material concludes with some appendices of information that may be of assistance.

Before introducing the quantities and units used in the field, it is important to distinguish between "radiation" and "radioactive material". Usually, the term radiation refers to energy flying through space, e.g., an x-ray or gamma ray. Radioactive material, on the other hand, means the presence of actual unstable atoms which are decaying through nuclear emissions. It then makes sense that the units generally used to measure radiation actually describe the effects on some "absorber" exposed to the radiation while the units used to measure radioactive material describe the number of disintegrations per time from the radioactive atoms present.

Since this book is distributed world-wide, some comments on the families of units in use around the globe are in order. Almost all developed countries have fully adopted the International System of Units, commonly called the metric system. An organization called the International Commission on Radiation Units and Measurements, the ICRU, is responsible for defining the metric system units and quantities for radiation protection purposes. Due to the extreme reticence of the United States of America to convert to metric measurement, the situation now exists in the U.S. in which two separate systems coexist in health physics regulations. The 1991 major

overhaul of U.S. regulations (Title 10 Part 20 of the Code of Federal Regulations for non Department of Energy radiation workers and Title 10 Part 835 for DOE workers) recognized both the new metric units and the old English or "conventional" units in use in radiation protection since the 1940s. The old units have been formally phased out in the rest of the world.

So, for purposes of understanding the early chapters of the book, the following radiation units will be loosely defined. A roentgen is a measure of the amount of x-rays or gamma rays reaching some measurement point. It does not have a metric equivalent unit as the ICRU has abolished it in favor of the rad and its metric equivalent the gray (Gy). The rad and gray measure the energy deposited in something being irradiated by any type of radiation. They are more universally usable than the roentgen which is limited only to x- and gamma rays. The last basic radiation unit needed at this time is the rem and the metric analogue the sievert (Sv). These units provide a measure that can be related to the effect of radiation exposure specifically on humans. They are similar to the previous units mentioned except that weighting factors are built-in to take into account the fact that different types of radiation produce different effects in humans for the energy that is deposited. For practical purposes:

1 gray = 1 sievert = 100 roentgen = 100 rad = 100 rem.

To express the quantity of radioactive material in a radioactive source, the metric system unit most commonly used is the becquerel (Bq). The corresponding English unit is the curie (Ci). A one-becquerel source undergoes one decay per second. On the other hand, a one-curie source has a decay rate of 3.7×10^{10} per second.

Turning now to the body of laws designed to make the radiation workplace safe and to protect the population from the harmful effects of radiation, again there are different families of regulations. Most of the world accepts the regulations developed by the International Commission on Radiological Protection, the ICRP. In their 2007 recommendations they established an annual dose limit of 20 mSv (2 rem) for radiation workers, averaged over 5 year periods, with no single year exceeding 50 mSv (5 rem). Annual limits in all national codes now include both external radiation exposure and the effects of internally deposited radioactivity in the human body when calculating radiation doses. The ICRP also limits individual members of the public to 1 mSv (10 millirem) of radiation dose in a single year.

In the United States, the basic ICRP regulations have undergone further refinement. The particular regulations that apply to a given U.S. radiation worker depend on that worker's employer. About 86% of the radiation work force falls under the jurisdiction of the Nuclear Regulatory Commission, NRC, either being directly licensed by NRC or working in an Agreement State which is bound by NRC regulations. 6% of U.S. radiation workers are covered by Department of Energy (DOE) regulations and the remaining 8% are under regulations issued by the Department of Defense, DOD.

The NRC rules are published in the Code of Federal Regulations, CFR, in Title 10, Part 20. They limit a radiation worker to 50 mSv (5 rem) of dose annually. This is significantly higher than allowed by ICRP recommendations. Members of the public are limited to 100 mrem (1 mSv) annually, from any single licensee. Radiation rules for the DOE work force have been codified and published in the Federal Register as Part 835 to Title 10 CFR. Chapter 15 of this text spells out the details of ICRP, NRC and DOE regulations. It also shows a comparison of the major differences.

Consistent with the previous editions of this book, two different typefaces are used to distinguish two different types of information. Material which falls in the mainstream of radiation protection and is basic information that any well rounded health physics technician should know is printed in this regular type.

Material which is considered supplemental is printed in this reduced size, bold type. This material may contain historical information, topics outside the mainstream or information at greater depth than needed to adequately perform as a radiation protection technologist. Students concentrating on the highlights of a chapter should skip this material.

Good luck in your exploration of the exciting field of radiation protection technology!

Preface to the Sixth Edition

Just as the manuscript for this Sixth Edition was about to be sent to the printer, a 9.0 earthquake struck Japan, and "Fukushima Daiichi" became a household word. Nuclear critics cried out with renewed vigor, filling the daily news with speculation while, at the same time, construction continued in Georgia on the first new nuclear plants to be built in decades. These are indeed exciting times!

Whether the resurgence in new U.S. power reactor construction continues, only time and circumstances will tell. In any case, I am seeing a definite increased interest in nuclear technology training. There has been almost an epidemic of new two-year college programs springing up, driven in part by the development of a uniform national curriculum under support from the National Energy Institute. This will hopefully provide the graduates to fill the huge numbers of positions projected to open in the near future as retirees leave the 104 currently operating plants in this country.

The Second Edition of **Basic Radiation Protection Technology** was released a short time after the Chernobyl accident. The focus of the third edition was the rewritten10 CFR 20/10 CFR 35 regulations and the fourth edition featured the MARSSIM (Multi-Agency Radiation Survey and Site Investigation Manual). The fifth edition followed the September 11th attacks on the U.S. Suddenly, "Nuclear Terrorism" became a real possibility. The "theme" of this Sixth Edition is nuclear technology training. To that end, I have added some new topics or expanded the existing ones to hopefully make the text more useful in the college arena. As always, your suggestions would be appreciated as to how successful I was in this goal.

I have continued the practice of placing supplemental material in a **smaller, bold typeface to distinguish it from the** more mainstream topics. On your first reading through a chapter, you might find it helpful to skip over the supplemental material. Shaded boxes set the Sample Problems apart from figures and illustrations. Each sample problem illustrates an application of the corresponding concept being discussed. A complete solution is provided as part of each Sample Problem. This is intended to be of use to student readers who have not yet acquired years of experience in the field.

The Chapter Outlines and Chapter Summaries, which first appeared in the third edition, are continued. I recommend that you read the summary before starting the Chapter. This will give you an idea of what is to be covered and how the various topics fit together with each other.

Each new edition is an opportunity to include additional topics not previously incorporated. This is both a blessing and a curse! It allows me to keep up with changes in the field, but it places me in the position of judge and jury as to what should be added or ignored. It also means that the page count keeps climbing. The new topics added, or significantly expanded, with this Sixth Edition include:

- AC electrical circuit theory
- Trigonometry
- Electron conversion and internal conversion decay processes
- Biodosimetry methods and techniques

- BEIR VII cancer risk estimates
- ICRP Publication 103
- Advanced power reactors
- Radiochromic dye self-indicating dosimeters
- Air sampling for radioiodine and tritium
- Homeland security instrument testing programs
- Yucca Mountain depository update
- West Texas new low level waste site
- Fukushima Daiichi reactors accident

The new topics just mentioned have been placed within the basic framework of the 15 main chapters. Three Supplemental Chapters bring the Chapter count to 18. The three Appendices from the Fourth Edition have been carried over as well. Each Chapter now concludes with "Other Resources", many of which can only be accessed online.

The complete worked out solutions to all of the text problem sets are available as an integral part of the Student Manual of the Pacific Radiation Self Study Course. The course also includes this textbook, a set of audio CD Chapter summaries, a reference library for health physics technology and a Final Exam/Certificate package. The Student Manual and all other Course components are available for purchase as separate items. (In December of 1999, this text and the accompanying Self Study Course were approved by the National Registry of Radiation Protection Technologists, NRRPT®, for 6 "Registration Maintenance Points" for RRPTs to maintain active registry status.)

Once again, thanks are in order for persons assisting with this work. Louie Lopez and David Wolf of Copy-Rite Press have coordinated the printing and binding of this edition, making that part of the project run most smoothly. My wife, Laurie, proofread every Chapter three times, a gigantic effort certainly deserving of a medal!

Eric Goldin, Ph.D., CHP, is a Consulting Nuclear Engineer for Southern California Edison's San Onofre Nuclear Generating Station. Eric produced the topic on Advanced Power Reactors in Chapter 6. He also helped update the data and figures on nuclear power in that Chapter.

Ken Smith, CHP is a Certified Health Physicist and founder of Class7Training LLC, an environmental health and safety and consulting company specializing in shipping radioactive materials by air. His co-author, Bridget Smith is a technical writer who holds an M.A. in education. Class7Training can be found on the web at http://www.class7training.com. Ken and Bridget totally revamped the Transportation topic in Chapter 11.

I sincerely hope that this book will be of use to all radiation protection technologists in their career development. Please keep suggestions for improvements coming in. Maybe there will someday be a Seventh Edition!

March 2011, Altadena, CA — Daniel A. Gollnick

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Supplements and Appendices

Unit 1

Radiation Protection THEORY


A HOME FOR YOUR HOUND

1

Basic Mathematics and Science Review

Outline of this Chapter

Chapter Summary

This introductory chapter is the first of six chapters which constitute the Radiation Protection Theory Unit. The theory forms a necessary foundation of knowledge upon which the more practical aspects of radiation protection technology are based. Later units will cover Radiation Protection Instrumentation and Radiation Protection Operations.

This first chapter reviews some basic skills and knowledge needed for the successful practice of radiation protection. Scientific notation is useful when calculations involve very large and small numbers. Calculators and computers ignore the errors associated with measured numbers so the technician needs skill in resolving the correct number of significant figures in computations.

Health physics practice often involves equations with power or exponential functions, so skill in handling powers and roots, exponentials, logarithms and trigonometric functions must be developed.

Atoms are composed of an outer layer of orbital electrons surrounding an inner, tiny nucleus composed of protons and neutrons. Although its diameter is only 1/10,000 the atomic diameter, over 99% of the atomic mass is contained in the nucleus. Protons and neutrons are, themselves, composed of smaller particles called quarks.

Newton's Laws of Motion are fundamental to understanding our physical universe. They are the basis for explaining the concepts of and interrelationships between force, work and energy.

Moving electrical charges create an electric current. The flow of electrons in DC electrical circuits is described by Ohm's Law, V = IR. AC circuits involve bi-directional flow of the electrons. This leads to a phase angle difference between the current and voltage if capacitors and/or inductors are included. In electronic devices, amplification is possible. Electronic vacuum tubes of the past have given way to semiconductor transistors which are more rugged and need less power to operate. They, in turn, have been replaced in state-of-the-art health physics instrumentation by integrated circuits and microprocessors.

Radiation protection technologists need a brief working knowledge of major human organ systems These are described along with a short discussion of some topics from the field of chemistry – atomic weight, electron shell theory, the periodic table and oxidation/reduction reactions.

Mathematics Review

Scientific Notation

Radiation protection technologists must often perform calculations with very large numbers and very small ones. For example, the number of molecules in one mole of a substance, Avogadro's Number, is huge compared to ordinary numbers while the diameter of the nucleus of an atom, in meters, is minuscule compared to everyday numbers. In performing arithmetic operations such as multiplying and dividing, it is easy to gain or lose a "few zeros." Even when you are using a pocket calculator, it is difficult to enter very large and very small numbers with exactly the right number of "button pushes." This is one reason technicians often make use of scientific notation, a convenient method for handling calculations with large and small numbers.

In Sample Problem 1, we see Avogadro's number and the nuclear diameter

Sample Problem 1

GIVEN:

In ordinary notation, 600,000,000,000,000,000,000,000 is the value of Avogadro's Number and the diameter of an atomic nucleus, in meters, is 0.00000000000001.

FIND:

An expression for these quantities in scientific notation.

SOLUTION:

$N_A = 6 \times 10^{23}$ atoms/mole = 6E23 atoms/mole.

The nuclear diameter is $d = 1 \times 10^{-14}$ meters = 1E-14 meters.

written in scientific notation. Normally, to write a number in scientific notation the first non-zero digit is written followed by a decimal point. Remaining significant digits are added after the decimal point. Finally, the number is completed by multiplying the written result by 10 raised to some power. Numbers larger than 1 are written with a positive power (exponent) and numbers smaller than 1 are written with a negative power. Also illustrated in the above Sample Problem is the "shorthand" way of expressing 10 raised to a power. The symbol "E" means "10 raised to the exponent." This form is occasionally used by some pocket calculators for scientific notation and by older computer printers which had difficulty printing superscripts and subscripts.

Significant Figures

Many of the numbers that are used in day to day operations by a technologist are obtained as an instrument reading from a scale or meter face. It must be kept in mind that a physical measuring instrument always has some inherent limitations in accuracy. Thus, a report of a measured value should not contain more "significant

figures" than can be justified by the instrument. The usual rule is that the instrument is read to the nearest mark that represents the smallest scale division. We then make a best guess (interpolation) of the fraction of the next scale division the instrument is reading. This gives us one more (and the final) significant figure in the measurement. An example is shown in Figure 1. To the correct number of significant figures, the

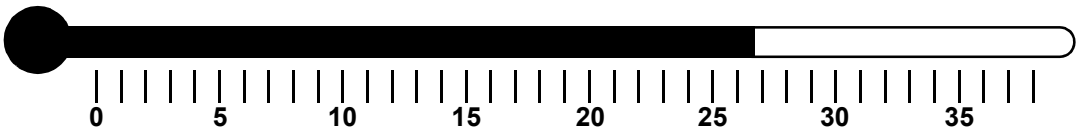


Fig. 1 - A thermometer

thermometer reads 26.7 °C. The 0.7° results from the interpolation between 26° and 27°. The problem with significant figures becomes very noticeable when using a scientific calculator. The calculator treats each number entered as having a long string of significant zeros at the end. For example, suppose the density of a piece of rock is being determined. The mass, obtained from weighing, is found to be 3.4 grams. The volume is measured to be 2.6 cubic centimeters. If the density (in grams per cubic centimeter) is now computed using a calculator, the result of 3.4 divided by 2.6 is shown to be 1.3076923 on a typical calculator that displays 8 figures. In dividing numbers accurate to 2 significant figures the calculator displays the answer with 8 significant figures.

Recognizing the limitations of the measuring instruments, it is necessary for the technologist to throw away many of the digits. Since this problem comes up so often, rules have been developed to handle these situations. Previous editions of this book have listed some of the common rules in use. However, a 1995 article in the journal **Health Physics** points out that these simple rules are not strictly correct mathematically [*Chamberless and Broadway, Significant Digits: Foundations, Myths and Utilization, Health Physics, 69:257-260; Aug.,1995*]. The bottom line? The technologist is cautioned to make use of common sense when performing calculations involving measured values.

Zeros to the right of non-zero digits present a problem when a number is written in ordinary notation. How many significant figures are there in the reported weight of 12,000 pounds for a loaded rad waste truck? Possible answers include 2, 3, 4, and 5 significant figures. If weighed on a scale with 10,000 pound smallest scale divisions, the answer has 2 significant figures. If weighed on a scale with smallest scale divisions of 10 pounds, the answer has 5 significant figures. Writing the weight in scientific notation, however, solves the ambiguity. In scientific notation, the two weights would have been reported as 1.2×10^4 pounds and 1.2000×10^4 pounds, respectively. Sample Problem 2 shows an example calculation. The number of significant figures shown is consistent with the precision of the given values.

Consistency of Units

Just as chemical equations must be balanced (the same number of oxygen atoms must be on both sides, etc.), so must mathematical equations be balanced in

terms of the dimensions or units. One cannot merely add mR and R numerical exposures without first converting them to the same unit. Under the press of time, such mistakes can and do occur.

One good technique which reduces the chances of making unit errors is to ALWAYS CARRY ALONG THE UNITS with every number written down. When the calculation is complete, units should cancel in the numerator and denominator to leave the final units desired for the result. This technique has the added advantage that, by keeping track of the units as they cancel, it will be obvious whether "conversion factors" go into the numerator or into the denominator of the calculation. (See Sample Problem 2).

Sample Problem 2

GIVEN:
Radiation doses of 2.123 sieverts and 21 millisieverts.
FIND:
The algebraic sum of these two doses.
SOLUTION:
1 Sv = 1,000 mSv.
So, 2.123 Sv x 1,000 mSv/Sv + 21 mSv = 2144 mSv.

{NOTE TO THE READER: Text which appears in this reduced, boldface type is considered "Supplementary Material," not essential to the mastery of the field of radiation protection technology. Such material may expand on a concept just presented, introduce new topics of related interest to radiation protection or provide historical background information.}

Graphing Practices

Often it is necessary to prepare graphs of physically measured data. A well laid out graph can greatly enhance the interpretation of the results. A number of suggestions follow which will improve the readability of a graph.

The graph should fill most of the page on the graph paper. Proper choice of values on the vertical and horizontal scales will assure this. For the sake of someone else reading data from your graph, try to choose the value of the small scale divisions to equal a 1, 2 or 5. Each axis should carry a label indicating the physical quantity plotted along it and the units that correspond to the scale (e.g., Dose Rate in mrad/hr or Time in hours). Individual measured values should be indicated by plotting a symbol at the correct point. Then a SMOOTH CURVE should be drawn through the data points rather than a series of lines connecting the points. In health physics, most of the quantities measured are smoothly varying. Finally, if error bars are used to indicate the uncertainty associated with the individually measured data points, they should be drawn at plus and

minus one standard deviation (a concept to be covered in Chapter 12) and connected by a solid line. A representative graph is shown in Figure 2 to illustrate the suggestions given.

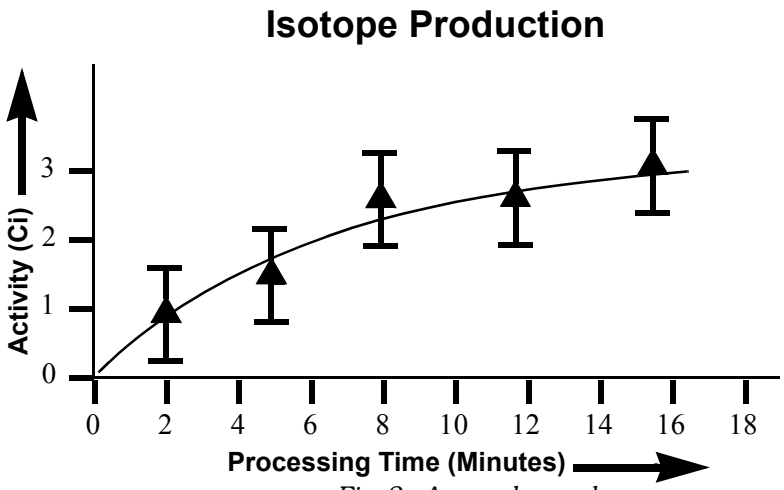


Fig. 2 - A sample graph

Power Functions

Occasionally the behavior of some quantity of interest in radiation protection is expressed mathematically as a power function. This means that one variable depends on another variable raised to some power. A practical example might be the excretion of certain internally deposited radioactive materials by way of urine as a function of

Sample Problem 3

GIVEN:
The following two problems involving power functions.
FIND:
The unknown quantity, x.
SOLUTION:

1. $x^3 = 2.67$
Find x
$x = \sqrt[3]{2.67}$
Enter 2.67 on calculator.
Press "xth root of y" button.
Enter 3. Press "=".
x = 1.39

2. $x^{-2.71} = 1.5 \times 10^{-3}$
Find x
$x = \sqrt[-2.71]{1.5E\text{-}3}$
Enter 1.5×10^{-3} on calculator.
Press "xth root of y" button.
Enter –2.71. Press "=".
x = 11. [2 sig. figures allowed]

time. To solve equations containing power functions, use is made of the key on a scientific calculator which takes an arbitrary root of a number. The key is often labeled as the Xth root of Y, $\sqrt[x]{y}$. Some examples of the solution of power function equations are given in Sample Problem 3.

Exponential Functions

Many physical phenomena that we measure in radiation protection technology grow or decay according to an exponential law. This means that their behavior depends on e, the base of natural logarithms (e = 2.718281....), raised to some positive or negative power. Common examples would be the physical radioactive decay of a nuclide or penetration of a thin shield by a beam of gamma radiation.

Mathematically, it is frequently found that the quantity that we are "solving for" is in the exponent of the exponential function. In the study of algebra, it is discovered that the operation of taking the natural log of e raised to any exponent gives the exponent back. (This should not be too surprising since it follows from the definition of logarithms. Recall that the log of 100 [10 to the 2 power] is 2 and the log of 1000 [1 X 10^3] is 3. Thus, since e is the base of natural logs, the natural log of e to the x power is x.) The shorthand way of writing "natural log" is ln. Some exponential function calculations are shown in Sample Problem 4.

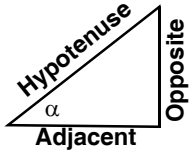
Sample Problem 4

GIVEN:
The following two problems involving exponential functions.
FIND:
The unknown quantity, x in example 1 and t in example 2.
SOLUTION:

1. $e^x = 13$	**2. $e^{-at} = 1500$, a = –0.0200.**
Find x.	**Find t.**
Take ln of both sides.	**Take ln of both sides.**
ln e^x = ln 13 = x.	**ln e^{-at} = ln 1500 = –at.**
So, x = 2.6.	**–at = 7.313.**
	So, t = 7.313 / 0.0200 = 366.

Trigonometric Functions

Trigonometry is the study of triangles. Some physical variables change in a manner best described by trigonometric functions. A trig function is merely the ratio of the lengths of certain sides of a right triangle. A familiar example would be the power line voltage measured at a wall receptacle as a function of time. It describes a sine wave, i.e., we say it varies sinusoidally with time. Figure 3 shows the names of the three sides relative to the angle α and lists the six possible trigonometric functions



Sine α = sin α = $\frac{\text{Opposite}}{\text{Hypotenuse}}$

Cosine α = cos α = $\frac{\text{Adjacent}}{\text{Hypotenuse}}$

Tangent α = tan α = $\frac{\text{Opposite}}{\text{Adjacent}}$

Cotangent α = cot α = $\frac{\text{Adjacent}}{\text{Opposite}}$

Secant α = sec α = $\frac{\text{Hypotenuse}}{\text{Adjacent}}$

Cosecant α = csc α = $\frac{\text{Hypotenuse}}{\text{Opposite}}$

Fig. 3 - Definitions of the six trigonometric functions

(ratios) that are possible.

A modern scientific calculator can easily perform computations involving trig functions. However, it is necessary that you tell the calculator the system used for entering angles. Many such calculators will accept angles expressed in degrees, radians or grads. A check of the instruction manual will show how to set the calculator to the different angle notations. (Since there are 2 π radians in a circle of 360 degrees, 1 radian = 57.3 degrees. Also, a right angle, 90 degrees, = 100 grads.) A calculation involving trigonometric functions is given in Sample Problem 5.

Sample Problem 5

GIVEN:
The following problem involving a trigonometric calculation.
FIND:
The unknown variable, Y.
SOLUTION:
10 Y = 6.82 tan 14° / sin 81°
From a calculator, tan 14° = 0.249; sin 81° = 0.988.
Therefore 10Y = 6.82 x 0.249 ÷ 0.988 = 1.719.
So, Y = 0.172

Basic Physics Review

Structure of Matter

In this section, a number of different topics from physics will be reviewed. First consider the structure of matter. If a substance such as ice or mica is divided in half, and then divided again and again, eventually a point will be reached at which a further sub-division will produce pieces which differ physically and chemically from the original substance. The smallest unit of a chemical compound which still retains the physical and chemical properties of the original substance is termed a molecule.

It is possible to separate the component parts of molecules. When this is done, the sub-molecular units are called atoms. An atom can also be defined as the smallest unit of a chemical element, such as gold or carbon, which has all of the physical and chemical properties of that element. Atoms are relatively small, but a variety of techniques are available to "smash" atoms into even smaller units.

When atoms are divided, the particles commonly found are electrons, protons and neutrons. The way in which these sub-atomic particles combine to form atoms will be discussed in Chapter 2. Electrons and neutrons released from atoms may be part of a radiation field which must be measured by a radiation protection technologist.

It is reasonable to ask what happens when electrons, neutrons and protons are sub-divided. According to the theory of elementary particle physics first proposed by Murray Gell-Mann in the 1950s, sub-atomic particles are themselves composed of smaller units called quarks and leptons. Gell-Mann suggested the name "quark" after a character in the book Finnegan's Wake. Lepton is from the Greek word meaning small or tiny.

There appear to be 6 different "flavors" of quarks, each in 3 different "colors" and 18 corresponding antiquarks (counting flavors and colors). Antimatter will be discussed in Chapter 2. Particles like protons and neutrons are combinations of three quarks. The quarks are held together by the action of elementary particles called gluons. Protons consist of two "up" quarks and one "down" quark. Neutrons consist of two "down" quarks and one "up."

One of the most intriguing properties discovered to date for quarks is the fact that they carry only a fraction of an electronic charge. For decades, physicists were convinced that the basic charge on an electron, e = 1.6 x 10^{-19} Coulombs, was indivisible. Quarks, however, carry charges of 1/3 and 2/3 of e.

Discovery of the top quark took place at Fermilab (near Chicago, IL) in March of 1995. Additional experimental work is underway to more firmly establish its properties. Some of the masses listed in Figure 3 carry large uncertainties as of 2010.

Electrons turn out to be one member of the second family of elementary particles, leptons. In contrast to quarks, leptons carry integer (whole number) amounts of the basic electronic charge, e. The other members of the lepton family include the muon (first mistakenly thought to be a meson), the tau particle discovered in 1976 and three kinds of neutrinos. A complete table of the "Standard Model of Matter" as of 2010 is shown in Figure 4.

Additional research is being vigorously carried on at high energy nuclear particle accelerator laboratories to further study the fundamental structure of matter. The properties of quarks, gluons and leptons undoubtedly still hold a few surprises for physicists.

If you <u>really</u> want to stretch your mind, you might consider what quarks and leptons are made of! Possibly, the answer lies in an exotic subfield of theoretical physics called Superstring Theory. This theory suggests that all quarks and leptons are made up of vibrating bits of energy, in the shape of strings, about 10^{-35} meters in length. Different modes of vibration are what determine which particle a given vibrating string corresponds to. The strings themselves may be composed of tiny charged black holes!

Finally, don't be too hasty to pat ourselves on the back for having

Flavor	Symbol	Charge	Mass-Energy as of 2010
		QUARKS	
Up	u	+2/3 e	0.003 GeV
Down	d	–1/3 e	0.006 GeV
Strange	s	–1/3 e	0.1 GeV
Charm	c	+2/3 e	1.3 GeV
Bottom	b	–1/3 e	4.2 GeV
Top	t	+2/3 e	171 GeV
		LEPTONS	
Electron	e	–1 e	0.000511 GeV
Muon	μ	–1e	0.1057 GeV
Tau	τ	–1 e	1.777 GeV
Electron Neutrino	ν_e	0	< 0.000002 MeV
Muon Neutrino	ν_μ	0	<0.2 MeV
Tau Neutrino	ν_τ	0	<18 MeV

Fig. 4 - The Standard Model of Matter

reached a rigorous understanding of the way our universe works. Astronomers have established that all of the visible matter in the universe only accounts for 20% of the known amount of matter - the remaining 80% is referred to as Dark Matter. Its composition is still unknown.

Newtonian Mechanics

In the closing years of the 17th century, Isaac Newton formulated what have become known as Newton's Laws of Motion. The three laws describe the behavior, in time and space, of objects set in motion. They can be used to describe the orbits of planets or satellites or applied to the trajectories of sub-atomic particles. Some of the useful concepts and formulas based on Newton's Laws follow.

A vector, in mathematics, is a quantity that has both a size (magnitude) and a direction. Velocity is a vector quantity. The magnitude of velocity is called the speed. Thus, a velocity might be written as 55 miles/hr North. In physics terms, the velocity is the time rate of change of position in some direction. The velocity vector just referred to above means that our position is changing toward the North at a rate of 55 miles each hour.

Another related vector quantity is acceleration. In physics terms, acceleration is the time rate of change of velocity in some direction. For example, the acceleration due to gravity is represented by a vector of magnitude 9.8 meters/sec per sec (written as m/sec^2) pointing down toward the earth's center. This means that if we drop an object such as a geiger counter off the top of a tall building, after 1 second, it will be traveling with a velocity of 9.8 m/sec toward the street. After 2 seconds, its velocity

VELOCITY = change of position/time, in a direction
ACCELERATION = change of velocity/time, in a direction
WORK = Force x Distance
POWER = Force x Distance/Time = Work/Time
POTENTIAL ENERGY = Mass x Height x Acceleration of Gravity
KINETIC ENERGY = 1/2 x Mass x Velocity^2

Fig. 5 - Some definitions from mechanics

will be 19.6 m/sec toward the street, etc. During any equal time intervals, the velocity would increase by the same fixed amount.

A force can be thought of as a push or a pull exerted on an object. If this push or pull causes the object to be displaced (i.e., move to another location) then it is said that work, in the physics sense, has been performed. The amount of work is calculated by multiplying the force times the distance moved. If a force of 15 pounds-force moves a lead brick 2 feet across a table top, then the work done was 2 feet times 15 pounds-force = 30 foot-pounds. Notice that for work to be done in the true physics sense, it is necessary that the object move. If a technician pushes very hard on a 6 foot thick concrete shield wall and burns up lots of calories in the process, she or he may lose some weight but no "work" has been done.

The time factor has not been mentioned yet in connection with work. The same amount of total work is done if the lead brick is moved the 2 feet in one second or 1 minute. If we take into consideration how fast the work is done, then the power is computed. Power is just the time rate of doing work. To calculate the power, the amount of work is divided by the time taken. In the present example, if the lead brick is moved 2 feet in 1 second by 15 pounds-force, the power expended is 30 ft-lbs/sec. This is about 1/20 of a horsepower. One hp = 550 ft-lbs/sec. That is, a "typical" horse (reference horse?) can lift a 550 lb weight 1 foot per second. This is way better than most health physics technicians can do!

In physics, energy is considered to be the ability to do work. This ability can be the result of an object being in some position (potential energy, e.g., water behind a tall dam) or it can result from an object being in motion (kinetic energy, e.g., the falling geiger counter). If the geiger counter were sitting on the edge of a table, it has potential energy relative to the floor. The amount of its potential energy is calculated from the product of its mass times its height times the magnitude of the gravitational acceleration, 9.8 m/sec^2. Note that the potential energy can be converted into kinetic energy of motion by nudging it off the edge so that it falls to the floor. On striking the floor, it has a kinetic energy exactly equal to the potential energy it had earlier sitting on the table edge. It will do work on the floor deforming and slightly heating the carpet. The equality of the kinetic and potential energies is called the Conservation of Energy Law. In the mks system (meter, kilogram, second) energy is commonly measured in joules (1J = 6.242 X 10^{18} eV = 0.7376 ft-lbs). The amount of kinetic energy associated with a moving object is calculated from the product of one-half times the mass times the square of the velocity. These equations are summarized in Figure 5. Sample Problem 6 gives an example.

As we will see in an upcoming Chapter, radiation carries energy. If this energy is deposited in some absorber, e.g., a lead brick or a radiation protection technologist,

Sample Problem 6

GIVEN:
A one horsepower gas engine is used to lift a 1,000 pound shield block.
FIND:
How fast can the block be raised? How long would it take to lift it 20 feet? What is the engine power rating in joules/second?
SOLUTION:
Power = Work/Time
1 hp = 550 foot-pounds/second. So, the lifting rate will be
550 ft-lbs per sec / 1000 lbs = 0.55 ft per second.
To lift the block 20 feet would take
t = Distance/Rate = 20 feet / 0.55 ft per sec. = 36.4 seconds.
The engine is rated 550 ft-lbs per sec /0.7376 ft-lbs per joule = 746 J/sec.

the temperature of the absorber is raised. To express the temperature of an object, some agreed upon scale of units must exist. In practice, technicians may encounter four of the several scales developed over time. These are the Celsius, Fahrenheit, Kelvin and Rankine scales. The scales are defined so that 1°C = 1°K, 1°F = 1°R and 1 °F = 5/9 °C. The Rankine scale, established in 1859, is sometimes used by engineers in North America. Both the Rankine and Kelvin scales start at absolute zero. In the Celsius scale, zero is the freezing point of water. A comparison of the four scales is made in the chart of Figure 6.

To convert between the Celsius and Fahrenheit scales, the following relationship is used: $T(°F) = 32° + \frac{9}{5}\ T(°C)$. The conversion between the Celsius and Rankine scales makes use of the relationship: $T(°R) = 491.67° + \frac{9}{5}\ T(°C)$.

Scale > >	° Kelvin	° Celsius	° Fahrenheit	° Rankine
Absolute Zero	0	-273.15	-459.67	0
Water Freezes	273.15	0	32	491.67
Water Boils	373.13	100	212	671.64

Fig. 6 - A comparison of some common temperature scales

Electricity

This topic deals with moving electrical charges. An object is said to carry an electrical charge when it possesses electric potential energy of position. This means that the object would feel a force in an electric field and would move either toward or away from the source of the field if it is free to do so. As an example, the collecting

electrode in an ion chamber radiation detector carries a positive charge and will exert an attractive force on an electron inside the chamber. Electric charge is either positive or negative. Many years ago it was decided that electrons would be assigned a negative charge. (The decision was based on an incorrect assumption by Benjamin Franklin that charge carriers in a wire are positively charged.) The rule for the direction of the force between charges is that like charges repel and unlike charges attract.

An important law of considerable use in radiation protection technology gives the strength of the force between two charged objects such as two electrons. This law is called Coulomb's Law after the French physicist who discovered it around 1780. The usual way of writing Coulomb's Law is shown in Figure 7. Notice that the force

$$\text{Force} = k\,(Q_1 Q_2)/r^2$$

where k = a constant of proportionality, dependent on units used
Q_1 = charge on one object
Q_2 = charge on second object
r = distance of separation of objects

Fig. 7 - Coulomb's Law

between charges is an inverse square law. The size of the force is inversely proportional to the square of the distance of separation. If charges are very close (small r) then the force is large. (An interesting question to be dealt with in Chapter 2 – how can all those protons exist side-by-side inside a nucleus? Coulomb's Law shows they would repel each other with tremendous forces.) Sample Problem 7 illustrates this important principle.

An ion is an atom which has had its neutrality upset by the removal or addition of electric charge, usually in the form of an electron. There are several ways to produce ions. One method of great interest in radiation protection is through irradiation. By bombarding something with radiation it is possible to remove electrons from the normally neutral atoms. The resulting positively charged atom and negative electron

Sample Problem 7

GIVEN:
Two parallel plates in a laboratory ion chamber are spaced 1.4 mm apart. They are charged up to 13 coulombs each, one negatively and the other positively.
FIND:
What change in the force between the plates would occur if the charge were raised to 27 C on each and the plates moved to 1.8 mm separation?
SOLUTION:
From Coulombs Law, $F = k\,Q_1\,Q_2 / r^2$. So the original force would equal

$$F = k \times 13 \times 13 / 1.4^2 = 86\ k \text{ of attraction.}$$

Now, the revised force will be

$$F = k \times 27 \times 27 / 1.8^2 = 225\ k.$$

So, the new force will be about 2.6 times stronger, and still attractive.

are termed an ion pair.

The flow of electric charge along a conductor such as a wire is called an electric current. The magnitude of the current is measured in amperes (abbreviated A – note that all units in physics that are named after a person have a capitalized abbreviation while other units such as the meter, are lower case, i.e., m). The current is analogous to the volume flow in a water pipe. One ampere of current is caused by exactly one coulomb of charge passing a point in the wire in one second. Since the charge is carried by electrons, each having a charge of 1.6×10^{-19} coulombs, a one ampere current is produced by 6.3×10^{18} electrons per second. A common "D" cell flashlight battery is capable of putting out approximately one ampere under full load.

The potential difference, often incorrectly called the "voltage," is the difference in electrical potential energy between two points in an electrical circuit. It is analogous to the pressure in a water pipe. The unit of potential difference is the volt, V (after the Italian, Professor Volta). It represents the work that would be done to move a unit charge between the measurement points.

In a water pipe, the friction in the walls and the buildup of corrosion tend to restrict the water flow. In an electrical conductor, the retarding of the current flow is termed the resistance. Resistance is measured in ohms. Resistors found in typical electronic circuits have resistances from a few 10s of ohms to about a million ohms. Generally, a material such as copper which has a low resistance is called a conductor. Materials such as glass and plastics have high resistance and are called insulators. Certain materials have resistances intermediate to conductors and insulators. They are called semiconductors. Silicon and germanium are examples. They have increasing applications in modern electronics.

For conductors there is a definite interrelationship between current, resistance and potential difference. This dependence is expressed by Ohm's Law. Figure 8 shows the three ways of writing it. The potential difference, current and resistance should be expressed in units of volts, amperes and ohms, respectively.

V = I R or I = V/R or R = V/I
where
V = Potential Difference in volts
I = Current in amperes
R = Resistance in ohms

Fig. 8 - Ohm's Law

A particle having a charge Q moving through a potential difference V will gain an amount of energy given by the product Q x V. If the charge is given in terms of the number of basic electron charges, then the energy is given in electron volts, abbreviated eV. A proton carries one positive electronic charge. If it accelerates through 500 V potential difference, it would gain an energy of 500 eV. An alpha particle (charge = 2) would gain an energy of 1000 eV or 1 keV under the same conditions.

Electric power is measured in watts (W). Power produced in a resistor is calculated as the product of the current (I) and potential difference (V). Some of these interrelationships are illustrated in Sample Problem 8.

Sample Problem 8

GIVEN:
A 12 volt automobile lamp has a rating of 30 W.
FIND:
The bulb resistance and the current that will flow, in milliamperes.
SOLUTION:
Electric power, W = V volts x I amps. So, I = W / V = 30 W/ 12 V = 2.5 A.
Since 1 A = 1,000 mA, 2.5 A = 2,500 mA.
The bulb resistance can be obtained from Ohm's Law,

$$R = V / I = 12\ V / 2.5\ A = 4.8\ \text{ohms}.$$

Electrical circuits are characterized by the direction of flow of the electrons. If electrons always travel in the same direction, the circuit is said to be a direct current or DC circuit. If electrons have bi-directional flow, changing direction from time to time, the circuit is of the alternating current or AC type. The potential difference measured at a wall outlet would be an example of AC. Figure 9 illustrates some waveforms that might be observed with an oscilloscope and indicates some common terms used to describe waveforms for periodic (repeating) waves.

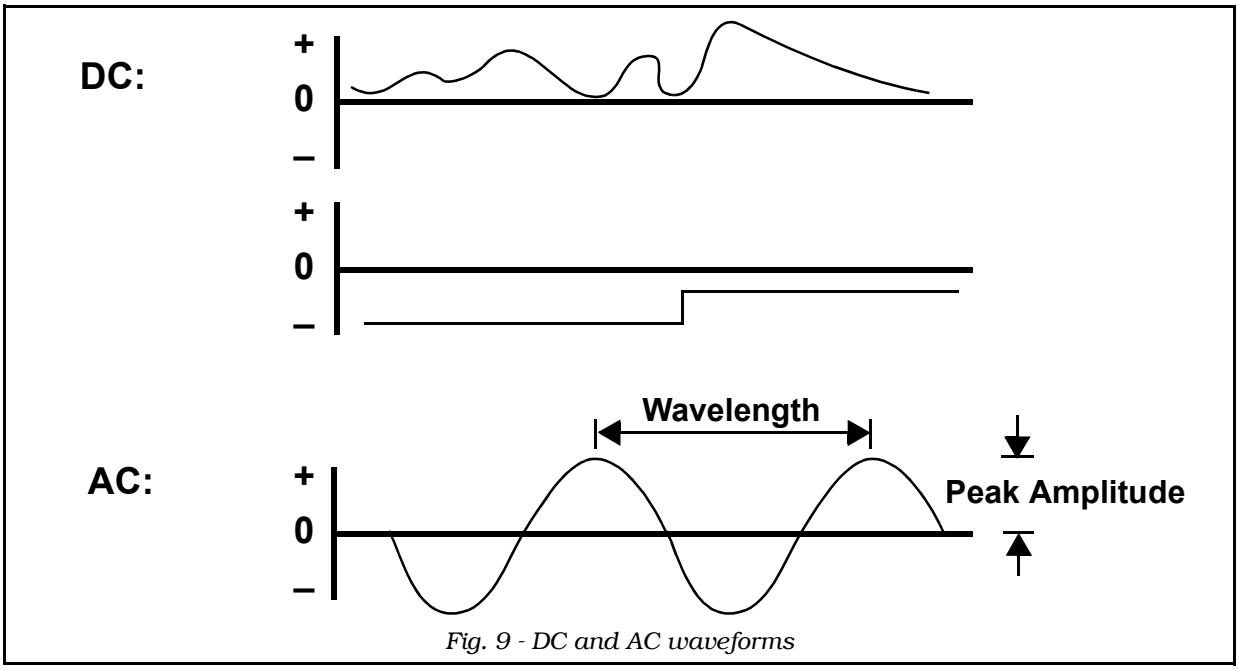


Fig. 9 - DC and AC waveforms

Note that periodic waveforms always have a fixed relationship between their wavelength and frequency. For electromagnetic waves traveling at the speed of light, the equation is:

$$\text{Wavelength} = c / \text{Frequency}. \qquad \text{[Eqn. 1]}$$

In this equation, c represents the speed of light in a vacuum ($c = 3 \times 10^8$ m/sec or 3×10^{10} cm/sec).

Electrical components can be connected together using conductors, e.g., a length of wire, so that a complete circuit can be made that allows the flow of electric current. Figure 10 shows the two common types of circuits, series and parallel. In a series circuit, there is only a single current path. The current in all of the components has the same value. In contrast, a parallel circuit offers more than one path and so the current splits up and different components will receive different currents. Sometimes, series and parallel configurations are both present to form a series-parallel combination circuit.

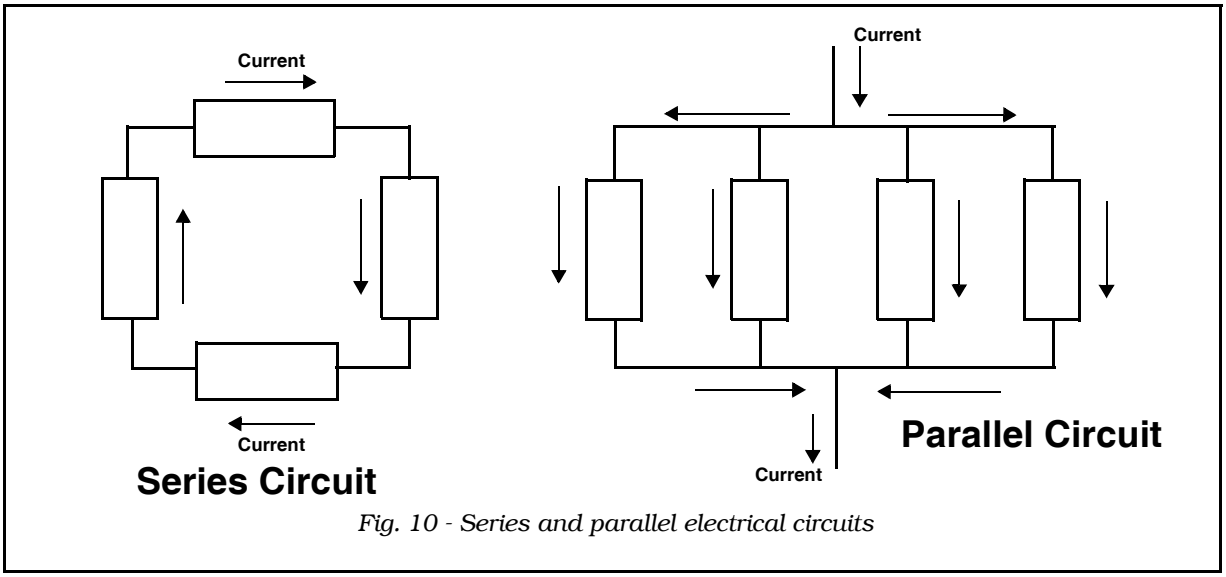


Fig. 10 - Series and parallel electrical circuits

In DC circuits (electron flow is in a single direction), electrical components can be represented as resistors. Currents and potential differences at various points in the circuit can then be calculated with Ohm's Law. In AC circuits (with bi-directional electron flow) the situation is more complex. For the remainder of this section, we will consider only circuits in which the current and potential difference are sine waves. All AC electric power systems world-wide follow this convention.

AC circuits can only be represented completely by including capacitors and inductors along with resistors. A capacitor is an electrical device that is capable of storing electric charge. The maximum amount that can be stored is given by the "capacitance" measured in farads. A one farad capacitor charged up to one volt will contain 1 coulomb of electrical charge. The ability of a capacitor to impede current flow is called the capacitive reactance of the device, X_c and it is measured in ohms. It is inversely proportional to the capacitance of the capacitor.

An inductor is an electrical device that resists a change in the current flowing in an AC circuit. A common example would be a coil of wire where the magnetic field produced by the varying current in each wire loop induces an opposing potential difference in adjoining loops that resists changes in the current flow through the wire. The ability of an inductor to resist current flow changes is given by the "inductance" measured in henrys. A one henry inductor will induce an opposing potential difference of 1 volt if the current through it changes at a rate of one amp per second. The

ability of an inductor to impede current flow is called the inductive reactance of the device, X_L and is measured in ohms. It is proportional to the inductance of the inductor.

The presence of either capacitance or inductance in an AC circuit complicates the analysis. In a resistor, the potential difference and the current are in phase. This merely means that the potential difference and the current reach a maximum at the same time. However, in a capacitor, the potential difference sine wave lags the current sine wave by ¼ of a wavelength. Similarly, in an inductor, the potential difference sine wave leads the current sine wave by ¼ of a wavelength. In an AC circuit containing resistors, capacitors and inductors, the potential difference and the current sine waves are some arbitrary fraction of a wavelength apart. The separation of the two waves is measured by the "phase angle," ϕ. In a given circuit, ϕ depends on the relative amounts of capacitance and inductance. Since a complete cycle is 360°, a quarter wavelength phase angle difference would mean $\phi = 90°$.

Turning our attention now to a review of electric power concepts, it was stated earlier that electric power in a resistor in a DC circuit is calculated as:

$$P(\text{watts}) = V(\text{volts}) \times I(\text{amps}).$$

But the DC case is special since the phase angle is 0° in a resistor. In the AC case, we must deal with a non zero ϕ due to the likely presence of capacitance and inductance in the circuit. In addition, both V and I are sine waves. Thus, at some instant in time, the instantaneous power in the circuit is zero and at other times it has some maximum value. Usually, for practical purposes, we are interested only in the average power (also called the actual power and the real power). This turns out to be:

$$P_{ave}(\text{watts}) = V_{rms}(\text{volts}) \times I_{rms}(\text{amps}) \times \text{cosine } \phi.$$

Here, V_{rms} is called the root-mean-square value of the potential difference, a kind of average. Ordinary voltmeters read the rms value in AC electrical circuits. Also, the cosine ϕ plays such an important role that it is given the special name "power factor." If a given AC circuit only contains resistors, the phase angle $\phi = 0°$. The cosine of 0° = 1 and the average power dissipated in the resistor reduces to the product of the rms potential difference and the rms current, analogous to the DC circuit case. Sample Problem 9 illustrates the application of an electric power calculation.

Sample Problem 9

GIVEN:

In a series circuit containing inductance, ϕ is measured to be 34° when used with a 120 V rms supply.

FIND:

The maximum rms current that could flow in the circuit without exceeding a P_{ave} of 2 watts.

SOLUTION:

The maximum, real power in the circuit will be P_{ave} = 120 V x I x cos 34° = 2 watts. So, I = 2/(120 x cos 34°) = 2/99.5 amps = 0.020 amp rms or 20 mA rms.

There is another way to understand the power factor. It is sometimes useful to define the "apparent power" as:

$$P_{apparent} = V_{rms} \text{ (volts) x } I_{rms} \text{ (amps)}.$$

The apparent power would be the value of the average power if the capacitance and inductance didn't cause the potential difference and current sine waves to get out of phase with each other. In other words, the apparent power represents some kind of a maximum ideal case. From this viewpoint, the power factor is seen to be:

$$\text{Power Factor} = \text{cosine } \phi = \text{Actual Power/Apparent Power}.$$

We will conclude this review of AC power theory by examining the general AC case of the DC Ohm's Law equation, V = I R. As just discussed, in AC circuits we usually represent the potential difference and current by their rms values V_{rms} and I_{rms}. The remaining question is how to represent the resistance R. The answer is that the resistance, R, is transformed into the impedance, Z, in an AC circuit. Then, the AC form of Ohm's Law becomes:

$$V_{rms} \text{ (volts)} = I_{rms} \text{ (amps) x Z (ohms)}.$$

Impedance is just the opposition to flow of an AC current. It can be all resistance, all reactance (capacitive and/or inductive) or a combination. The total impedance is given by:

$$Z = \sqrt{R^2 + (X_L - X_C)^2}.$$

The final topic in Electricity is a brief review of AC power transmission. The most common system used throughout the world's transmission grids is three-phase electric power. Three wires are used, each transmitting a 50 or 60 Hz sine wave, but the three waves are separated from each other by one-third of a cycle (90°). Figure 11 shows graphically the three-phase relationship.

There are advantages to a three-phase system as compared to a single-phase system. The electric utility companies find that it is cheaper as it uses smaller conductors to carry the same power. Many large industrial motors are designed to operate only on a

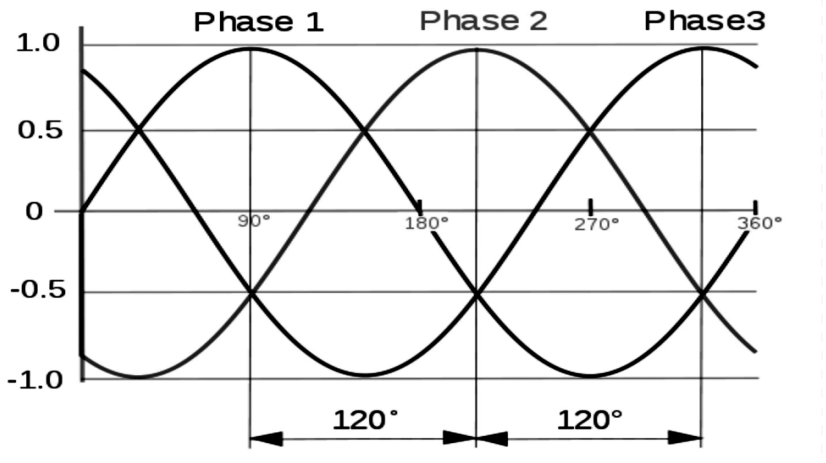


Fig. 11 - Three-phase electric power

three-phase system. The idea for three-phase systems was patented in 1887 by Nikola Tesla, a famous inventor and electrical engineer.

Electronics

The field of modern electronics had its birth in the invention of the vacuum tube by Lee de Forest in 1906. Such tubes are constructed by sealing metal electrodes inside of an evacuated glass tube. Figure 12 is a photo of a couple of "modern" vacuum tubes. A vacuum tube diode con-

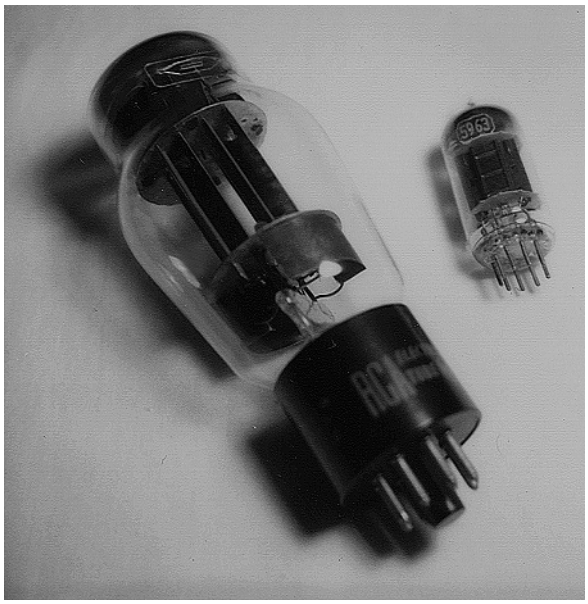


Fig. 12 - A pair of electronic vacuum tubes

tains two basic electrodes. An electrically heated filament acts as a thermionic emission source of electrons and is termed a cathode. The anode or plate attracts electrons by exerting a Coulomb force when it is at a positive potential relative to the cathode. Such a device works as a one-way "check valve" for the current. A diode is thus capable of acting as a rectifier for changing alternating current to direct current. The tube acts like an open switch if the anode is negative with respect to the cathode as the electrons are repelled away from the anode.

A triode is a three electrode vacuum tube. An additional electrode, in the form of a fine mesh screen, is placed close to the cathode. Because of the inverse square term in Coulomb's Law a small potential difference on this "grid" electrode will exert a much larger force on the electrons than even a large positive potential on the anode. Changing the potential on the grid thus causes large changes in the electron current flow between cathode and plate. In this way, small changes in the grid circuit cause

large changes in the anode circuit so the tube acts as an amplifier. Connecting a series of tubes together in the proper way will give an electronic amplifier with a large power gain. Such a device would be used in an audio system to drive a loudspeaker. Amplifiers are also needed in many types of radiation detection instruments used in radiation protection technology. This is due to the fact that the signals from common detectors are very small.

The next era in electronics was ushered in with the invention of the transistor at Bell Labs in 1948. Transistors consist of three very tiny sections of semiconductor material that correspond to the three electrodes of a vacuum triode. The current is produced in an emitter section and flows to a collector section with the flow being controlled by a signal on the base section. Transistors can also be connected together to provide amplifiers with large power gain. Some of the advantages of transistors over tubes are that they operate at lower potentials (thus making portable instruments possible without large battery packs), at lower power levels (reducing problems caused by heat, such as the need for cooling multi-tube circuits), and transistors are much smaller and more rugged.

Another big advance took place with the introduction in the 1970s of integrated circuits. (ICs were invented independently by Kilby at Texas Instruments and Noyce at Fairchild in 1958-59.) In these devices the term "microelectronics" has true meaning. Through the use of successive steps of etching and masking of layered semiconductor slabs, transistors, resistors, capacitors and other circuit components can be packed together on a tiny chip. Individual structures had sizes in the range of 10 to 15 microns - millionths of a meter. The silicon crystals at the heart of the devices are virtually perfect crystals - only 1 atom in typically 10 million-million is out of place in the lattice structure. Impurities are measured in the tenths of a part per billion range. A typical integrated circuit of the 1970s contained thousands of individual electronic components in an area less than a mm on a side.

Taking the next step, entire electronic circuits were encapsulated into a single "chip" component that plugs into a socket in an instrument. Such devices are finding application in many types of digital health physics instruments. Large integrated circuits are used in computer applications. Such "microprocessors" are the basis for electronic calculators and many other consumer electronics products. [Following a 20-year legal battle with the U.S. Patent Office, it was announced in 1990 that Gilbert Hyatt, a Los Angeles engineer, was the inventor of the microprocessor, in 1968, rather than Intel Corporation. Hyatt can now collect royalties for all such devices manufactured since 1968!]

A development of the 1980s was the introduction of VLSI - very large scale integration. These devices contain circuitry as complex as 100

Los Angeles city street maps contained on a single wafer the size of a thumbtack. A single chip contains over a million components. (An Intel 8086 processor of 1978 vintage had 29,000 transistors. By 2007, single chips had over 10 billion transistors!) Individual structures in today's technology are less than 0.8 micrometers in size. Figure 13 illustrates some common transistor packages, integrated circuits and microprocessors. Figure 14 is a photomicrograph showing an integrated circuit wafer with leads welded to the pads.

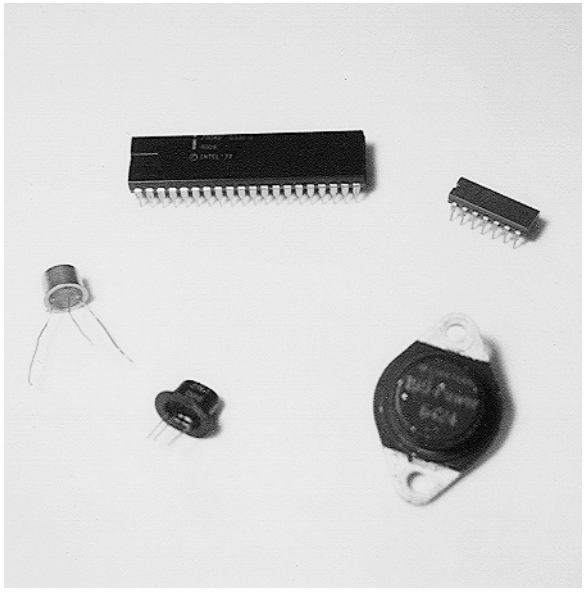


Fig. 13 - Semiconductor devices

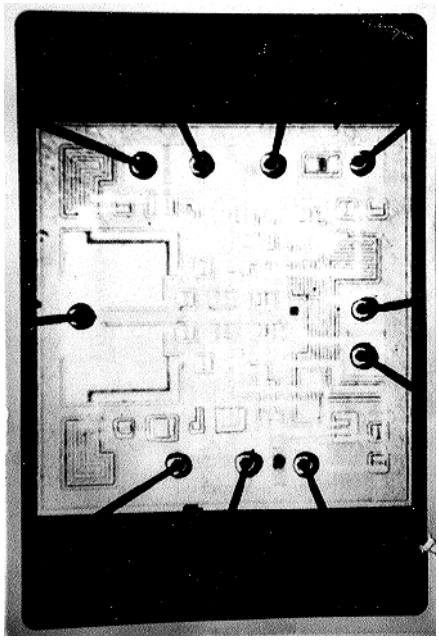


Fig. 14- IC wafer with leads attached

Other Sciences

Biology

This section will review some of the basics of biology applicable to radiation protection technology. In Chapter 4 the effects of ionizing radiation on cells and humans will be examined in some detail.

The basic building blocks of animals are cells. Cells are usually composed of protoplasm which is mostly water (about 80%) which contains carbohydrates, fats, proteins, salts and dissolved gases. The inner structures, called organelles, are contained within a sac-like membrane. The membrane allows the passage of various ions, nutrients and wastes into and out of the cell. Through metabolic activity, the cell can generate sources of energy, repair damage, construct needed structures and produce daughter cells by cell division (mitosis). Before dividing, the cell prepares a

duplicate set of chromosomes, using the existing set as a pattern. The normally diffuse chromosomes then gather into clumps, and the two identical sets are pulled to opposite ends of the cell. The cell then pinches off in the center, and two daughter cells begin the cycle anew.

The human organism is a complex collection of cells organized into various organ systems and different types of tissues. There are four different recognizable tissue types in the human body. Epithelial tissues all have cells closely joined to each other and they form the internal and external body surfaces. Connective tissues contain cells which are widely separated by extracellular material. Included in this group are the various blood cell types, ligaments, cartilage and bone. Humans have three different types of muscle tissues. These include smooth, cardiac (heart) and skeletal. Muscle cells are capable of contraction under electrical stimulation. The last tissue type is nervous tissue. The neurons are the basic cell type. They are able to conduct an electrical impulse signal.

The major organ systems of the body are very briefly reviewed as follows:

1) INTEGUMENTARY - Includes skin, hair and sweat glands. Holds other organs in and protects them from outside environmental agents.

2) SKELETAL - The bones and joints. Provides the basic support for other organs and attachment points for muscles.

3) CIRCULATORY - Includes the heart and the blood and lymph vessels. Allows for the transport of oxygen and nutrients to tissues and removal of waste products.

4) DIGESTIVE - The mouth, esophagus and small & large intestines. Dissolves food and breaks it down to usable components which are then absorbed into the bloodstream.

5) URINARY - Includes kidneys and bladder. Filters dissolved wastes resulting from cell activity.

6) REPRODUCTIVE - The gonads (ovaries & testes) and uterus. Produces ova and sperm which combine to create a new organism (embryo).

7) ENDOCRINE - Internal glands. Produce and release hormones into tissues or blood. Includes pituitary, thyroid and adrenals. The hormones are vital to proper metabolism and growth.

8) RESPIRATORY - The airways and lungs. Provide a system for oxygen and carbon dioxide exchange in the blood.

9) MUSCULAR - Body muscles. Provide for locomotion, lung inflation and heart contraction.

10) NERVOUS - The brain, spinal cord and nerve fibers. Transmits signals for conscious and subconscious control of body functions.

Chemistry

A few concepts from the field of chemistry will conclude this chapter. At the present time (the beginning of the 21st century), 116 different chemical elements have been identified. The heaviest elements have, of course, been artificially produced in laboratories. A substance is classed as an element if it cannot be changed into simpler substances through chemical reactions or chemical changes. Compounds are

substances made up of combinations of elements in fixed, simple numerical proportions. Water is a compound composed of two parts of the element hydrogen and one part of the element oxygen.

The atomic weight of an atom is the weight of one atom compared to a reference standard atom. At present, the reference chosen is the most abundant isotope of carbon which is assigned a chemical atomic weight of exactly 12.0000000. Based on this standard, an oxygen-16 atom weighs 15.9949 and uranium-238 weighs 238.0508 atomic mass units (one amu = 1/12 the weight of a carbon 12 atom). The molecular weight of a molecule is the sum of the atomic weights of its constituent atoms. The gram molecular weight, or mole, is just an amount of a compound equal to the molecular weight in grams. For water, the molecular weight would be 1 + 1 + 16 = 18 so a mole of water weighs 18 grams. It has been shown that a mole of anything contains Avogadro's number, 6.025 E23 of molecules. In gaseous form, a mole of a substance occupies 22.4 liters volume at standard temperature and pressure (STP).

The elements are composed of a nucleus surrounded by electrons. The electrons are thought of as occupying various shells at fixed distances from the nucleus. The innermost electron shell is called the K shell and can hold, at most, two electrons. The succeeding shells are labeled L, M, N, etc. The maximum number of electrons per shell is given by twice the square of the shell number. (The M shell holds 2 X 3 X 3 = 18). Elements having the same number of protons but different numbers of neutrons are called isotopes.

If the elements are arranged in order of increasing atomic number (number of protons) and then divided at the points where the chemical properties recur, the arrangement into rows is called the periodic table of the elements. The similarities in properties of elements is seen to be due almost completely to the similarities in the numbers of electrons in their outermost or valence shell.

The valence of an atom is the ability to bind to other atoms. It is an integer. It represents the number of hydrogen atoms that one atom of some element can combine with. In water, an oxygen atom binds with 2 hydrogen atoms so the valence of the oxygen is two in this case.

Through chemical reactions, atoms can combine in different ways to produce other compounds. These reactions involve the interchange of electrons. If an element loses electrons in a reaction, it is said to be oxidized. Similarly, the gain of an electron is called a reduction reaction. Such reactions will be mentioned in later chapters as they are of relevance to the field of dosimetry.

To determine whether a solution is acidic or basic, we often measure the pH. The pH is defined to be the negative of the common logarithm of the concentration of the H^+ ion, i.e.,

$$pH = -\log_{10} [H^+].$$

It is known that a solution of pure water also has a small concentration of H^+ and OH^- ions which naturally result from dissociation of the H_2O molecules. These two ionic concentrations are exactly equal and have the value of 1.0×10^{-7} moles per liter. Thus, the pH of pure (neutral) water is $-\log 1.0 \times 10^{-7} = -(-7) = 7$. Acids have a pH which is less than 7.0 and bases have a pH larger than the neutral value of 7.0.

Problem Set

1. Write down 100,098, 0.00345 and e in scientific notation.

2. Calculate the numerical product of Avogadro's number and the nuclear diameter in scientific notation and ordinary notation.

3. If X is equal to 0.89 and Y is an angle of 30°, what is the quotient

$$X^{-3.5} / 0.5 (\sin Y)^2 ?$$

4. A voltmeter has a meter face with 20 scale divisions covering the range 0 to 2 volts. What is the largest number of significant figures that would be justified in reading this meter?

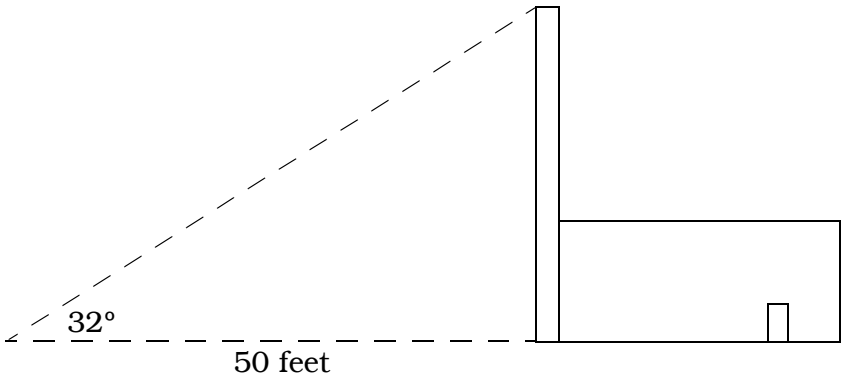
5. What units would properly result from the quotient of ((R-m)/(mA-sec)) by ((V/sec) X (1/R))?

6. Find t if $2.4\ e^t = 8.0$.

7. If 2.47 E2 = x to the 4th power, find x.

8. Calculate $e^{3\cdot}$

9. A rad protection technician measures out a distance of 50 feet from the base of a rad waste incinerator smokestack. He then determines that the top of the stack is at an observation angle of 32°. How tall is the stack?



10. Find n when 17 n/tan 235° = (cost 41°)(sin -89°) where the angles are in degrees.

11. A molecule is the smallest unit of a ? which still has all of its physical and chemical properties.

12. Calculate the work done in lifting a 26 lb. lead brick 3 feet in 2 sec.

13. What would be the velocity just before impact of a falling 10 lb. geiger counter that hit the pavement with a kinetic energy of 4000 (lb. mass)(ft/sec)^2?

14. What is the more common way of writing the following equation?

$$v = (2\ E/m)^{1/2}$$

15. An electron accelerator produces 1.5 E14 electrons per second at the target. What is the target current?

16. If the target current in problem 15 is conducted to ground potential through a 3 megohm resistor, what potential difference would appear across the resistor?

17. In 1991, 111 operating U.S. nuclear power reactors generated 613,000 GW-hrs (gigawatt-hours) of electric power with an average capacity factor of 71% (ratio of electricity generated to electricity that could have been generated at continuous full-power operation). Calculate the combined average AC current flowing through 115 volt AC transmission lines to carry this energy. (Assume that the voltage and current are exactly in phase so Ohm's Law applies directly.)

18. An AC circuit is measured to have 1200 ohms of inductive reactance and 5400 ohms of resistance. If 0.044 amps rms of current flow when this circuit is connected to a 240 V rms AC supply, what capacitive reactance is present?

19. By what amount is the magnitude of the Coulomb force changed if one of the charges is doubled and the separation halved?

20. Discuss the process of mitosis.

21. Name four major human organ systems and briefly describe their functions.

22. What occurs in a reduction chemical reaction?

23. What volume would be occupied, at STP conditions, by 2 moles of radioactive Kr-85 gas? What would be the mass of this gas?

S-1. A neutron is composed of particles having fractional charges that are called?

S-2. Briefly describe a microprocessor and give one application of such a device.

Other Resources

Basic Mathematics:

Look for basic review books at local bookstores. “Schaum’s Outline Series in Mathematics” and the “Barnes and Noble Series” have been found to be useful references by many students.

Basic Science:

The local library should have a few beginning college level textbooks. Look for titles in General Physics, Modern Physics, General Biology, Human Anatomy, and General Chemistry

Quarks and Leptons:

Visit the website of the Particle Data Group of Lawrence Berkeley National Lab sponsored by the U.S. Department of Energy and the National Science Foundation at http://ParticleAdventure.org for the latest developments and for a very detailed but readable description of the “Standard Model of Matter.”

2

Theory of Radioactivity

Outline of this Chapter

Chapter Summary

This second chapter begins the actual material that constitutes radiation protection technology. Chapter 1 was intended as a review of topics in mathematics and science. The new material begins with the historical evolution of the present ideas of the nature of the atom. An understanding of the Bohr Model of 1913 is sufficient for the needs of health physics technicians.

Then, the problems in unraveling the secrets of the atomic nucleus are tackled. In their attempts to probe the nucleus, physicists discovered the neutron and how to cause nuclear reactions. A study of binding energy led to the hypothesis that heavy atoms might be split, causing a large energy release. This led to the establishment of the Manhattan Engineering District which developed nuclear reactors and nuclear weapons.

Natural radioactivity was discovered just before the close of the 19th century by Becquerel. Eventually three natural radiations were discovered – alpha, beta and gamma. Alpha and gamma decay were readily understood. Beta decay was understood theoretically in 1931 but it took until 1953 to experimentally catch the necessary neutrino, a particle predicted in 1931 to accompany every beta decay. Electron capture and internal conversion decay are also discussed briefly.

A single universal law mathematically describes all radioactive decay. Mathematically it is an exponential law. The activity of a sample or the number of radioactive atoms in a sample decreases as e^{-time}. Knowing the activity at a particular time, and the half-life of the sample, the activity at any later or earlier time can be easily calculated.

On occasion, a parent radionuclide decays to a radioactive daughter that also decays to form a decay chain. The long decay chains of heavy radionuclides found in nature are discussed including the equilibrium conditions they sometimes exhibit.

Atomic Structure

Thomson Model

Around the close of the nineteenth century, a number of basic facts were known about the atom as a result of various experiments. In 1897, English physicist Sir J.J. Thomson was able to demonstrate that all atoms contained negatively charged particles about 1/2000th the mass of a proton. He gave them the name "electron." X-ray scattering work gave the number of electrons per atom, i.e., the atomic number. Millikan's famous oil drop experiment in 1909 established the actual charge on a single electron, the electronic charge, as $e = 1.6 \times 10^{-19}$ coulombs. Thomson proposed a model for describing the internal structure of atoms. The model became known as the Thomson Plum Pudding Model. It pictured an atom as being a semi-rigid spherical positively charged "pudding" (as in English cooking – a heavy bread) in which the negative electrons were randomly embedded as "plums." (See Figure 1). The size of the atom was correctly deduced to be about 10^{-8} cm.

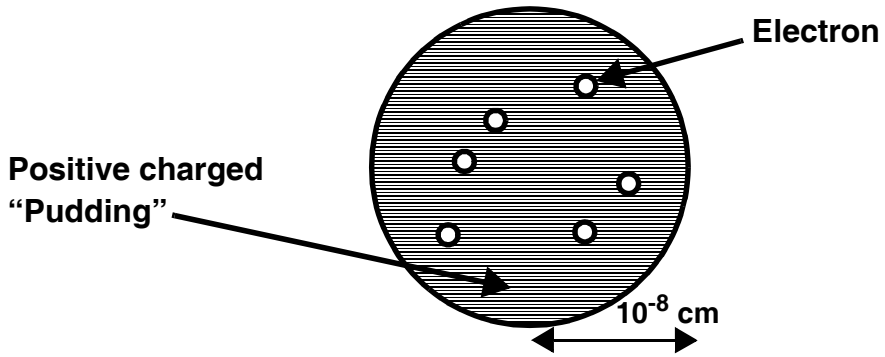


Fig. 1 - Thomson's Plum Pudding atomic model (1909)

Rutherford Model

In 1911, another English physicist, Ernest Rutherford, conceived of a way to experimentally study the internal structure of the atom. Rutherford had worked 12 years earlier under J.J. Thomson on a scholarship at Cambridge in the Cavendish laboratory. His reasoning can be understood by reference to the hypothetical experiment that is pictured in Figure 2. Imagine two identical cardboard boxes. One is stuffed completely full with sawdust. The other contains the same total mass as the first, except the mass is in the form of steel ball bearings which are suspended by strings throughout the volume of the box. According to Rutherford's thinking, the box which contained the ball bearings could be discovered by firing rifle bullets into the

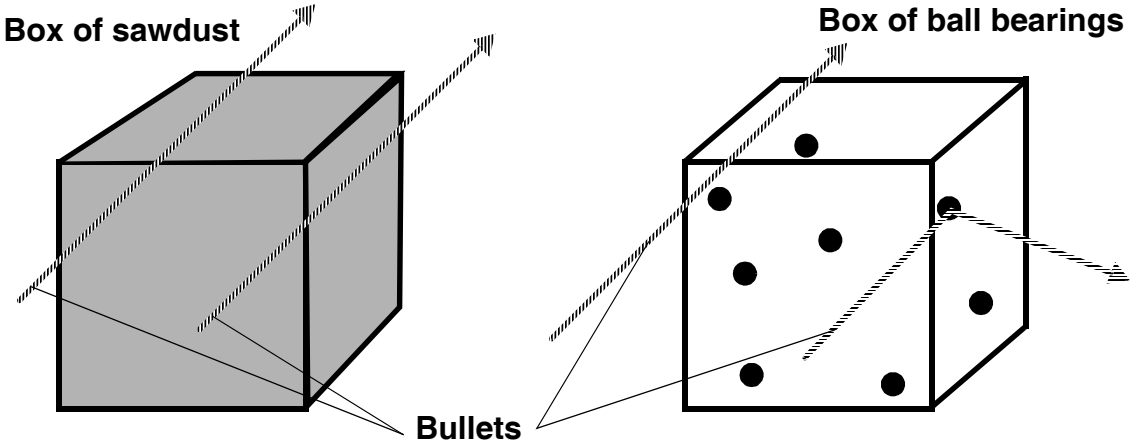


Fig. 2 - The basis of Rutherford's scattering experiment

ends of the boxes, and watching the trajectories of bullets leaving the back side. The box containing sawdust would slow up the bullets slightly but would not produce any other effect. In the case of the box containing ball bearings, most of the bullets would also exit along the line of entry. However, occasionally a bullet would strike the concentrated mass of a ball bearing and would undergo a large change in direction, possibly leaving through the side of the box. According to Thomson's model, the atom was like the sawdust filled box with the contained mass spread out uniformly over the atomic volume. If small enough bullets could be found, the model could be tested by following Rutherford's suggestion.

The actual experiment in "Rutherford Scattering" was carried out two years later by Geiger and Marsden (associates of Rutherford) using the apparatus sketched in Figure 3. For bullets, they selected alpha particles from radioactive radium. The

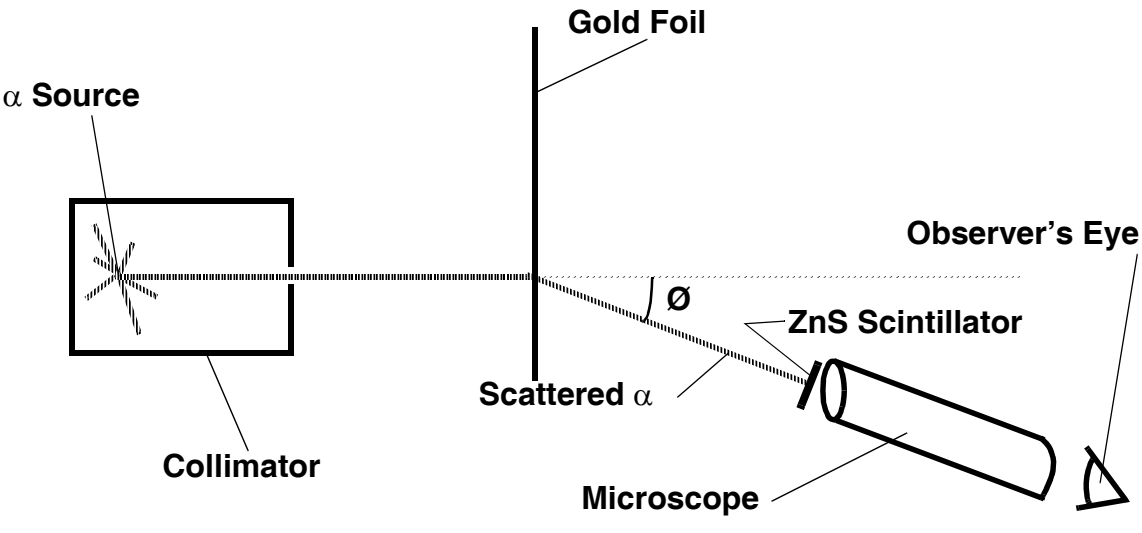


Fig. 3 - Geiger and Marsden's experimental setup

atoms that were studied were gold, in the form of a very thin foil only a few atoms thick. This prevented confusion in the analysis due to the possibility of more than one gold atom deflecting an alpha particle. Gold has an atomic mass of about 200 compared to the mass of 4 for the alpha particles (which are just helium nuclei, as we will see shortly).

The paths of the exiting alphas were found by using a ZnS (zinc sulfide) scintillating screen. This material gives off a tiny flash of light when struck by an alpha particle. Geiger and Marsden used a microscope focused on the ZnS screen to detect the emerging alphas.

Before the experiment got underway, Rutherford calculated the maximum angle of deflection possible for any alpha particle assuming the Thomson Plum Pudding Model was correct. According to the laws of physics, he calculated that no alpha particle could be deflected more than 1/100,000th of one degree. This small deflection would be almost impossible to measure experimentally. When the results were in, it turned out that alpha particles were occasionally observed with deflections of well over 90°. Using the new results, Rutherford calculated that most of the mass of the atom must be concentrated in a size of about 10^{-12} cm, that is, in an object 10,000 times smaller than the atom itself. He then proposed a new model for describing atoms. The Rutherford Model is shown in Figure 4. He used the term "a whirligig

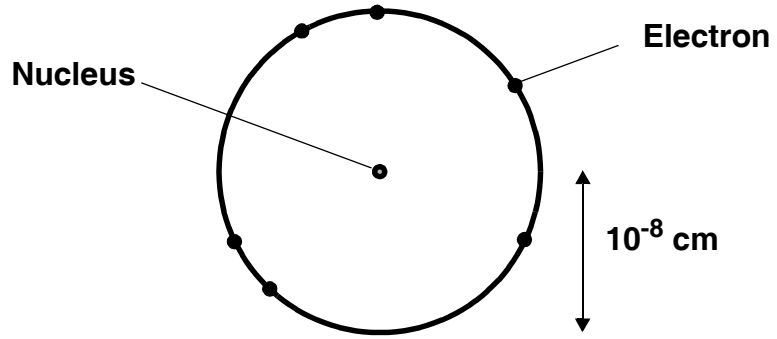


Fig. 4 - Rutherford's atomic model (1913)

affair" to describe it.

In this model, the mass is concentrated in what he named the nucleus. The electrons were pictured as flying around it on the outside. To get some idea of the size scale involved here, if the atom were magnified to be several miles across, the nucleus would only be the size of a grapefruit. See Sample Problem 1.

Ernest Rutherford started his experimental physics career in New Zealand in 1893. Figure 5 shows him at work in his laboratory in 1894. Here he discovered a method for detecting wireless signals. This research was abandoned when Roentgen announced the discovery of x-rays. Marconi later used the method for international wireless telegraphy.

While working as a Professor of Physics at McGill University in Canada, Rutherford and Frederick Soddy discovered radioactive elements transform into other forms with unique, different half-lives. He was awarded a Nobel Prize (in Chemistry) for this discovery.

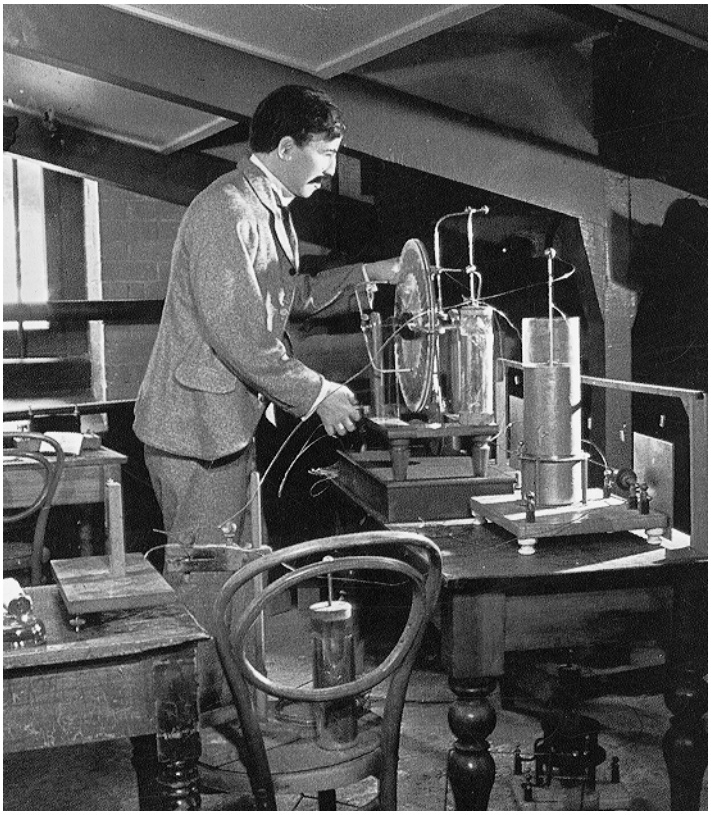


Fig. 5 - Rutherford at work in New Zealand, 1894

Sample Problem 1

GIVEN:

In the Rutherford Model of an atom, the nucleus has dimensions of about 10^{-12} cm while the atomic size is around 10^{-8} cm.

FIND:

The % of the atomic volume that is occupied by the nucleus.

SOLUTION:

The volume of a sphere is given by $4/3\ \pi\ r^3$ where r is the radius. Thus, the nucleus has a volume of $4/3\ \pi$ x 10^{-36} cm^3 and the atom $4/3\ \pi$ x 10^{-24} cm^3.

The % occupied by the nucleus = 100% x Nuclear Vol./Atomic Vol.

= 100% x 10^{-36} $cm^3/10^{-24}$ = 100 x $10^{-36\ +\ 24}$ % = 1 x 10^{-10} %!

Bohr Model

In 1913, Danish physicist Niels Bohr, who had studied under Rutherford, made use of a new theory of mathematical physics called quantum mechanics to develop the Bohr Model of the atom. In this model, Bohr assumed that electrons

revolving around the nucleus had to have integer values of angular momentum. In this case, only a limited number of orbits are allowed. The electrons are thus pictured as occupying "electron shells" around the nucleus. The smallest diameter (innermost) shell is called the K shell; the next larger is named the L shell, etc. Figure 6 shows the Bohr atom.

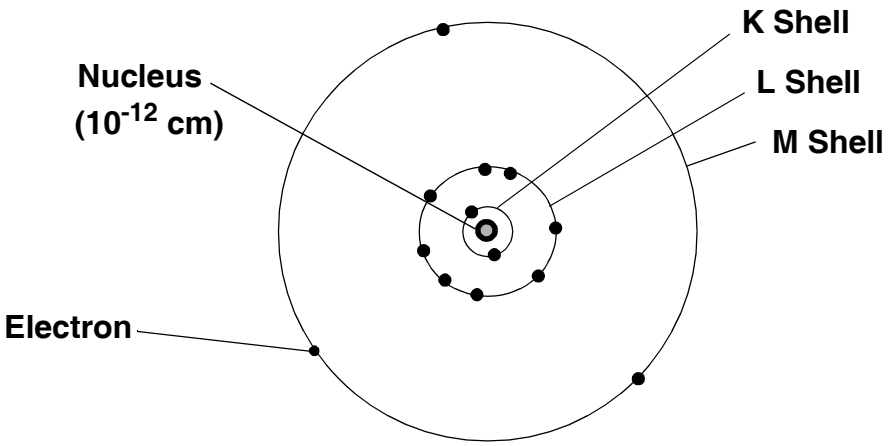


Fig. 6 - The Bohr atomic model (1913)

Although the Bohr Model is, today, not the most comprehensive or accurate model of atomic structure, it does explain atomic structure sufficiently for the needs of radiation protection technology. The existence of characteristic x-rays is explained as resulting from electrons jumping from a higher shell to a lower one in energy. The energy difference between the upper and lower shells is given to the x-ray. Since each shell in an atom has a well defined characteristic energy, these energy differences will all have the same value for atoms of the same element (a characteristic "fingerprint" for each element). Measurement of the energy of these x-rays thus determines the elements in a sample. This is the basis for an analytical technique called x-ray fluorescence analysis.

Nuclear Structure

Nuclear Model

Having obtained a reasonable understanding of the nature of atomic structure, the physicists next turned their attention to trying to gain a better picture of the structure of the positively charged atomic nucleus. One of the most fruitful suggestions was made by Rutherford. Having been so successful in uncovering the secrets of the atom by his suggested alpha scattering experiment, he again suggested using alpha particles to probe the nucleus. In 1919, while bombarding nitrogen gas with ^{214}Po alpha particles, Rutherford observed that a hydrogen nucleus was released approximately once for each 50,000 collisions. He proposed that the particle be

named a proton (from Greek meaning “first”). Thus, the proton was confirmed as being at least one of the nuclear ingredients. It turns out that he was seeing the first artificially induced nuclear reaction in history. By remembering that the equations for nuclear reactions must balance like chemical equations, it is clear that Rutherford had caused nitrogen atoms to transform into oxygen atoms. The reaction is shown schematically in Figure 7.

$$^{4}_{2}\mathrm{He} + {}^{14}_{7}\mathrm{N} \rightarrow {}^{17}_{8}\mathrm{O} + {}^{1}_{1}\mathrm{H}$$

Fig. 7 - First artificial nuclear reaction (Rutherford, 1919)

The subscripts give the atomic number or number of positive charges on the nucleus, and the superscripts give the atomic mass numbers of the nuclei. Since one element is converted to another (the atomic number changes), this is called a TRANSMUTATION reaction. Through induced nuclear reactions, the dream of the ancient alchemists has been realized – conversion of one element to another.

In 1930, Bethe and Becker were shooting alpha particles at a beryllium target. They discovered a mysterious penetrating radiation leaving the interaction site. They observed that if it was allowed to strike a block of paraffin, a proton was ejected. Nothing that was known at the time had similar properties. This baffling situation was resolved in 1932 by Chadwick, a former student of Rutherford, in confirming the existence of a new particle named the neutron, a contraction of the term “neutral proton.” It was thought to weigh approximately the same as a proton, but carried no net charge. This accounted for its very high penetrating ability. With this understanding, the reaction caused by Bethe and Becker was as shown in Figure 8. This is again a transmutation reaction – beryllium is transformed into carbon.

$$^{4}_{2}\mathrm{He} + {}^{9}_{4}\mathrm{Be} \rightarrow {}^{12}_{6}\mathrm{C} + {}^{1}_{0}\mathrm{n}$$

Fig. 8 - The 1932 discovery of the neutron

In a lecture in 1920, Rutherford proposed the existence of a neutral particle formed by a combination of an electron and a hydrogen nucleus.

With the discovery of the neutron, sufficient understanding was obtained of the nucleus to comprehend most of the reactions and radiations of interest to radiation protection. The major ingredients of the nucleus are listed with their properties in the chart in Figure 9. Because the masses of nuclear particles are small compared to everyday experience, a special system of measuring mass has been introduced. The basic unit is 1/12 the mass of a ^{12}C atom and is called an “atomic mass unit” or amu. One amu has a mass energy equivalent of 931.48 MeV. The mass energy is the equivalent energy that would be released if all the mass were converted to pure energy.

PROPERTY	Proton	Neutron
Net Charge (e)	+1	0
Net Charge (Coul.)	1.6×10^{-19}	0
Rest Mass (kg)	1.673×10^{-27}	1.675×10^{-27}
Rest Mass (amu)	1.00727	1.00866
Mass Energy (MeV)	938.25	939.55
Spin	1/2	1/2
Magnetic Moment	+2.79	-1.19

Fig. 9 - The nuclear particles

The magnetic moment describes how the particle behaves in a magnetic field. If it has a magnetic moment that is not zero, a particle will tend to align with an external magnetic field (similar to a compass). Since the neutron has no net charge, it was expected by early physicists that it would have zero magnetic moment. The value of -1.19 was thus a big surprise. It indicates that even though the neutron carries no net charge, it must contain equal amounts of positive and negative charge separated by a small distance within the particle. Thus, it acts like a small bar magnet in an external field. Looking back from the perspective of the present day, this unexplained neutron magnetic moment was an indication that the neutron was a much more complex particle than first suspected. Modern theories of quarks finally explain the non-zero magnetic moment. Neutrons are now pictured as containing two "down quarks" of -1/3 e charge each and one "up quark" with charge +2/3 e. The separated positive and negative charges produce the magnetic moment for the particle.

Also note that the mass of a proton or a neutron is much larger than the sum of the three constituent quarks that compose them (see table in Figure 4 of Chapter 1). The reason is because most of the mass of a proton or neutron is from the energy needed to confine the quarks inside of them.

In summary, the nucleus is now thought to contain the atomic number, Z, worth of protons and the neutron number, N, of neutrons. The Z determines the chemical element where hydrogen has Z = 1, helium has Z = 2, etc. The atomic mass number, A, is related to the Z and N as

$$A = Z + N.$$

With the help of nuclear accelerators, scientists continue to make additions to the upper end of the periodic table of the elements. By colliding heavy ions together at high energy, new elements can be artificially produced. For example, in accelerator experiments in 2002 and 2006 at Dubna, Russia, laboratory scientists accelerated ^{48}Ca ions (Z = 20) to high kinetic energy and then produced collisions with ^{249}Cf ions (Z = 98). In a

very small number of cases, the ions fused together to form a radioactive atom of element number 118 (20 + 98 = 118). The newly formed atoms all decayed in about a millisecond. This is the highest atomic number element produced and verified as of 2010.

Atoms which have the same Z but differ in N and A are called isotopes. It should be noted that the atomic mass number A is NOT THE SAME as the chemical atomic weight, which is the average of all the isotopes of an element weighted according to their relative abundances.

Occasionally, you may encounter the term isotones. Isotones are nuclei which have equal numbers of neutrons. For the sake of completion, isobars are nuclei with the same atomic mass number, A.

There are frequent occasions when it is not necessary to distinguish between neutrons and protons in a nucleus. In this case, a generic term, "nucleon," is used. A nucleon refers merely to a nuclear particle, which can be either a neutron or a proton.

Nuclear Stability

A stable or non-radioactive nuclide is one whose atoms do not decay. But from a theoretical standpoint, it would be necessary to observe these atoms for an infinite time span to be absolutely sure that they were stable. In other words, nuclei which are actually radioactive but have extremely long half-lives may be mistaken for stable nuclei unless a careful measurement is made. In recent years, nuclei which had formerly been considered stable have been found to actually be radioactive. An example is Lanthanum-138 which turns out to have a half-life of 10^{11} years, about 7 times the estimated age of the universe. Perhaps the "granddaddy" of them all is Tellurium-128 with a measured half-life of 1.5×10^{24} years! (As of 2010, the age of the universe has been determined to be 13.7 billion years.)

Aside from these unusual cases, it is possible to make a list of nuclei which in all likelihood are truly stable. If the neutron number N and atomic number Z are calculated for each of these, an interesting pattern emerges. A graph of N versus Z for stable nuclei would have the general shape shown in Figure 10.

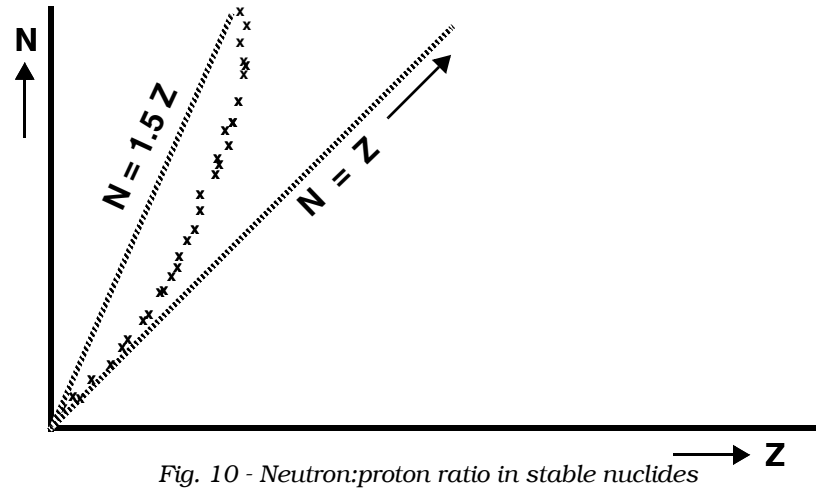


Fig. 10 - Neutron:proton ratio in stable nuclides

The dashed line at 45 degrees represents nuclei which have Z = N, i.e., equal numbers of protons and neutrons. Notice that for light nuclei (elements below iron in the periodic table) the neutron and proton numbers are often equal. As the atomic number increases, however, the stable nuclei gradually fall more and more above the line of Z = N. For the heaviest stable nuclei, N is approximately 1.5 times Z. This behavior is explained by recalling that protons all carry a positive charge and hence will strongly repel each other, according to Coulomb's Law, when forced close together in a nucleus. This highly disruptive force is overcome in stable nuclei by the presence of a force even stronger than the Coulomb force. Within nuclei, a strong attractive force, called the Nuclear Force, acts between neutrons, between protons and between a neutron and proton. By adding relatively more neutrons than protons, additional "glue" is added to heavy nuclei without causing more instability by adding more positive charge. It should also be mentioned that the nuclear force is an extremely short-range force. It acts over a distance equal to a couple of proton diameters at most, and rapidly falls to zero beyond that. The nuclear force is responsible for the binding energy that holds the nucleus together.

Another use can be made of Figure 10. In most cases, radioactive decay takes place in such a way that the resulting daughter nucleus is more stable than the parent. Thus, particular decay modes are favored in which the daughter moves closer to the stability line. Fission products result from the splitting of very heavy nuclei, e.g. ^{235}U. The heavy parent nuclide has many more neutrons than protons as seen in Figure 10. When one of these nuclides splits into two pieces, each "fission fragment" will then have a large neutron excess since middle of the table elements have closer to equal numbers of neutrons and protons. Consequently, these fragments will be negative beta emitters as some neutrons will transform into a proton and an electron (the beta particle). Therefore, a more favorable (decreased) neutron to proton ratio results.

The other side of the coin is represented by activation products in positive ion nuclear accelerator facilities. The bombardment of accelerator targets and accelerator components by positive ions frequently produces radioactive daughter products following neutron emission. As a result, they tend to have a deficiency of neutrons and so decay modes are favored in which the neutron to proton ratio is increased, e.g., positron emission. In positron decay, a proton "converts" into a neutron and positron. [NOTE: Many nuclide charts, such as the General Electric "Chart of the Nuclides" included as part of the PRC Radiation Protection Technology Self Study Course, plot nuclides on a graph of atomic number (vertical axis) vs. neutron number (horizontal axis). This is the reverse of Figure 10 which plots neutron number on the vertical axis.]

Binding Energy

The amount of energy that holds the constituent neutrons and protons together can be calculated fairly readily. If a table of nuclear masses of the isotopes is consulted, it is soon clear that a given nuclear mass is always less than the sum of the masses of the N neutrons and Z protons that make it up. The difference in mass is

called the mass decrement. This is the mass that has gone into the binding energy.

Many years ago Einstein declared the equivalence of mass and energy in his famous equation, $E = mc^2$. According to Einstein, mass and energy are just two different expressions of a more fundamental substance, "mass-energy." Thus, it is theoretically possible to change between forms, that is, matter into energy and energy into matter. In the next chapter, evidence will be presented to show that this interchange between mass and energy does occur often in the laboratory. The binding energy of a given nucleus is thus calculated by multiplying the mass decrement by the square of the speed of light, c. A mass-energy calculation is shown in Sample Problem 2.

Sample Problem 2

GIVEN:

A chunk of iron has a mass of 1 kg.

FIND:

The energy that would be released if all the mass were converted to energy. Also, what energy would be equivalent to a single proton from this mass?

SOLUTION:

The Einstein equation applies directly. $E = mc^2$. Here m = 1 kg. c, from Chapter 1, is 3×10^8 m/sec. So, E (J) = 1 (kg) x $[3 \times 10^8 \text{ (m/sec)}]^2 = 9 \times 10^{16}$ J. For the proton, m = 1.673×10^{-27} (kg) from Figure 9. So, E (J) = 1.673×10^{-27} (kg) x $[3 \times 10^8 \text{ (m/sec)}]^2 = 1.506 \times 10^{-10}$ J. Using the conversion factor from page 12, this energy is $E = 1.506 \times 10^{-10}$ J x 6.242×10^{18} eV/J = 9.4×10^8 eV or about 940 MeV. {A more precise value for c would give the Fig. 9 value.}

An extremely important lesson is learned if the binding energy calculation is carried a bit further. If the total binding energy for a nucleus is divided by the mass number, A, the result will be the average binding energy per nucleon (proton or neutron). This actually represents the amount of mass donated by each proton and neutron to build the nucleus. (Yes, those protons and neutrons incorporated into nuclei really WEIGH LESS than "free" protons and neutrons). A graphical plot of the average binding energy per particle versus the mass number is shown in Figure 11.

The fact that this curve reaches a maximum near the middle and sags to lower values at either end of the periodic table predicts the possibility that the energy contained in matter CAN BE RELEASED. The two processes by which this can occur are called fission and fusion. An understanding of the implications of this graph led to the Manhattan Project and the development of nuclear power.

Before showing how the average binding energy graph predicts the energy released in fission or fusion, a brief historical review of the discovery of fission is in order. In 1934, Enrico Fermi was conducting experiments in which he bombarded uranium with neutrons in an attempt to artificially produce transuranic elements, i.e., elements with an atomic number greater than that of uranium. In 1938, Hahn, Strassman and Ms. Lise Meitner began similar experiments, based on Fermi's work, in Berlin. Shortly thereafter, as a result of increasing tensions in Germany preceding World War II, Ms. Meitner was able to escape into the Netherlands. The experiments continued and in December, Hahn wrote to Meitner (then in

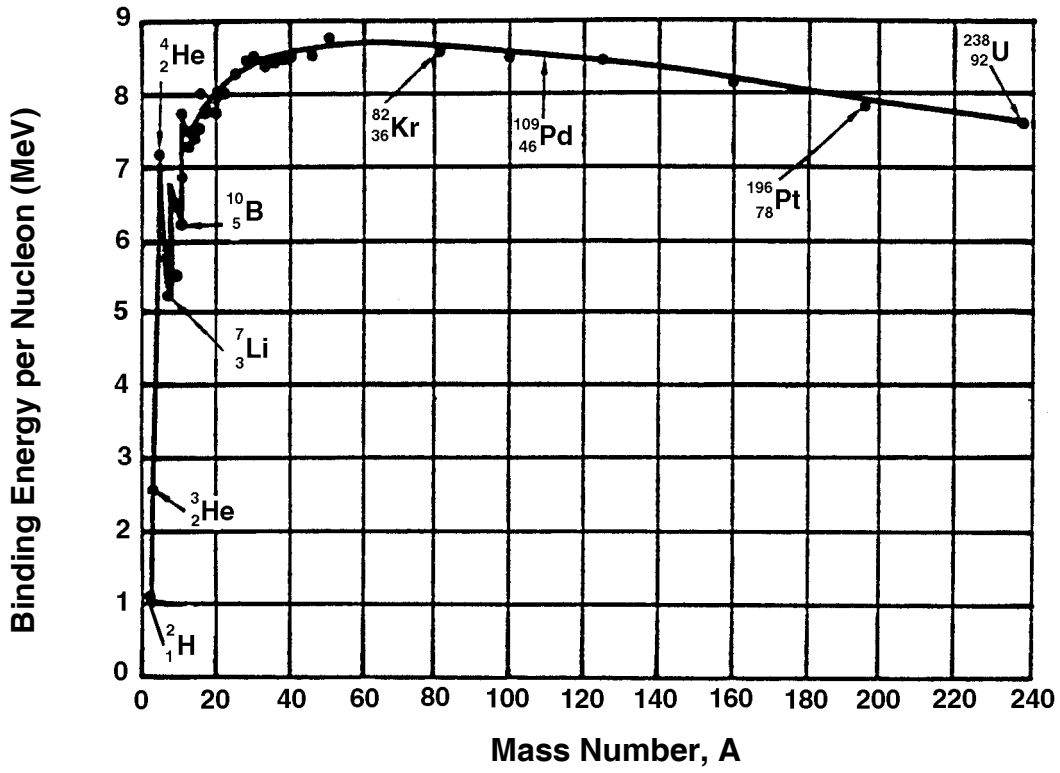


Fig. 11 - Average binding energy per nucleon

Stockholm, Sweden) that they were able to chemically identify barium in the products of a uranium bombardment experiment. Meitner spent the Christmas holiday with her nephew Otto Frisch, a physicist currently working with Niels Bohr in Copenhagen. She was the first to suggest that this was an example of the conversion of matter into energy as predicted by Einstein. On New Year's Day, Meitner returned to Stockholm and Frisch to Copenhagen. Two days later, Frisch was able to describe the experimental results to Bohr.

In a matter of days, Niels Bohr left Denmark to visit Einstein and Fermi in the USA. Bohr had accepted an appointment for six months at Princeton. He was accompanied by his former student and colleague Leon Rosenfeld. After a nine day journey by ship, they landed in New York on January 16, 1939. That same evening, Rosenfeld broke the news about Hahn and Strassmann's discovery at a meeting of the Princeton Physics Journal Club. A week later, Frisch coined the word fission to describe the process whereby uranium atoms appeared to be split into lighter elements. (He borrowed the term from biology where it referred to bacteria separating.)

A short time later, the scientists demonstrated the possibility of a chain reaction in which each fission might initiate another fission in a different uranium atom. Based on binding energy considerations, they realized that this would release huge amounts of energy. Impressed by this possibility, and with the knowledge that the Germans were pursuing the

same research direction, Einstein wrote a letter in August, 1939 to President Roosevelt, explaining, "... the element uranium may be turned into a new and important source of energy in the immediate future....This new phenomenon would also lead to the construction of bombs, and it is conceivable - though much less certain - that extremely powerful bombs of a new type may thus be constructed." Einstein then suggested the U.S. start looking for supplies of uranium ore and that government funds be used to "speed up the experimental work, which is at present being carried on within the limits of the budgets of University laboratories." Eventually, during the summer of 1942, the Manhattan Engineer District was formed for the purpose of building an atomic bomb.

The energy release in fission is rather simply calculated from the average binding energy curve shown in Figure 11. For elements in the vicinity of uranium (way out there at A = 235) the average energy released by each proton and neutron to form a uranium atom is about 7.6 MeV. If uranium is split into roughly equal halves (fission fragments), the resulting atoms would have atomic masses, A, of about 118. In this region of the curve, nucleons must give up about 8.5 MeV of their mass-energy to form elements. Thus, in the fission fragments which result from the split uranium, each proton and neutron is "too heavy," having only given up 7.6 MeV. They solve this problem by each giving up an additional 0.9 MeV (8.5 - 7.6 = 0.9). Thus, the total energy release, PER URANIUM ATOM, is about

0.9 MeV per particle X 235 particles = 212 MeV.

Most of this energy is in the kinetic energy of motion of the fission fragments and will thus result in a large heat release when they slow down and come to rest. More information on fission will be given in the next section.

Nuclear Decay Processes

Radioactivity

Uranium ore has been mined from deposits in the Erz mountains between the Czech Republic and Germany since the mid-1800s. These mines had been active since the early 1500s due to the discovery of silver deposits in the area. The discovery of uranium in the pitchblende ore revitalized the mining industry. It was used to produce a deep orange color in glass and in the glazing applied to pottery before it was fired. (In fact, uranium has been used in this same manner up until the past few years. Bright orange dishes, e.g., Fiestaware, and pottery items made more than ten years ago will show a reading of tens of mR/hr on contact with a geiger counter.) In the year 1896, Henri Becquerel discovered that the Czech Republic ore was able to produce an exposure similar to light on photographic plates still in light-tight housings. Working in Paris with several tons of the same ore, Marie and Pierre Curie were able to separate out two different species which were even more radioactive than uranium. These were named Polonium and Radium. They discovered that the various radioactive materials emitted three different radiations. Deciding to use the letters of

the Greek alphabet in order, Ernest Rutherford named them as follows:

alpha - deflected in magnetic field, + charge
beta - deflected in magnetic field, - charge
gamma - unaffected by magnetic fields, no charge.

Each of the processes leading to the production of these basic radiation types will now be explored.

Alpha Decay

Ernest Rutherford was the first to identify and to name alpha particles. Knowing that radon gas, a daughter product of radium decay, gave off alpha particles in its decay, he sealed some radon inside an evacuated glass tube. After allowing some time for radioactive decay, he excited the resulting gas with a high voltage electrical discharge (the principle of the neon sign). Using an optical spectrometer, he observed the characteristic emission wavelengths of the element helium. This showed that the alpha particle was nothing more than the nucleus of a helium atom. The general equation for alpha decay is given in Figure 12. The P represents the parent nucleus, and the D is referred to as the daughter nucleus.

Notice that the mass numbers, A, and the atomic numbers, Z, must balance (add up to the same value) on both sides of the equation. Since these are nuclear reaction equations rather than chemical reactions, the electrons are not considered. In nature and as a result of artificial production, there are about 160 known radioac-

Alpha:

$$^{A}_{Z}P \rightarrow {}^{A-4}_{Z-2}D + {}^{4}_{2}He + \text{Decay Energy}$$

Example:

$$^{226}_{88}Ra \rightarrow {}^{222}_{86}Rn + {}^{4}_{2}He + 4.8\ \text{MeV}$$

Fig. 12 - General equation for alpha decay

tive nuclei that decay by alpha emission. The alphas are released with energies that usually fall within the range 3.5 to 10 MeV. Many common alpha emitters have between 4 and 6 MeV of energy. The observed half-lives cover a wide range, from less than a microsecond to over 10 billion years. Note that all alpha decays are a transmutation as the atomic number is decreased by two units due to the +2 alpha charge.

The energy released in alpha decay can be calculated similarly to the procedure for binding energy. If the mass of the daughter products is subtracted from the parent mass, the resulting difference is analogous to the mass decrement discussed above. The mass difference multiplied by 931.48 MeV/amu is the energy release for the reaction. In alpha decay, almost all of this appears as kinetic energy of the alpha particle.

Beta Decay

After careful measurements, Ernest Rutherford finally concluded that negative beta particles were nothing more than atomic electrons. Thus, the general equation that describes beta decay can be written as illustrated in Figure 13. Notice that there are actually two different equations which are both properly called beta decay. The first equation involves the emission of a negative electron, a process called negatron emission to distinguish the reaction from the second process. In the other equation, the particle given off is a positron.

A positron is often mistakenly thought of as a positive electron. Then, if positive electrons existed, when they encountered an ordinary negative electron, the attractive Coulomb force would cause the two particles to accelerate toward each other. They would collide and then the two equal but opposite charges would mutually neutralize. This would leave two neutral electrons. Both theory and experiment suggest that neutral electrons cannot exist in nature. Actually a positron is the antiparticle of an electron. The properties of antiparticles are the "mirror image" of the normal particle. They have the opposite sign charge of the normal particle and their spin about an internal axis is in the opposite direction. Antiparticles are as inherently stable as their normal counterparts. All elementary particles have an antiparticle. Antiprotons were discovered in 1955 and antineutrons in 1956. (The antineutron has a positive magnetic moment; see Figure 9, this chapter). It is speculated that somewhere distant in the universe stars and planets exist which are composed of antimatter. Hydrogen would consist of an antiproton encircled by a positron, etc. When antiparticles collide with their normal counterparts, both particles annihilate and their mass is converted completely to electromagnetic energy which radiates from the site. Since electrons and positrons have a rest mass energy of 0.511 MeV, two annihilation gamma rays of 0.511 MeV each are radiated away following their interaction. If a blob of antimatter (perhaps an antiradiation protection technologist?) were to encounter normal matter, the two would mutually annihilate down to the last electron and positron, with a tremendous release of energy (calculated from $E = mc^2$.).

In alpha decay, the energy available to the alpha particle, due to the mass decrement between parent and daughter products, appears as kinetic energy of motion of the alpha. If an alpha spectrometer, a detection system capable of measuring alpha

Negatron: $${}^{A}_{Z}P \rightarrow {}^{A}_{Z+1}D + {}^{0}_{-1}e + {}^{0}_{0}\bar{\nu} + DecayEnergy$$

Example: $${}^{3}_{1}H \rightarrow {}^{3}_{2}He + {}^{0}_{-1}e + {}^{0}_{0}\bar{\nu} + 18keV$$

Positron: $${}^{A}_{Z}P \rightarrow {}^{A}_{Z-1}D + {}^{0}_{-1}e + {}^{0}_{0}\nu + DecayEnergy$$

Example: $${}^{22}_{11}Na \rightarrow {}^{22}_{10}Ne + {}^{0}_{-1}e + {}^{0}_{0}\nu + 2.4MeV$$

Fig. 13 - Two examples of beta decay

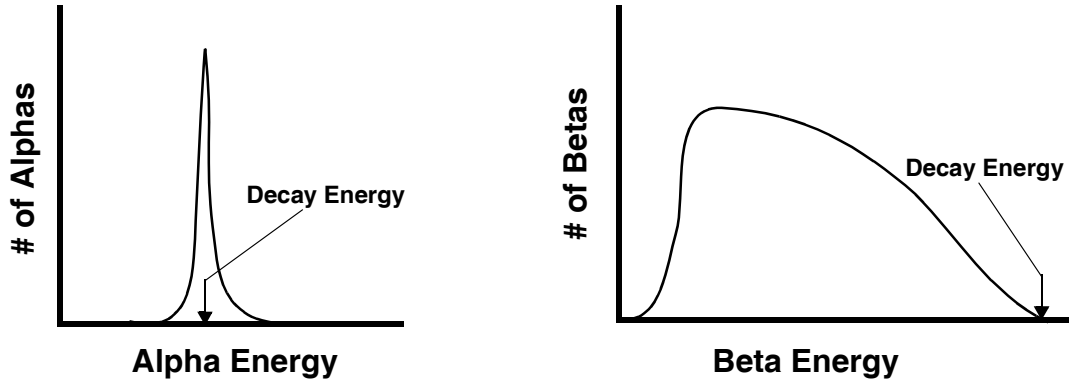


Fig. 14 - Typical alpha and beta energy spectra

energies, is placed near a monoenergetic alpha source, the energy spectrum shown in Figure 14 results. Shown for comparison is a typical beta source energy spectrum.

Early physicists were greatly surprised to find that virtually none of the beta particles emitted from a beta source carried a kinetic energy equal to the decay energy. Additionally, the individual beta particles being emitted did not even all have the same energy as alpha particles do. The betas have a whole range of energies up to a maximum which is the energy released in the decay process. In beta decay, we often call this energy the endpoint energy or $E_{maximum}$. On the average, betas carry <u>about 1/3 of the endpoint energy</u>. The big question of the day – where did the rest of the energy go??

It should be noted that the 1/3 rule just stated applies to <u>negatrons</u> only. Since a positron carries a positive charge, by Coulomb's Law, it will be repelled when ejected from the nucleus. This gives a positron an extra "kick" in energy. Furthermore, since the charge on the nucleus (Z number) is one of the factors in Coulomb's Law, the higher the Z number of the positron emitter, the greater the "kick" in energy. Thus, the <u>average</u> positron carries between 35% and 63% of the endpoint energy, dependent on Z number and the endpoint energy.

Numerous attempts to find the missing energy in the form of some radiation coming from the sources proved fruitless. A number of physicists felt that beta decay actually violated the universal energy conservation rule. In physics, energy may be transformed from one kind to another, but it cannot be created or destroyed - total energy is always constant. This difficult problem was finally resolved by Wolfgang Pauli who postulated, in 1931, the existence of neutrinos, a "massless uncharged particle" which carried energy and momentum. The name, first proposed by Enrico Fermi, translates from Italian as "little neutral one," and is eminently suitable. The neutrino was felt to be virtually non-interacting which explained why it wasn't detected.

For decades, scientists were convinced that the neutrino mass was zero. Then, in the summer of 1998, the huge underground neutrino detector Super Kamiokande in Japan provided strong experimental evidence that the three neutrinos in nature actually have measurable masses. The table in Figure 4 of Chapter 1 lists the best estimates that we currently

have for these masses. The detector contained over 50,000 tons of ultra-pure water and 13,000 photomultiplier tubes buried one-half mile deep under a mountain. It was operated by a team of 120 physicists.

Following Pauli's announcement, intensive work began at numerous research centers around the world in an attempt to be the first to detect a neutrino. The neutrino ultimately proved to be so elusive that it remained undetected for another 22 years. The first experimental detection of neutrinos was accomplished by Reines and Cowan of Los Alamos Scientific Laboratory, after 4 years' work, in 1953. They made use of a huge liquid scintillation detector of about 10 cubic feet in volume viewed by 90 photomultiplier tubes. The apparatus was set up next to a large operating nuclear power reactor at DOE's Savannah River Plant in South Carolina, The beta activity from fission products in the fuel elements produced a neutrino flux of about 5×10^{13} per square cm per second. They eventually detected a net count rate due to neutrinos of 2.9 counts per hour. Figure 15 shows their apparatus. About 4 decades after their pioneering experiment, Reines was awarded the 1995 Nobel Prize in physics for discovery of the neutrino.

In the preceding paragraph, the neutrino was said to be "virtually non-interacting." Clearly, if neutrinos never interacted with matter, Reines and Cowan would not have been able to detect them. It turns out

Los Alamos National Laboratory

Fig. 15 - Reines and Cowan's neutrino detector

that neutrinos are affected by the weak force. Physicists have identified four fundamental forces in nature: gravitational, electromagnetic, weak and nuclear. (Strictly speaking, there are now only three left since Weinberg, Glashow and Salam won the 1985 Nobel Prize for combining together the electromagnetic and weak forces into a single force now called the electroweak force.) Neutrinos interact via the weak force, the same force which controls the radioactive decay of many nuclei and fundamental particles. The reaction which led to the experimental discovery of the first neutrinos was just the "inverse" of the beta decay of neutrons, specifically, $\nu + p \rightarrow n + e^+$.

Every now and then, one of the zillions of neutrinos passing through the scintillation fluid in Reines' and Cowan's detector would be absorbed by a proton. As indicated, this would then release a neutron and a positron into the fluid. The neutron presence was detected by "spiking" the scintillation fluid with a cadmium compound. Cadmium emits several gamma rays of unique energies when it captures a neutron. The positron was detected from the two annihilation gamma rays at 511 keV each. Both events occurring in the correct time sequence signaled the detection of a neutrino.

It is now recognized that a neutrino or antineutrino is released in every beta decay. In the beta decay equations of Figure 13, the neutrino is written as a lower case Greek nu (ν) and the nu with the bar above it ($\bar{\nu}$) represents an antineutrino. Based on careful measurements, it is possible to calculate the average distance of travel of a neutrino through matter, the so-called mean free path. A rather commonplace 1 MeV neutrino would travel, through solid lead, an average of 50 light years (3×10^{14} miles) before interacting. (Considering the price of a lead brick today, this experiment isn't likely to ever be done). Neutrinos, of course, present no health physics problem. Since they are virtually non-interacting, very little energy is deposited and, therefore, almost no dose results to personnel exposed to neutrinos.

Actually, it is possible to calculate personnel doses from neutrinos. Even though the interaction probability is small, it has been measured. For neutrinos below 1 GeV of energy, the neutrino dose to tissue is due to the deposited recoil kinetic energy of atomic electrons that have elastically scattered in a neutrino collision. Our sun bombards us continuously with a neutrino flux of about 5 million per square cm per second. Over the course of one year, the resultant tissue dose is about 10^{-5} microrem.

Most radioactive nuclides with $Z < 82$ decay by one of the two beta decay processes. Remember that the betas emitted have a range of energies up to the value of the decay energy, $E_{maximum}$. On the average, they carry 1/3 E_{max} and the neutrinos 2/3 E_{max}. Beta emitters found in nature or artificially produced have E_{max} energies ranging from 0.018 to about 3.6 MeV. (See Sample Problem 3).

The name "beta particle" was proposed by Ernest Rutherford "for convenience" while working at McGill University in Canada around 1900.

Gamma Ray Emission

The third type of radioactive decay discovered by the early radiation was decay by gamma ray emission. Gamma rays were soon found to be another electromagnetic

Sample Problem 3

GIVEN:

Fictional nucleus Gollnickium-280 has an atomic number of 130.

FIND:

The daughter nuclide if it 1) decays by negatron emission and 2) decays by positron emission.

SOLUTION:

From figure 13, negatron decay raises the Z by 1 unit, leaving the A unchanged so the new daughter would be $^{280}_{131}D$.

Similarly, positron decay lowers the Z number by 1 unit and doesn't change A. Thus, the new daughter of this positron emitter would be $^{280}_{129}D$.

radiation like light or x-rays. In fact, gamma rays originate in atoms in a way very similar to x-rays. X-rays result from atomic electrons falling from a higher energy shell to a lower energy shell in the atom. Gamma rays result from transitions of the nucleus from a higher to a lower energy state. Again, the energy difference is a fixed value for a given isotope of a nuclide and so the gamma rays all have the same characteristic energy which can be used to signal the presence of that particular radionuclide in a sample.

Since the gamma decay doesn't involve the gain or loss of protons or neutrons, the general equation is slightly different from the equations for alpha and beta decay (see Figure 16).

Gamma: $^{A}_{Z}P^{*} \rightarrow {}^{A}_{Z}P + \gamma +$ **Decay Energy**

Example: $^{137}_{56}Ba^{*} \rightarrow {}^{137}_{56}Ba + \gamma +$ **0.662 MeV**

Fig. 16 - General equation for gamma ray emission

The asterisk ("star") to the upper right of the parent symbol, P, indicates that the parent is in an excited energy state. Gamma emission is thus merely the de-excitation of a nucleus. The "parent" and "daughter" nuclides are the same. The emission does not represent a transmutation.

Although the nucleus usually emits the gamma ray within a microsecond or less of becoming excited, there are some nuclides where the gamma release takes much longer. These nuclides are referred to as metastable. To indicate that a particular nuclide is metastable, a lower case m is attached to the mass number, e.g., Tc-99m. This particular radionuclide has great usefulness in diagnostic nuclear medicine procedures. It reduces the radiation dose to the patient as the gamma ray is the only radiation emitted. With most gamma emitters, the gamma ray is emitted along with

an accompanying alpha or beta particle. These particles then produce a radiation dose to the patient without providing any useful diagnostic information since all of their energy is absorbed inside the patient. The half-life of ^{99m}Tc is 6 hours.

Gamma rays emitted from radionuclides cover a wider range of energies than either beta rays or alpha particles. The lowest known gamma ray energy is 0.008 MeV (from Er-169) and the highest is 7.11 MeV (from N-16). The most common gamma ray energies in the occupational environment range from about 0.15 to 1.5 MeV.

Fission Decay

In discussing nuclear binding energy the energy release in fission was calculated. Actually, the first observed release of fission energy from a sustained nuclear chain reaction occurred at 3:36 PM on December 2, 1942. Enrico Fermi was in charge of 42 other scientists who constructed an "atomic pile" or nuclear reactor beneath the University of Chicago athletic stadium. This first reactor consisted of a stack of graphite moderating blocks and uranium fuel and was designated CP-1 for Chicago Pile #1. One of the few photographs of the CP-1 reactor that is known to exist is shown in Figure 17. It was taken during addition of the 19th layer of graphite.

The main safety feature was a control rod tied with a rope to a balcony. A scientist, Norman Hilberry, stood nearby with an axe to cut the rope in case of excess power. He, in fact, is credited with coining the word "SCRAM" to refer to rapid shutdown of a reactor. Dr. Hilberry referred to his job as that of the Safety Control Rod Axe Man! Other safety features included three scientists positioned above the pile with pails of cadmium solution (a strong neutron absorber) to pour on the pile in case of mishap. The pile was critical for 28 minutes that afternoon. Figure 18 is an artist's re-creation of the momentous occasion. None of the scientists had thought to bring a camera! A bottle of Chianti wine did, however, appear!

There actually are several different types of fission decay. Thermal fission is the type which powered the Chicago pile. It involves the absorption of a thermal (low energy, slow speed) neutron by substances such as U-235 or Pu-239. Such materials are said to be fissile. When the mass-energy of the slow neutron is added to the fissile nuclide, the resultant nuclide is unstable and decays by separating into two pieces, called fission fragments, which fly away from each other due to the strong Coulomb repulsion. In the case of U-235, about 208 MeV of energy is released in the process. Of the total, 83% goes to kinetic energy of the fission fragments, 6% is immediately released as gamma rays and the remaining 11% is eventually released in the various radioactive decays of the fission fragments. These products of the fission have a rather asymmetric (unequal) mass distribution. The two fission fragments or products are most often formed in the ratio of 140:95. The separation of the uranium atom into exactly equal pieces occurs 1000 times less frequently. Over 200 separate nuclides have been identified in the products of uranium fission. Figure 19 shows the distribution of fission products from U-235 fission.

A second type of fission is called fast fission because it is caused by the capture of a high energy or fast neutron. One isotope which undergoes fast fission is ^{238}U.

Courtesy Argonne National Laboratory

Fig. 17 - Photo of the CP-1 "atomic pile" taken during the construction phase

Such nuclides are referred to as fissionable. Typically, the chances of a fissionable nuclide capturing a fast neutron to cause fission is several hundred times smaller than fissile nuclei capturing a thermal neutron.

The last type of fission is spontaneous fission. In this process, a nuclide is able to decay by fissioning without capturing a neutron. It has a characteristic half-life for these decays. Spontaneous fission was first discovered around 1940. Since then,

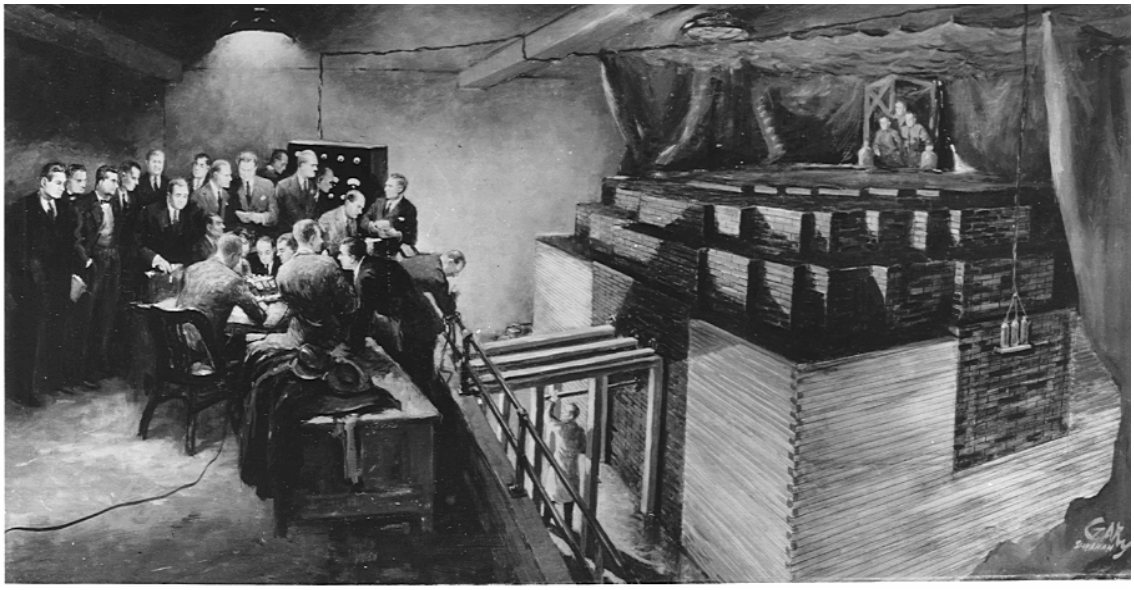


Fig. 18 - Initial criticality of world's first reactor

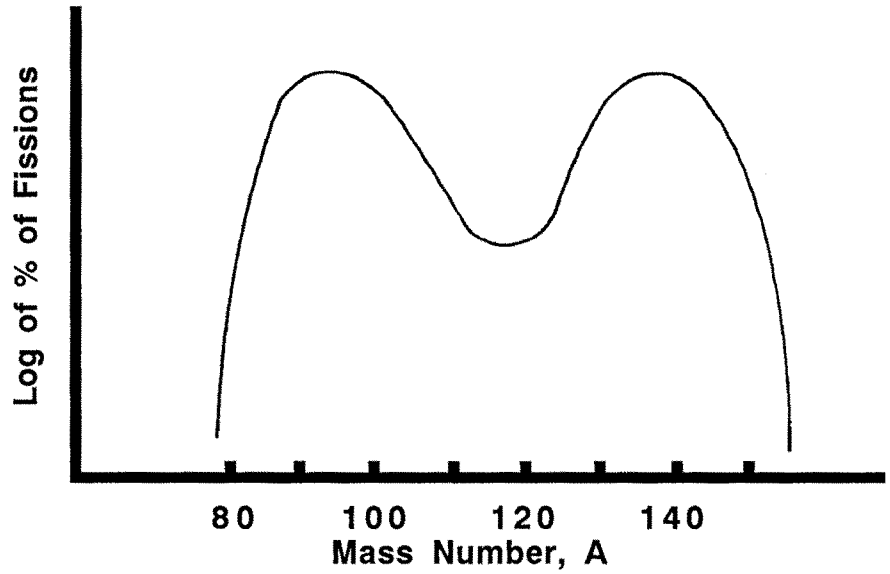


Fig. 19 - Distribution of U-235 fission products by mass number

some 30 spontaneous emitters have been found. Generally the fission decay competes with alpha decay as a second possibility for the nucleus. Usually the half-lives are very short. One popular exception is californium-252. This nuclide has an effective half-life of 2.6 years. Small sealed Cf-252 sources will simulate the neutron output of a reactor, since reactor neutrons originate by fission. They are frequently used in the radiation protection technology field to calibrate neutron survey meters and personnel radiation dosimeters. The sources have had some use in medicine as well, being used

to treat solid tumors in the body through implantation of needles containing the Cf-252 radioisotope.

Other Decay Processes

The major radioactive decay processes have been covered previously. Now, a couple of minor contributors to radioactive decay will be covered.

Electron Capture, often abbreviated EC, is a decay mode available to neutron deficient nuclides. As discussed in the Nuclear Stability section earlier in this Chapter, nuclei generally prefer to disintegrate in such a way that the resulting daughter nuclide is closer to the line of stability. Therefore, nuclides deficient in neutrons like to convert one of their protons into a neutron to become "less deficient." This conversion can be accomplished in two ways. Positron emission and EC are in competition. The "winner" is determined by the specific mass-energy values of the parent and daughter nuclides.

The actual process of electron capture merely involves the nucleus grabbing one of the orbiting atomic electrons. Although our previously developed model of the atom treated these electrons as revolving in orbits with precisely determined radii, the application of quantum mechanics to atomic electrons showed that the electrons actually tend to "stray from the path." More evolved atomic models treat the electron as spread out over a probability cloud of various defined shapes. The density of the cloud at any particular point is the probability of finding the electron at that precise point. The shape of the probability cloud for the innermost K shell electrons overlaps the atomic nucleus which tells us that part of the time, the K shell electrons are inside the nucleus! Thus, the chances are high that during electron capture, it is indeed a K-electron that disappears, in comparison to a L-electron, or M-electron, etc.

The defining equation for EC is given in Figure 20. Note that the daughter

Electron Capture: $${}^{0}_{-1}e + {}^{A}_{Z}P \rightarrow {}^{A}_{Z-1}D + {}^{0}_{0}\nu + DecayEnergy$$

Example: $${}^{0}_{-1}e + {}^{51}_{24}Cr \rightarrow {}^{51}_{23}V + {}^{0}_{0}\nu + DecayEnergy$$

Fig. 20 - General equation for electron capture

nuclide is the same one that would have resulted if the parent had emitted a positron. However, in contrast to positron decay, only two particles are released in EC. Therefore, the neutrinos in EC decay have fixed energies. They are not required to share the energy with positrons of various energy. Also, in contrast, the disappearance of an orbital electron in EC means that a higher shell orbital electron will drop down to fill the vacancy. In falling from a higher to a lower energy level, the excess energy (energy difference) is released. It is termed a characteristic x-ray because the energy is characterized by the specific chemical element of the daughter nuclide. [Characteristic x-rays are discussed further in Chapter 6.]

The final decay process to be considered here is internal conversion. Just as

positron decay and electron capture competed with each other, now internal conversion and gamma decay compete. Again, the daughter nuclide is the same in both cases. The process involves a nucleus in an excited state. Excited nuclei are rather rare. They much prefer to live in an unexcited (or "ground") state and, in most cases, rid themselves of the extra energy almost instantaneously. The abnormal circumstances that lead to their excitation usually involves a preceding nuclear decay process. In the example used previously for gamma decay, the excited ^{137}Ba* was produced when a Cs-137 nucleus underwent beta decay.

The actual process is pretty straight forward. Instead of the excited nucleus using its extra energy to produce and release a gamma ray, it transfers the decay energy to an orbital electron, usually a K-electron. Since this electron is bound to the nucleus by the Coulomb force, it "uses up" an amount of energy called the electron binding energy to escape the atom. Thus, all of these conversion electrons, from a given excited nuclide, are emitted monoenergetically (with the same energy). The emission energy is calculated simply as the decay energy minus the electron binding energy.

Sometimes internal conversion electrons from a given radionuclide are measured to have two different energies. This is a result of some L-electrons being "converted" along with the more common K-electrons.

The defining equation for internal conversion is shown in Figure 21. Note again

Internal Conversion: $^{A}_{Z}P^{*} \rightarrow {}^{A}_{Z}D + {}^{0}_{-1}e + DecayEnergy$

Example: $^{198}_{80}Hg^{*} \rightarrow {}^{198}_{80}Hg + {}^{0}_{-1}e + 0.412MeV$

Fig. 21 - General equation for internal conversion decay

that the conversion electron doesn't have to share the decay energy with a neutrino. So while the electron released in internal conversion is physically identical to a beta particle released in beta decay, they are given different names because the conversion electron comes from the atomic electron orbits and the beta particle comes from the atomic nucleus.

Radioactive sources of internal conversion electrons are more than just a scientific curiosity. They have a practical radiation protection use. Special health physics instruments capable of measuring beta energies need to be calibrated. Ordinary beta sources emit a full spectrum of energies and so are unsuitable. The monoenergetic conversion electrons are perfect for this application. Bi-207 is particularly sought due to the 38 year half-life.

Radioactive Decay Law

The rate at which a radioactive substance decays, i.e., the activity (to be defined in Chapter 5), appears to be an unalterable constant for each radionuclide. This decay rate and the associated half-life are unchanged by huge gravitational

forces, freezing to temperatures near absolute zero or placement in strong magnetic fields. Furthermore, the mathematical equation describing the decay is identical for alpha decay, negatron decay, positron decay, gamma decay and spontaneous fission decay. This equation is called the universal radioactive decay law.

In order to make sense of the decay law equation, consider an analogy with the life insurance business. Assume at some starting time there are N policyholders. To establish the premium rates, it is necessary to calculate the expenses over some period such as a year, i.e., the number of deaths per year of the clients. After a little thought it becomes clear that the number of deaths of the policyholders (assuming they occur randomly in time) depends only on two factors. The number of deaths is proportional to the time period of observation. On the average, twice as many death claims would be expected in a two year period compared to a one-year time span. Secondly, the number of deaths would be proportional to the overall number of policyholders. Large companies would have more claims than small companies. We can express these facts mathematically as a proportionality. To convert a proportionality to a math equation, a constant of proportionality, lambda (λ), is inserted. Finally, since the number of policyholders is decreasing (because they are dying off), a minus sign is put in to indicate the direction of the change in the number of policyholders. The result is shown below as Equation 1. The Δ is the Greek letter delta, commonly used to indicate a change in a variable.

$$\Delta N = - \lambda N \Delta t \qquad \textbf{[Eqn. 1]}$$

The radioactive decay equation is identical. In this case, the N would now represent the number of radioactive atoms at some starting time. Then ΔN is the change in N, i.e., the number of decays during a time interval Δt. The disintegration rate can easily be found by dividing both sides of Equation 1 by Δt:

$$\Delta N / \Delta t = - \lambda N \qquad \textbf{[Eqn. 2]}$$

The left-hand side of Equation 2 is now a radiation quantity called Activity, with units of disintegrations per time. Making this substitution, Equation 2 becomes:

$$\text{Activity} = A = \lambda N \qquad \textbf{[Eqn. 3]}$$

This equation can be used to give the <u>magnitude</u> of the disintegration rate of a sample containing a known number of atoms. (The minus sign has been dropped as it only referred to the fact that the number of atoms in the sample was <u>decreasing</u> with the passage of time.) If the time were expressed in seconds, then, the activity would be in dps, i.e., disintegrations per second or becquerels (as we will see in Chapter 5).

If Equation 2 is subjected to a mathematical operation from the calculus called integration, one form of the radioactive decay law results:

$$N_t = N_0 \, e^{- \lambda t}. \qquad \textbf{[Eqn. 4]}$$

If the operation of integration is unfamiliar, this should not cause concern. Usually a knowledge of calculus is not required in Radiation Protection Technology. This is the only place it is used in this textbook. However, the resulting equation should be learned.

The half-life, T, is the time required for half of the sample to decay. In the next half-life, half of the REMAINDER decays leaving 1/2 X 1/2 = 1/4 of the original. Only after an infinite time span will all of the sample have decayed. Equation 4 is usually written in a more useful form. Generally, tables of values for the decay constant,

Sample Problem 4

GIVEN:

A wipe sample shows an activity of 6,500 Bq. A week later, this same sample has an activity of 4,200 Bq.

FIND:

The half-life, in days, of the activity on the sample.

SOLUTION:

Use Equation 7. A_0 = 6500 and A_t = 4200. Divide both sides of the equation by A_0. Then, A_t/A_0 = 4200/6500 = 0.646 = $e^{-0.693\ t/T}$. Next, take the natural logarithm of both sides. This gives ln 0.646 = -0.437 = -0.693 t/T.

But t = 1 week = 7 days. So, T = -0.693 x 7 days / -0.437 = 1.59 x 7 days.

Or, finally, T = 11.1 days.

lambda, are not available. But, λ is related to the half-life T, by the following relationship:

$$T = \ln 2/\lambda = 0.693/\lambda. \qquad \textbf{[Eqn. 5]}$$

If this substitution is made, the radioactive decay law can be written in the common form as shown in Equation 6:

$$N_t = N_0\ e^{-0.693\ t\ /\ T} \qquad \textbf{[Eqn. 6]}$$

where: N_o = starting # of atoms at time t = 0

N_t = number of atoms after time t

T = sample half-life in the same time units as t.

Substituting Equation 3 into Equation 6, the most useful form for technicians results.

$$A_t = A_0\ e^{-0.693\ t/T} \qquad \textbf{[Eqn.7]}$$

Sample Problem 4 shows a decay equation calculation. The number of atoms in a sample or the sample radioactivity decreases exponentially with the passage of time.

Note: Developer Ray McGinnis has generously provided the freeware program, ***Rad Pro Calculator,*** to the radiation protection community. It is designed to make activity calculations such as Equation 7 easy to carry out. Visit www.radprocalculator.com for the latest version. Ray has adapted the software to run on several platforms.

The decay law was discovered experimentally by Ernest Rutherford in 1900. He measured the decay of "thoron gas" from a thorium source and found the radioactivity decreased by half each minute.

It should also be noted that the decay law applies to the sample as a whole. It does not, and cannot, tell precisely when any given single nucleus will decay. It gives the probability of decay of the nuclei.

Equation 3 has another use. In the earlier discussion of nuclear stability, reference was made to some measured half-lives for radionuclides that greatly exceed the age of the universe. How can these be measured in the laboratory within a time frame short enough that the technicians aren't forced into retirement from the job? The radioactive material is

weighed (to enable the simple calculation of N in equation 3). Then, this sample with the known number of atoms is placed in a counter and the disintegration rate measured (the A in equation 3). Dividing A by N leaves the decay constant which can easily be converted to the required half-life from equation 5.

Equation 3 for the activity can be used for one more useful concept. Since for any activity, $A = \lambda N$, then if $N = N_0$, the starting # of atoms in the sample, it follows that $A_0 = \lambda N_0$ would be the initial disintegration rate at t = 0. If, for the sake of argument, the activity did not decrease exponentially with time, but instead remained constant, then in time t there would have occurred A_0 x t disintegrations (# of disintegrations = rate x time). The time that it would take to "use up" the entire sample of N_0 atoms can thus be obtained by setting it equal to A_0 t:

$$A_0 t = N_0 \text{ or } t = N_0 / A_0.$$

Substituting from the activity equation, $N_0 / A_0 = 1 / \lambda = t$. This special time is called the "average lifetime" of the sample.

To conclude this section, we need to consider the case of radioactive daughters. If a radioactive parent decays to a daughter that is also radioactive, we have what is referred to as a radioactive decay chain. The extreme cases are the three natural radioactive decay chains. The starting parent nuclides are Th-232, U-238 and U-235. The half-lives of all three are over 700 million years. These chains are called "natural" since each of the parents is found naturally in the earth's crust. Each chain has many generations of radioactive daughters. The U-238 chain has the most - 14 generations before it terminates in a stable daughter nuclide. Interestingly, all three chains end with some stable isotope of lead. A number of well known, useful or infamous radioactive materials are members of the natural decay chains. In addition to the famous starting parent nuclides, examples include Ra-226, Rn-222, Po-210, and Ac-227.

A quick aside. Early radiochemists working to figure out all the members of these complex chains realized that there ought to be a fourth one. They called it "the missing series." Subsequent hard work eventually led to the discovery of the missing radionuclides. The starting parent is Np-237, a 2 million year half-life artificially produced parent. It results from the decay of americium-241, a commercially important nuclide. (Think smoke detectors!) All Np chain half-lives are short enough so that even if neptunium was present in the earth's crust at the time the earth formed (about 4.5 billion years ago), it would have long since decayed.

A graphical representation of two of the natural chains is given in Figure 22. Alpha decays are represented by a downward vertical move of two steps and negative beta decays by an upward 45° to the right move of one step.

The mathematics of the chain decay are reasonably complicated. Details can be found in advanced health physics or modern physics texts. However, the results of the theory are pertinent to radiation protection technology. The fundamental general equation that covers all the possible cases of parent/daughter half-lives gives the number of daughter atoms, N_D, at any time, t. It is expressed in terms of the original number of parent atoms at the start, t = 0, i.e., $N_{P,0}$ and in terms of the respective radioactive decay constants λ_P and λ_D:

$$N_D = N_{P,0}\frac{\lambda_P}{\lambda_D - \lambda_P}(e^{-\lambda_P t} - e^{-\lambda_D t}). \qquad \textbf{[Eqn.8]}$$

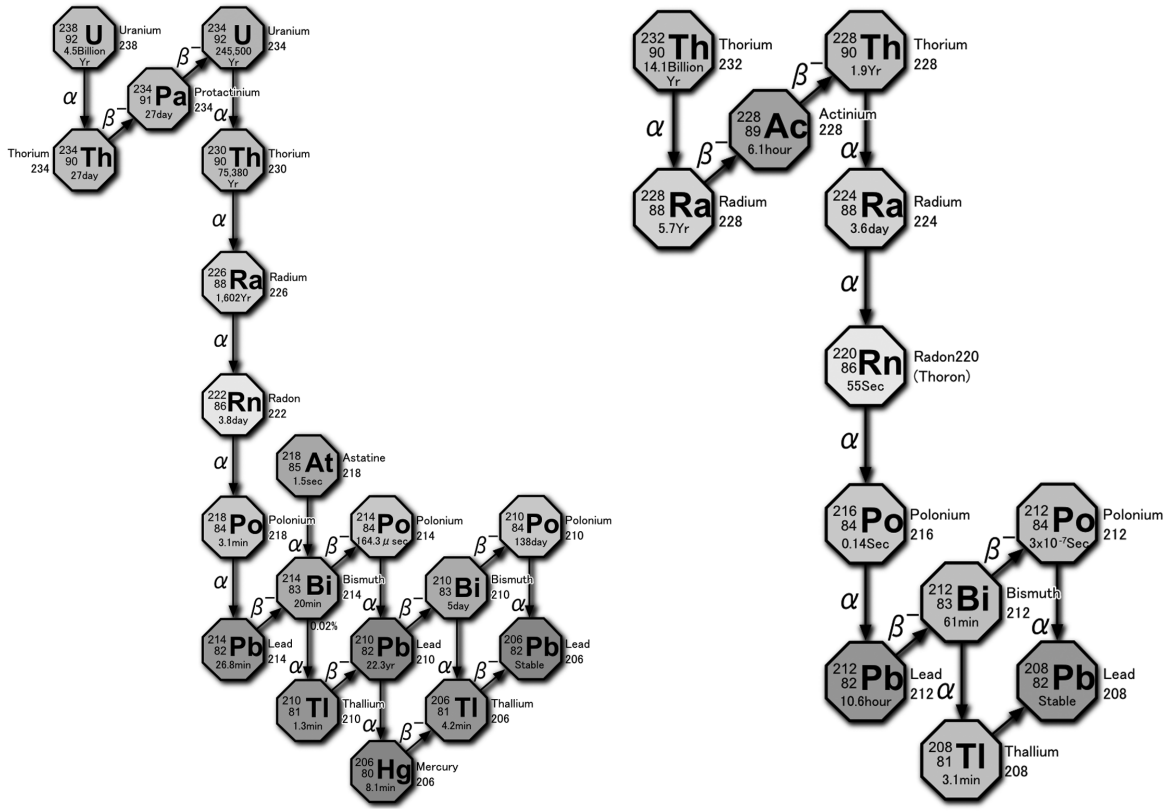


Fig. 22 - The uranium-238 and thorium 232 natural radioactive decay chains

This general form has three possible solutions as follows:

CASE 1 - Parent half-life huge, daughter half-life small

Under these conditions, the parent acts like a constant activity source, i.e., no loss through decay. The equation becomes

$$N_D = N_{P,0}\frac{\lambda_P}{\lambda_D}(1 - e^{-\lambda_D t}). \qquad \textbf{[Eqn. 9]}$$

The number of daughter atoms begins at t = 0 and grows exponentially. After about 7 daughter half-lives, the exponential term falls to zero and Equation 9 reduces to $N_D \lambda_D = N_P \lambda_P$. But Equation 3 tells us that the product of N and the decay constant l is an activity; the parent and daughter have the same activity in, e.g., microcuries. This condition of equal activities is called secular equilibrium.

CASE 2 - Parent half-life somewhat longer than daughter half-life

Here, the parent is decaying slower than the daughter, but the half-life difference isn't great. The daughter starts out at zero and climbs as parent atoms transform into daughter atoms. But the supply of parent atoms is not inexhaustible as it was in Case 1. Eventually, the production of daughters will be limited by the reduced number of parents available and so the daughter activity will decrease over time with the half-life of the parent. This condition of the parent and daughter both decaying with the same half-life is termed transient equilibrium.

Case 3 - Parent half-life shorter than the daughter half-life

Now we have a situation where no form of equilibrium is possible. Initially the parent is rapidly decaying but the daughter activity is growing rapidly. Soon, only the radioactive daughters remain. After a long elapsed time, equation 9 reduces to

$$N_D = N_{P,0}\frac{\lambda_P}{\lambda_P - \lambda_D}e^{-\lambda_D t}. \qquad \textbf{[Eqn. 10]}$$

This show us that the daughter activity will decay with the half-life of the daughter. A graphical representation of the first two cases is given in Figure 23. Case 3 graphically looks like the transient case except the daughter half-life is followed.

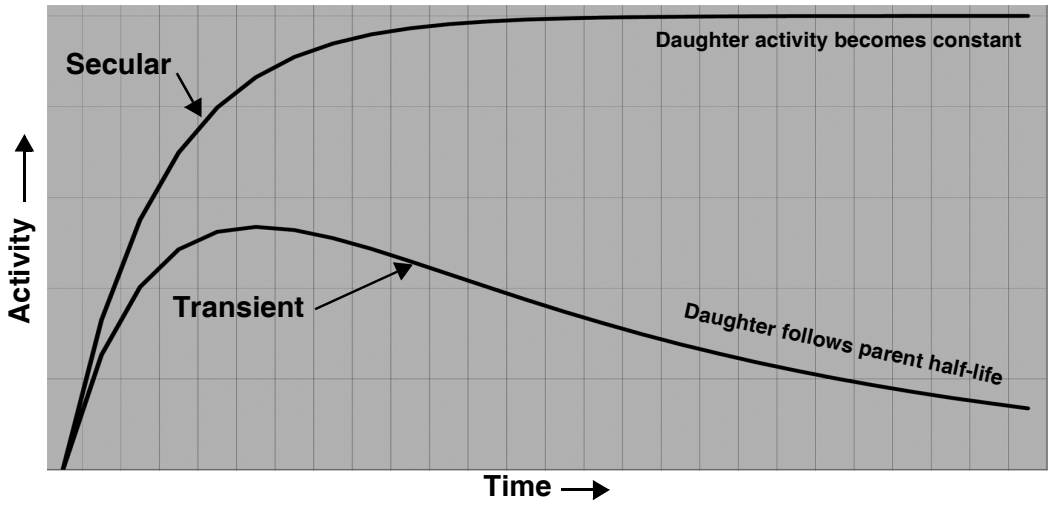


Fig. 23 - Graphical examples of secular and transient equilibrium for radioactive decay chains

Problem Set

1. Make a sketch of an atom. Label the parts. Indicate the size scale.

2. About how many times smaller than the electron outer orbit diameter is the nuclear diameter?

3. Name two reasons why gold is a good choice of material for observing Rutherford Scattering.

4. Define the terms “isotopes,” “fission,” and “binding energy.”

5. Describe the change in atomic number and mass number of a parent nucleus under: a) alpha decay, b) negatron decay, c) positron decay. Which of the above are transmutations?

6. Why do heavy nuclei near the end of the periodic table have a neutron to proton ratio of about 1.5?

7. Show how the average binding energy curve in Figure 11 predicts that energy will be released in a fusion reaction.

8. Name two ways in which the Coulomb and nuclear forces differ.

9. Calculate the average energy of the negatron and neutrino emitted in the beta decay of ^{90}Y.

10. What isotope results from the alpha decay of ^{210}Po?

11. For convenience we often distinguish gamma and x-rays by their different formation mechanisms. Describe each mechanism.

12. Neutrinos emitted in electron capture decay are monoenergetic, while neutrinos emitted by the competing decay mode, positron emission, are emitted with a wide spectrum of energies. Why would measurement of the neutrino spectrum NOT be a great way for an RP technologist to determine how much EC decay is occurring in a wipe test sample?

13. Mercury K-electrons are bound with an energy of 83 keV. What is the energy of the converted K-electrons in the decay of Hg*-198 ?

14. A radioactive sample decays to half the original number of atoms in 1 day. What will the half-life be 2 days later?

15. If a radioactive sample has 4×10^6 radionuclei at a point in time, and there are 50,000 decays observed over a 15 minute period, calculate the radioactive decay constant lambda for this nuclide.

16. A radioactive sample is measured to have a decay rate of 2000 dps (disintegrations per second), and a decay constant of 0.02/min. Calculate the decay rate that this sample would have had one day earlier.

17. What % of the original activity is left in a sample that has decayed for 8 half-lives?

18. A hospital needs 1 millicurie of a radioactive pharmaceutical with a half-life of 12 hours. If delivery time takes 3 days, how much activity must be shipped to the hospital by the supplier? How many microcuries will remain a week after arrival of the 1 mCi?

19. Which of the following common isotopes has the shortest half-life? Which has the longest? Ra-226, Co-60, Cs-137, I-131, U-238

S-1. Technicians working at reprocessing plants which recycle spent fuel from nuclear reactors seldom encounter positron emitters in the work environment. What is the reason for this?

S-2. Calculate the number of half-lives in the average lifetime of a sample.

S-3. Calculate the total binding energy and average binding energy per nucleon of the ^{12}C atom, given that a proton has a mass of 1.007277 amu, a neutron is 1.008665 amu, an electron is 0.0005486 amu and ^{12}C is 12.00000 amu.

S-4. Calculate the energy of the alpha particle emitted in the decay of ^{226}Ra. The nuclei involved have the following masses given in amu: ^{226}Ra = 226.0249495, ^{222}Rn = 222.01712 and ^{4}He = 4.002603.

S-5. Which radionuclides in the following list are likely to be positron emitters? Why? 56-Ba-124, 56-Ba-144, 29-Cu-62, 80-Hg-206.

S-6. Which of the following two positron emitters would be expected to have an average energy which is the largest fraction of its endpoint energy, 79-Au-184 or 55-Cs-123? Why?

S-7. Five grams of the fictitious radioisotope Go-152 are measured to have a net disintegration rate of 59 disintegrations per week. What is the half-life, in years, for this nuclide?

Other Resources

1. “Nuclear Radiation Physics” by Lapp and Andrews, Prentice-Hall, Inc., New Jersey, 1972. (Out of print, but very readable if you can find a copy in a library. Used copies are frequently offered on www.Amazon.com.)

2. “Elementary Modern Physics” by Richard Weidner and Robert Sells, Alternate Second Edition, Allyn and Bacon, Inc., Boston, 1976. (Out of print. Used copies are frequently offered on www.Amazon.com.)

“The First Reactor” from “Understanding the Atom Series,” Division of Technical Information, U.S. Atomic Energy Commission, U.S. Dept. of Energy, Library of Congress Catalog Card Number 50-60514. (Available from The Library of Congress, Photoduplication Services, Washington, DC 20540).

3. “Modern Physics” by Kenneth S. Krane, John Wiley & Sons., Inc., New York, 1995.

4. “Radiological Physicists,” Juan del Regato, American Institute of Physics, New York, 1985. (Contains detailed biographies of many historically important physicists mentioned in this text such as Roentgen, Curie, Rutherford, Compton and Fermi.)

3

Interactions of Radiation with Matter

Outline of this Chapter

Chapter Summary

Ionizing radiations (particles or rays that can cause the ejection of orbital electrons from absorber atoms) can be divided into two classes. Indirectly ionizing radiations must first transfer energy to some charged particle in an absorber. Then, the charged particle transfers some or all of this energy to the absorber. Neutrons and photons are the two commonly encountered indirectly ionizing radiations. Directly ionizing radiations carry a charge and so can deposit energy directly into an absorber. Alphas and betas are two examples.

For radiation protection purposes, neutrons are classified as either slow, meaning relatively low energy, or fast, meaning high energy. Slow neutrons often undergo radiative capture in which the neutron is caught by an absorber atom which then releases energy as a gamma ray. Sometimes the slow neutron capture leads instead to emission of a charged particle, an alpha for example. Finally, fission is an important neutron interaction utilized in nuclear reactors to produce energy.

Fast neutrons typically undergo scattering interactions where energy is transferred through collisions. These collisions can either be elastic, in which kinetic energy is conserved, or inelastic, in which some energy appears as a gamma ray.

Photons are the other main indirectly ionizing radiation. Gamma rays and x-rays usually interact in one of three processes. The atomic number of the absorber and photon energy determine which of the three is most likely. In the photoelectric effect, the photon is totally absorbed by an orbiting electron which is then ejected from the atom. In Compton Scattering, only part of the photon energy is transferred to an electron. The balance is given to a lower energy photon released from the interaction site. In pair production, if a photon carries more than 1.022 MeV of energy, it can disappear and reform as an electron - positron pair of particles. The positron usually annihilates to release two 511 keV gamma rays. Photons interact only by following the laws of chance. Thus, they do not have a range in an absorber. Instead the mean free path is used to estimate the average distance of travel before an interaction.

Charged particles are directly ionizing. They generally deposit energy in an absorber through ionization (ejection of orbiting electrons), excitation (raising of orbital electrons to higher energy levels within the atom) or bremsstrahlung (photons released as the charged particle experiences the Coulomb force from the nucleus). They give up energy continuously along the path they follow through an absorber. If the rate of energy transfer is known, i.e., the stopping power, then, their range can be computed. Charged particle range depends on the charge of the particle, its energy, and the Z number and density of the absorber

Indirectly Ionizing - Neutron Interactions

Introduction

Ionizing radiations are all capable of removing electrons from absorber atoms. [This text does not cover non-ionizing radiations such as laser light, ultraviolet, ultrasound, Cerenkov radiation and microwaves.] The ionizing radiations are further subdivided into two categories - indirectly ionizing radiation and directly ionizing radiation. Indirectly ionizing radiations include neutrons and gamma rays. They first transfer energy to charged particles, e.g., protons or electrons in the absorber. These secondary charged particles then produce ionization directly via the Coulomb force interaction. Directly ionizing radiations include charged particles such as alpha and beta rays. They eject orbital electrons from atoms directly via the Coulomb force.

Ernest Rutherford was a graduate student of J. J. Thomson when Roentgen announced the discovery of x-rays in January of 1896. Thomson soon had built an x-ray tube and, together with Rutherford, began a series of experiments. They found that x-rays were able to produce positive and negative charges in air. In November, 1896, Thomson and Rutherford published a paper in which they originated the term "ionizing radiation" to describe this property of x-rays.

Probably the most difficult monitoring problem facing the radiation protection technologist is a radiation field containing neutrons. The mechanisms of neutron interactions are strongly energy dependent. Neutrons are uncharged and so they do not directly produce ionization in radiation detectors. Existing neutron detectors usually are quite energy dependent. The biological effects caused by neutron exposure also are energy dependent. Neutrons are almost never encountered without an accompanying gamma ray field. Even if a technologist was measuring a source which did emit mono-energetic neutrons (rare to encounter), the field would contain a wide spectrum of energies because of interactions of the primary neutrons with the air and the walls, ceiling and floor of the area. For these reasons, in radiation protection technology the simplifying assumption is frequently made that neutrons monitored are either "slow," meaning thermal or low energy, relatively slow moving neutrons, or "fast," which means high energy (fast moving) neutrons. NEUTRONS ONLY OCCASIONALLY INTERACT AND RELEASE ENERGY ALONG THEIR PATH OF TRAVEL.

In nuclear engineering, it is necessary to be more precise in defining neutron energies. In this case, neutrons are normally divided into several energy classes. Various authors argue about the exact energy ranges covered, but they are approximately as follows:

0 - 0.025 eV, cold neutrons
0.025 eV, thermal neutrons
0.025 - 0.4 eV, epithermal neutrons
0.4 - 0.6 eV, cadmium neutrons
0.6 - 1 eV, epicadmium neutrons
1 eV - 10 eV, slow neutrons
10 eV - 300 eV, resonance neutrons
300 eV - 1 MeV, intermediate neutrons

1 MeV - 20 MeV, fast neutrons
>20 MeV, relativistic neutrons

Note: A thermal neutron is one which has the same energy and moves at the same velocity as a gas molecule does at a temperature of 20° C. The velocity of a thermal neutron is 2200 m/sec, about 5,000 miles per hour!

Slow Neutron Interactions

The probability of interaction of slow neutrons is particularly dependent on their energy. The numerical probability (referred to as the "cross section") usually falls off inversely with the square root of energy. See Sample Problem 1. Neutron cross sections are measured in barns. One barn is equal to 10^{-24} square centimeters. Three of the more common types of slow neutron interactions are shown in Figure 1 along with a sample reaction of interest in radiation protection. In radiative capture the neutron is caught by a nucleus of an absorber atom which then leads to the emission of a photon (the capture gamma ray) to rid the atom of the excess energy. A similar process is involved in charged particle emission except that the excess energy is now in the form of an energetic charged particle rather than a photon. The final process, fission, was discussed at length in Chapter 2.

1. Radiative Capture: $^{1}_{0}n + ^{A}_{Z}P \rightarrow ^{A+1}_{Z}D + ^{0}_{0}\gamma$

Example: $^{1}_{0}n + ^{1}_{1}H \rightarrow ^{2}_{1}H + ^{0}_{0}\gamma$

2. Charged Particle Emission: $^{1}_{0}n + ^{A}_{Z}P \rightarrow D$ + Charged Particle

Example: $^{1}_{0}n + ^{10}_{5}B \rightarrow ^{7}_{3}Li + ^{4}_{2}He$

3. Fission: $^{1}_{0}n + ^{A}_{Z}P \rightarrow$ Fission Products

Example: $^{1}_{0}n + ^{235}_{92}U \rightarrow$ Fission Products

Fig. 1 - Slow neutron interaction mechanisms

Slow neutrons were discovered in 1934 by Enrico Fermi. He noticed that a fast neutron source produced more radioactivity in a piece of silver when the silver lay on a wooden tabletop (lots of moderating hydrogen) compared to a marble table.

Fast Neutron Interactions

Elastic and inelastic scattering are the chief processes by which fast neutrons produce dose in tissue. In elastic scattering, all the energy remains in the form of kinetic energy of motion. This is the familiar case of billiard balls bouncing off each

Sample Problem 1

GIVEN:
A 100 keV slow neutron is traveling through tissue.
FIND:
The change in its chances of interacting in the tissue as it slows to 10 keV.
SOLUTION:
The probability of interaction is inversely proportional to the square root of the energy. The new energy is 10/100 = 0.1 of the original. The square root of 0.1 = 0.32. The inverse of 0.32 (reciprocal) is 3.2 so the neutron has increased its chances of interacting by 3.2 times or 320%.

other. Recall that the maximum energy transfer in such collisions occurs when both objects that collide have the same mass. (To illustrate, a collision between a fly and a speeding automobile causes little energy transfer, but a head-on collision between two autos causes large energy transfer.) To most effectively stop fast neutrons it is necessary to allow them to collide with target atoms having a nuclear mass equal to the mass of a neutron. The best material is thus something rich in the element hydrogen since the mass of a proton (hydrogen nucleus) closely approximates the neutron mass. Shields of water, wax, concrete and various plastics are commonly used. In collisions with the protons in these materials, the neutrons lose half of their energy, on the average, per interaction. It turns out that elastic scattering is the mechanism responsible for about 80% of fast neutron dose to human tissue.

Inelastic scattering occurs if some of the collision energy is used to raise the target nucleus into a higher nuclear energy level. Subsequently, the nucleus will de-excite with the emission of a gamma ray. This is considered desirable, from a shielding standpoint, as the photon is easier to attenuate than a high speed neutron. Iron has a particularly high cross section for inelastic scattering of fast neutrons. In Chapter 11 we will see it is used for this purpose around high energy nuclear accelerator facilities. Scattering interactions are shown in Figure 2.

Fast neutrons also interact to cause emission of particles or capture gamma rays. Typically, a proton, alpha or a pair of neutrons are the particles released in the

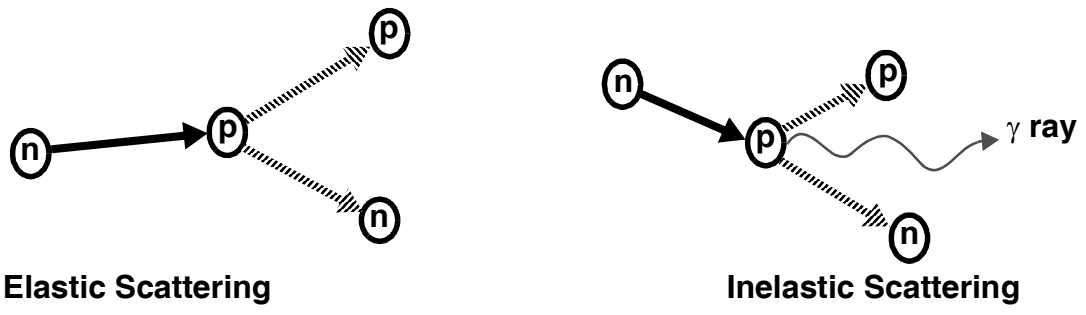


Fig. 2 - Fast neutron interaction mechanisms

interaction. If the isotope that remains following the interaction is radioactive, the process is called neutron activation. It is possible for neutrons of any energy to produce activation, not just fast neutrons. An example of a practical use of neutron activation would be production of medical radioisotopes in a small nuclear reactor. An example of a problem caused by neutron activation would be production of hazardous Co-60 levels in reactor cooling systems due to neutron activation of cobalt compounds present in trace amounts in the water.

Indirectly Ionizing - Photon Interactions

Introduction

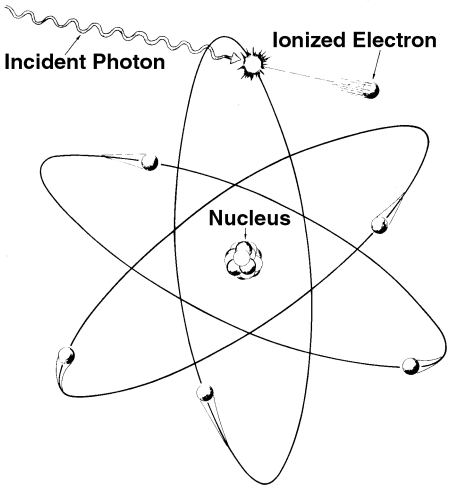
Photons are the second major type of indirectly ionizing radiation. They are merely bundles of electromagnetic energy. The word is a Greek term meaning "light," and it was first suggested for electromagnetic radiation by Arthur H. Compton, the discoverer of Compton Scattering. Some common examples of photons include gamma rays, x-rays and bremsstrahlung.

In an absorbing material, PHOTONS ONLY OCCASIONALLY INTERACT AND RELEASE ENERGY ALONG THEIR PATH OF TRAVEL. For example, only half of the gamma rays emitted by Co-60 will interact in passing through the abdomen of Reference Man. The rest will pass right on through and exit from the back side with their full energy intact. We will see below that charged particles give up energy rather continuously along their path, and that charged particles all give up some energy in passing through an absorber. Because of the erratic energy loss behavior for photons we are forced to discuss the probability or chance of an interaction rather than the amount of energy lost over some path length. Generally, the chance of avoiding an interaction decreases exponentially with absorber thickness. Thus, only an infinitely thick absorber or shield could stop all incident photons. In practical cases, nuclear facilities don't have room for infinitely thick shields so we put as much material in the path as possible and reduce the beam intensity (and hence the dose rate) to a reasonably low level.

Photons are characterized by their mean free path. This is defined as the average (mean) distance of travel in the medium between interactions. Note that while an absorber which is the thickness of the range of a beam of charged particles will remove 100% of them from the beam, over the distance of one mean free path, each photon in the beam has only a 50% chance of interacting.

Photoelectric Effect

The first of the three major mechanisms by which photons interact with matter is the Photoelectric Effect. This process is illustrated in Figure 3. The incoming photon which "decides" to interact by the photoelectric effect transfers its full energy to an inner shell orbital electron, almost always one in the K shell. The chance of an L shell electron participating is less by a factor of 1000. The energized electron uses the excess energy to escape from the nucleus, i.e., it gives up the binding energy, and



Chance per gm $\propto Z^3/E^3$

Fig. 3 - Photoelectric Effect mechanism

escapes the atom with the remaining energy given to it by the photon. Note that in a photoelectric interaction, ALL PHOTON ENERGY DISAPPEARS IN THE INTERACTION. In the remaining two types of interactions to be discussed, residual photon energy is still present after the interaction.

An important consideration in each of the three mechanisms for photon interactions is the way in which the chance or probability of the interaction depends on both the photon energy and the atomic number of the absorbing medium. In the case of the Photoelectric Effect, the relative probability of an interaction per gram of absorber is directly proportional to the cube of the atomic number, Z, and inversely proportional to the cube of the energy of the photon. This relationship is also shown in Figure 3. The direct dependence on Z^3 means that the chance of the photoelectric effect occurring is extremely dependent on the elements which compose the absorber. If the absorber is a chemical compound rather than elemental, the effective atomic number is used. For example, soft human tissue has an effective atomic number of 7.5. See Sample Problem 2 for a calculation. The significance of the inverse cubed dependence on energy is that photoelectric effect is the predominant mechanism only for LOW ENERGY PHOTONS. A criterion for deciding whether a given photon energy is low, medium or high will be given later in this chapter.

Some additional comments on the "effective atomic number," Z_{eff}, of a compound are in order. This concept is often used when calculating the chances of a photon interacting in some absorber which contains more than one element. (In the case of a single element absorber, there is no problem – the photoelectric probability is proportional to Z^3. But what is the "Z" of water?) The earliest derivation of the equation for calculating Z_{eff} for a mixture of elements was made in 1937 by W. Mayneord in London's Royal Cancer Hospital.

$$Z_{eff} = \{a_1 Z_1^{2.94} + a_2 Z_2^{2.94} + \ldots\}^{1/2.94}$$

The a_1 and a_2 etc. are the fraction of the total electrons in the compound that are in the 1st, 2nd, etc. element of the compound. Thus,

Sample Problem 2

GIVEN:

A technician accidently activates an x-ray tube with the lead shield removed.

FIND:

How much more effective was the lead compared to the technician's soft tissue in stopping the low energy x-rays?

SOLUTION:

Low energy x-rays are absorbed by the photoelectric effect and thus the absorber effectiveness is directly proportional to the cube of its Z. The relative effectiveness of two absorbers is just the ratio of the cube of their Z #s. Thus, lead is $82^3/7.5^3$ or 1,307 times more effective per gram than tissue in stopping the x-rays.

for water, H_2O, there are 10 electrons so a_1 = 2/10 for the hydrogen and a_2 = 8/10 for the oxygen. The 1/2.94 exponent on the bracket means that you calculate the value in the bracket and then take the 2.94th root on a calculator to get Z_{eff}.

Applying the equation to some compounds of interest in radiation protection, the following results are obtained for the effective atomic number of the compounds listed for low energy (photoelectric) interactions:

Compound	Z_{eff}
Air	7.64
Soft Tissue	7.51
Water	7.42
Bone	11.6 - 13.9
Acrylic Plastic	5.85
Polyethylene	5.27

It is worth noting that air, water and soft tissue all have the same effective atomic number within ± 0.1 unit, i.e., 7.5.

Compton Scattering

The events which occur in a Compton Scattering type of interaction are shown in Figure 4. The incoming photon transfers a portion of its energy to an orbital electron. A lower energy photon then leaves, in a different direction, with the remaining energy. Practically speaking, any orbital electron which has a binding energy less than about 10% of the photon energy is available to interact by a Compton process. Thus, virtually all the orbital electrons except the two in the innermost K shell are available. (Generally, the K shell binding energy is too high for them to "scatter.") Note again that the interaction does not remove all the photon energy from the incoming ray – a residual "Compton photon" is produced. The Compton electron is ejected from the atom with the energy it receives minus the electron binding energy.

Many technologists have raised the question of whether the scattered photon is actually the "same" photon that struck the electron only at reduced energy. The field of quantum electrodynamics answers this question. NO! The scattered photon is actually a different one created at

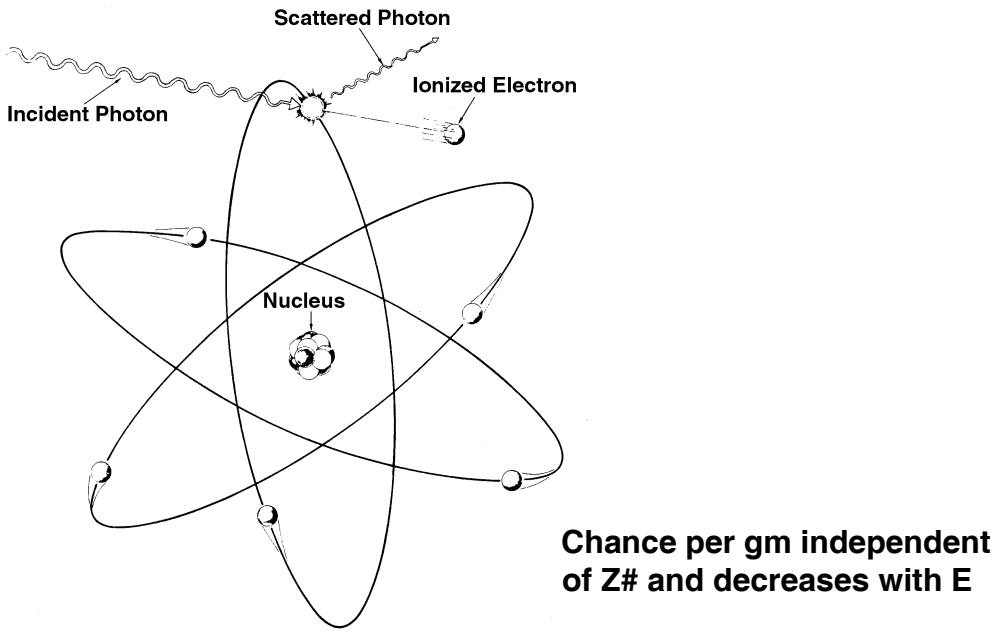


Fig. 4 - Compton Scattering mechanism

the time of the interaction. The theory even says that the scattered photon can be emitted before the incoming photon is absorbed!

Whereas the probability of a photoelectric event is strongly atomic number dependent, the probability or chance for Compton Scattering, per gram of absorber, is INDEPENDENT OF ATOMIC NUMBER for Compton Scattering. This can be explained by reference to a little known fact of nature. All elements, with the sole exception of hydrogen, have approximately the same number of electrons per gram (about 3 X 10^{23} or half of Avogadro's number). Since virtually all these electrons are available to participate in a Compton scattering process, a gram of almost everything provides the same probability of interaction. Thus, from a shielding point of view, a gram of popcorn provides about the same shielding as a gram of lead for cobalt-60 gamma rays (1.25 MeV average). Admittedly, due to the density difference, a larger thickness of popcorn is required to make an equivalent shield compared to lead. Still the fact remains that lead is not inherently superior in shielding gamma rays due to its high atomic number but only because it is more dense.

The probability, per gram of absorber, for Compton scattering does depend on the energy of the photon. Generally, the probability decreases as the energy increases. The actual mathematical description of this behavior cannot be expressed in simple terms, as was the case for the photoelectric effect.

The mechanism of Compton Scattering was first fully explained in 1923 by physicist Arthur H. Compton. He correctly suggested that the photons could be thought of as carrying a bundle of energy like a billiard ball, and that the resulting angles and energies of the electron and residual photon could be computed using the classical laws of physics, i.e., conservation of energy and momentum. This explanation was initially rejected by others because it was felt that photons and other forms of electromagnetic energy were really waves which could not exhibit particle-like properties. Compton received the Nobel Prize in physics in 1927 for this discovery.

Pair Production

The third and final major mechanism for photon interactions is Pair Production. This interaction mechanism is illustrated in Figure 5. In the vicinity of the nucleus of an absorber atom, the incoming photon suddenly disappears and in its place appears a pair of particles - an electron and a positron. This pair has been created out of the pure energy carried by the photon. Recall that Einstein predicted this possibility in his famous $E = mc^2$ formula introduced in Chapter 2. If the entire rest mass of an electron is converted into energy, 0.511 MeV is produced. The positron, being the anti-particle of the electron also "weighs" 0.511 MeV of mass-energy. Since fractions of particles do not exist, the minimum amount of mass-energy that is needed to make the complete pair of particles is 0.511 MeV + 0.511 MeV = 1.022 MeV. This energy must be carried by the incoming photon, as a minimum, in order for the interaction to proceed by the process of pair production. The term "threshold" is applied to the process to indicate this requirement.

It should be noted that while in theory a 1.023 MeV photon could pair-produce, the likelihood (probability) is almost zero. In fact, if the absorber is tissue, pair production interactions account for less than 10% of the total interactions for photon energies up to 5 MeV.

The probability of pair production occurring, per gram of absorber, is directly proportional to the atomic number, Z. Thus, the Z dependence is much less pronounced for pair production than for the photoelectric effect which showed a Z^3 dependence. In terms of the photon energy, the probability per gram increases with energy (above the 1.022 MeV threshold energy), in a non-simple mathematical way. Referring back to the earlier Sample Problem 2 of lead and tissue, a gram of lead will attenuate high energy gamma rays, through Pair Production, by about 82/7.5 times or only 11 times more than a gram of tissue.

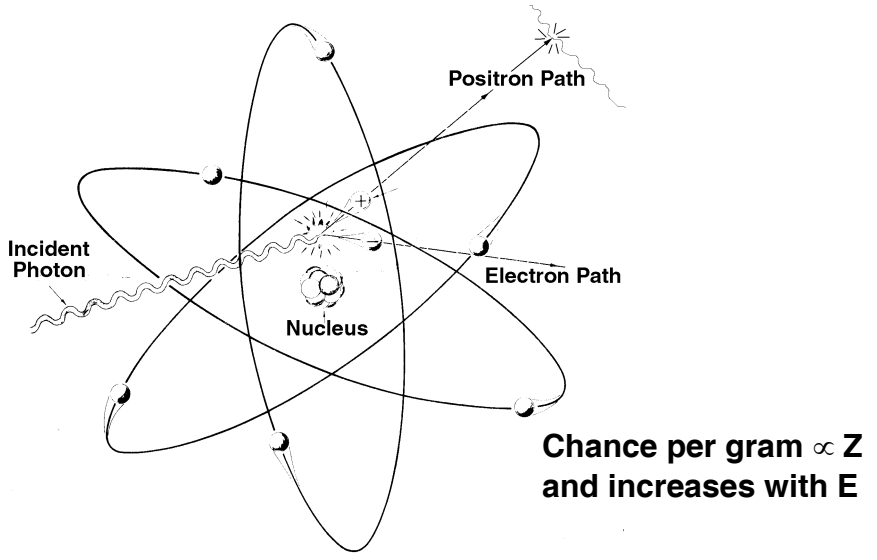


Fig. 5 - Pair Production

Sample Problem 3

GIVEN:

A photon has an energy of 1 MeV.

FIND:

What is the frequency and speed of this photon? What is the chance of pair production occurring in a lead brick?

SOLUTION:

From Fig. 6, E = h f so f = E/h = (1 MeV x 10^6 eV/MeV) / 4.14 x 10^{-15} eV-sec or 2.4 x 10^{20} /sec or 2.4 x 10^{20} Hz. All electromagnetic radiation travels at the speed of light, c = 3 x 10^8 m/sec. The energy is below the threshold of 1.022 MeV so pair production is not possible.

Consider now the fate of the pair of particles formed from the photon. Any excess energy (energy > 1.022 MeV) carried by the photon is shared equally between the electron and positron, in the form of kinetic energy which will carry them away from the formation site. These particles, being charged, continuously lose energy as they move through the absorber. When the positron loses most of its kinetic energy, it will be captured by the attractive Coulomb force of some nearby electron and the two will annihilate with the release of two photons each carrying 0.511 MeV of energy traveling in exactly opposite directions. Thus, the process of pair production and annihilation, which has been observed countless times in the laboratory, demonstrates the validity of the Einstein mass-energy relationship in both directions – energy into matter and matter into energy.

In performing energy calculations it often is necessary to compute the energy carried by a photon. Figure 6 shows the simple relationship that exists between the photon energy, E, and the frequency, f, of the electromagnetic wave packet. The constant of proportionality, h, is called the Planck Constant in honor of the German physicist Max Planck. See Sample Problem 3.

For electromagnetic photons, E = h f

where h = Planck's Constant and f = the frequency.

Note: h = 4.14 x 10^{-15} (eV-sec) or 6.63 x 10^{-34} (Joule-sec)

Fig. 6 - The energy of a photon

Absorption and Attenuation Coefficients

It is desirable to be able to describe the loss of energy and intensity of a photon beam penetrating an absorber. The relevant concepts involve the absorption coefficient and attenuation coefficient of the medium. These coefficients represent the probability or the cross section for interaction. The total probability of some interaction taking place is given by the Greek letter mu, μ, which is defined as:

μ = **total linear attenuation coefficient**

$= \mu_{pe} + \mu_{cs} + \mu_{pp}$.

It physically represents the sum of the individual probabilities, μ_{pe} etc., for photoelectric, Compton and pair production interactions. Since the chance of each of these separate interactions is energy dependent, μ will also depend on photon energy. A generalized plot of μ versus photon energy is shown in Figure 7.

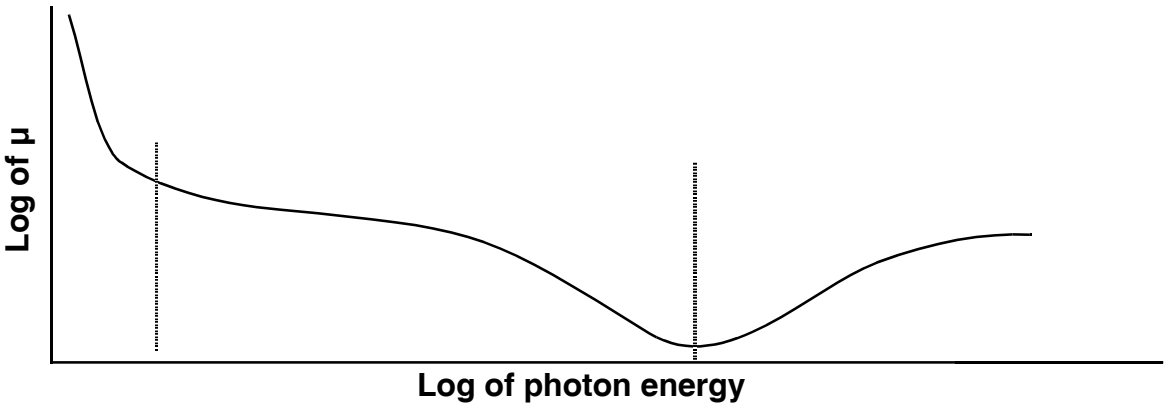


Fig. 7 - The linear attenuation coefficient vs. energy

Notice that this curve has three distinct regions. This is due to the fact that there are three separate interaction processes contributing to the total probability. Also, note that the plot is a log-log plot so as to cover a wide range of values. The nature of the curve becomes even clearer when the three individual components making up the total are shown. Figure 8 illustrates this by plotting the photoelectric, Compton and pair production cross sections (probabilities) as dashed lines with the solid line representing their algebraic sum on the log-log plot.

Figure 8 also provides a method for defining what was meant earlier in this chapter by low, medium and high energy photons. The "dividing line" between low and medium occurs at the energy where the photoelectric and Compton interactions

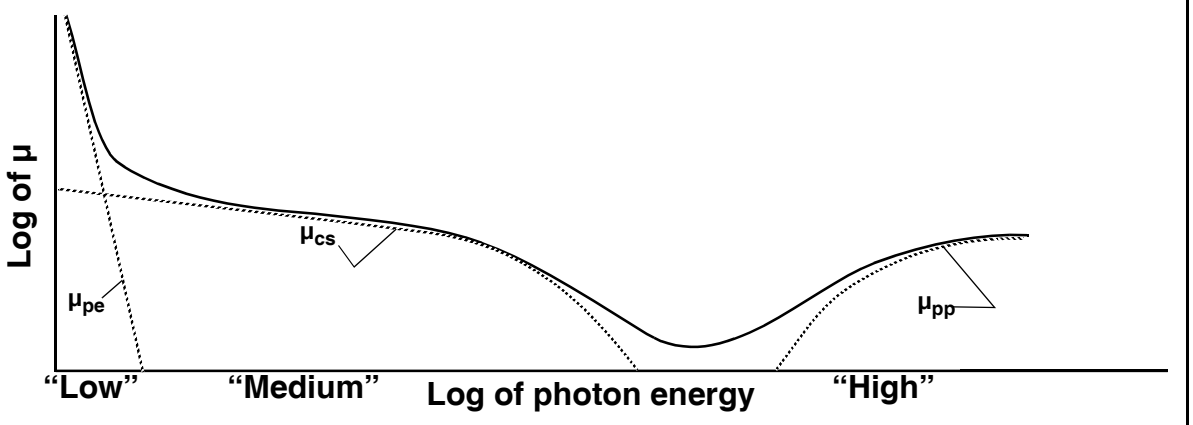


Fig. 8 - The components of the linear attenuation coefficient

are equally likely, i.e., at the intersection of the first two dotted lines representing those processes on the curves in Figure 8. In tissue the energy at which this intersection occurs is about 25 keV; in aluminum and bone about 50 keV; in lead, about 700 keV. These different energies in different materials again emphasize the Z dependences of the various processes. The line between medium and high energy photons is the energy of equal probability of the Compton and pair production interactions. In tissue this occurs at 10 MeV, in bone and aluminum at 7 MeV and in lead at 3 MeV.

Another way of thinking of the actual meaning of the coefficient μ is that it gives the fraction of the photons in the beam which interact per unit distance of travel. If μ is expressed in per cm units (cm^{-1}), then it is numerically equal to the fraction of interactions, by any process, in an absorber of 1 cm thickness. As described earlier, μ will depend on the number of electrons in the path. Hence it changes with absorber density. Ice, water and steam have quite different values of μ at any given energy even though they are the same chemical substance. To eliminate this possible annoyance, the density dependence is often removed by dividing μ by the density. This gives a new coefficient, called the total mass attenuation coefficient. The total mass attenuation coefficient, μ/ρ, represents the probability of interaction per unit density of material. It is expressed in units of square cm per gram:

μ (/cm) ÷ ρ (gm/cubic cm) = μ/ρ (/cm)/(gm/cubic cm) = μ/ρ (cm^2/gm).

The theory of photon interactions predicts that attenuation will decrease the beam intensity exponentially with depth into the absorber. This is written in equation form as shown in Figure 9. To make practical use of this equation, the value for μ is obtained from a plot of μ vs. energy such as Figures 10 – 12. Sample Problem 4 below illustrates an attenuation law calculation.

$$I_x = I_0 e^{-\mu x}$$

where I_x = photon intensity after x cm of penetration

and I_0 = unattenuated photon intensity

Fig. 9 - Exponential attenuation law for photons

Sample Problem 4

GIVEN:

1 MeV gamma rays are emitted by an underwater source.

FIND:

What effect would 2 cm of water have on the intensity of the beam?

SOLUTION:

From Fig. 10, μ = 0.07/cm for 1 MeV photons. The intensity with the water present, I_x is related to the intensity without the water, I_0, by the equation in Fig. 9, i.e.,

$$I_x / I_0 = e^{-\mu x} = e^{-(0.07/cm)(2\ cm)} = e^{-0.14} = .87 = 87\%.$$

So the 2 cm of water would reduce the intensity from 100% to 87% or by 13%.

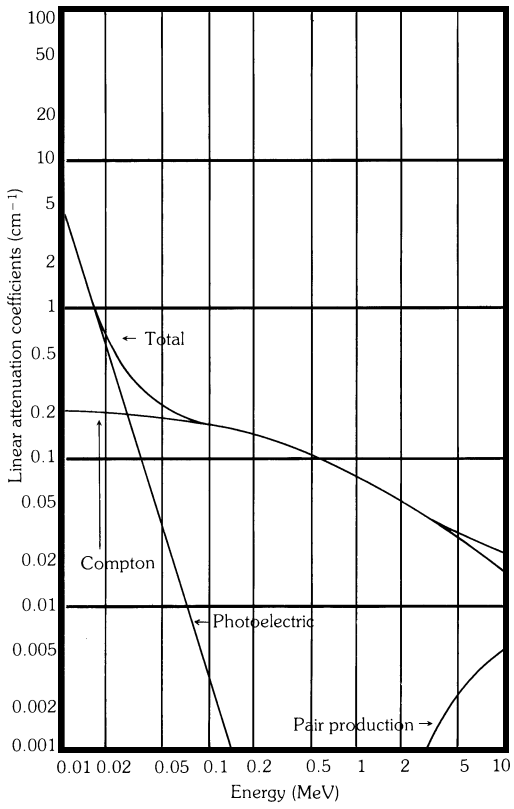


Fig. 10 - Linear attenuation coefficient of water

Courtesy of Harshaw/Filtrol Partnership

In practice, the exponential attenuation just discussed only strictly applies to THIN shields. This is due to the underlying assumption, in the derivation of the applicable theory, that an interaction is equivalent to the total removal of that photon and all of its energy. As stressed earlier, only the photoelectric interaction truly removes all photon energy (by converting it to kinetic energy of an orbital electron). In Compton and pair production interactions, residual photon energy remains in the form of lower energy gamma rays moving through the absorber. Thus, the exponential attenuation equation will UNDERESTIMATE the dose rate behind a shield. This problem can be dealt with in two ways. One is to insert a "correction factor" into the exponential attenuation equation to take into account the buildup of residual photon energy which contributes to dose. This buildup factor method is discussed more fully in Chapter 11.

The second approach to calculating the effect of a thick absorber on dose rate is to make use of the linear and/or mass energy absorption coefficients. These are represented by the symbols μ_{en} and μ_{en}/ρ. The linear energy absorption coefficient, μ_{en}, represents the fraction of energy actually removed from photons in the beam per unit distance (e.g., per cm), while the corresponding mass coefficient, μ_{en}/ρ, gives the fraction of energy removed per unit density of absorber. Since dose is energy deposited per unit mass, calculations using absorption coefficients instead of attenuation coefficients more closely estimate the reduction in dose rate as a result of adding shielding around a photon radiation source.

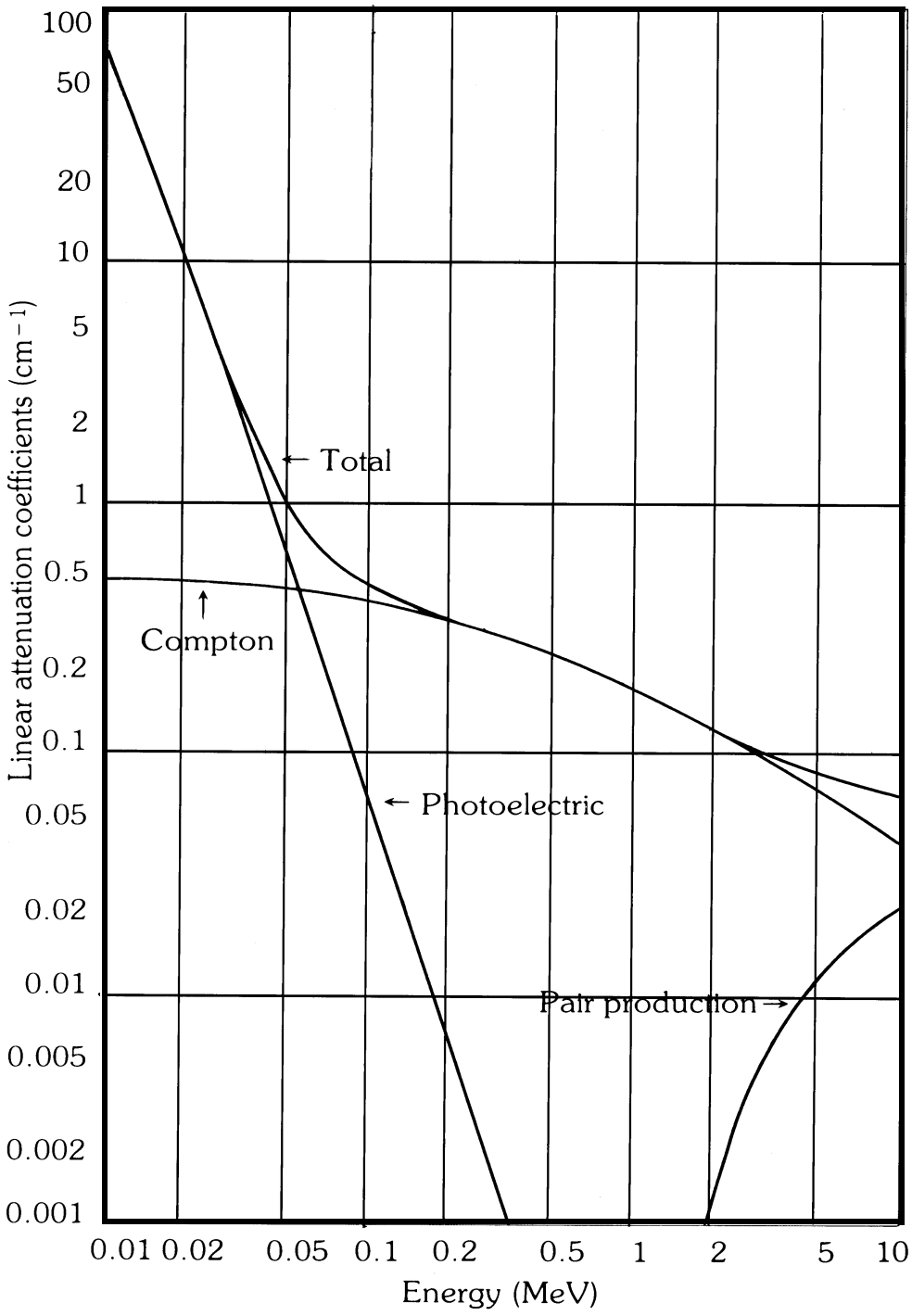


Fig. 11 - Linear attenuation in aluminum

Courtesy of Harshaw/Filtrol Partnership

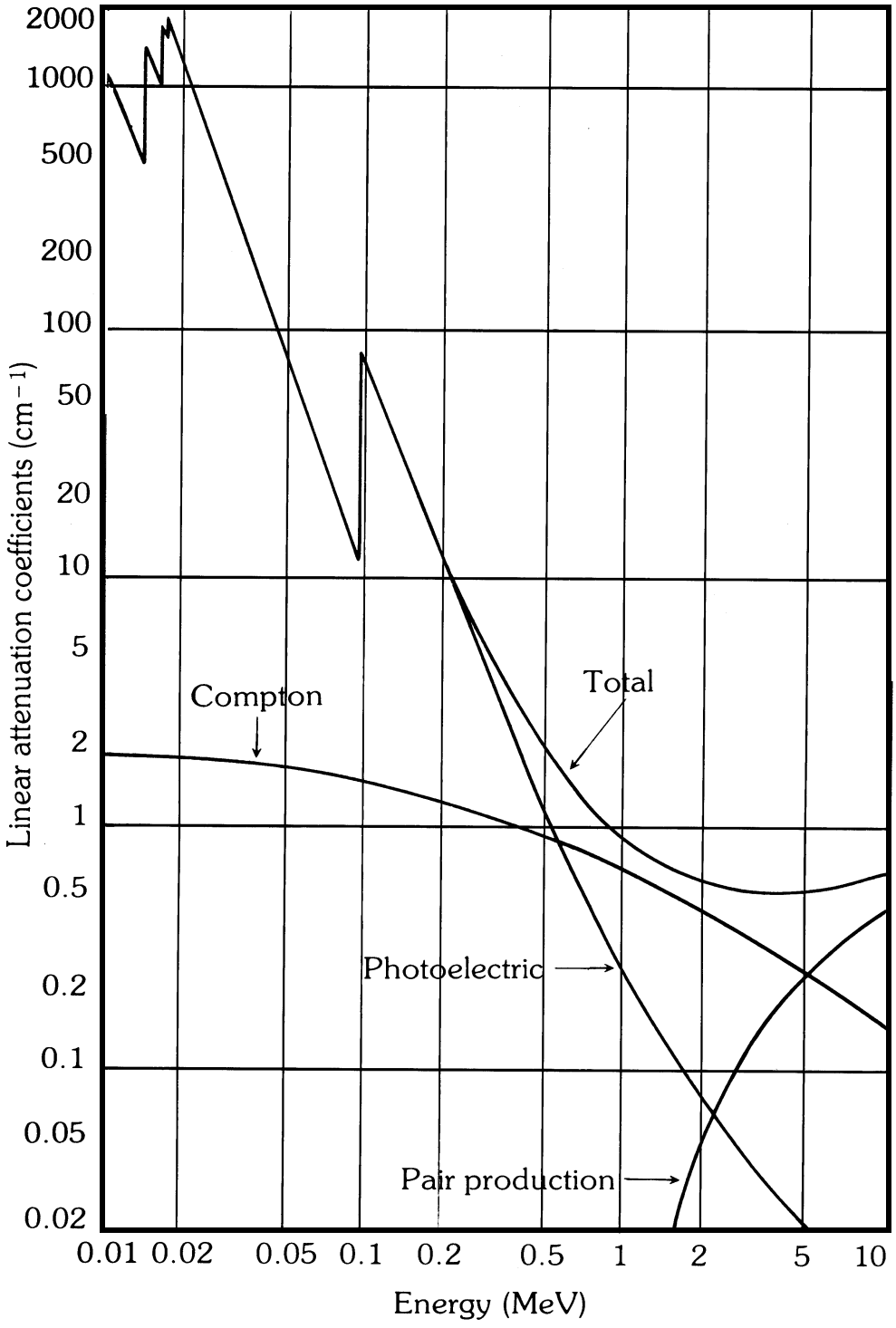


Fig. 12 - Linear attenuation in lead

Courtesy of Harshaw/Filtrol Partnership

In an attempt to summarize the behavior of photons as they interact with matter, a series of computer simulations is presented to conclude this section. Using a random number generator and given the interaction probabilities at various energies, a computer can visually show the fate of photons striking an absorber. In the examples given here in Figures 13 and 14, a slab of water (or, for all practical purposes, soft human tissue) one centimeter thick is struck by 25 gamma rays arriving from the left-hand side of the slab (rectangle). Straight lines exiting out the right-hand side represent gammas that have passed through unattenuated. Lines ending inside the slab are photoelectric events. Low energy Compton events are shown by the dashed Compton gamma ray leaving the site. Higher energy Compton events also show the heavier track of the Compton electron. Two heavy tracks leaving a site represent pair production, the tracks being, of course, the electron and positron which have high kinetic energy directed toward the right in the slab. Note how the number of gammas making it through without interacting increases as the energy is raised.

Directly Ionizing - Charged Particle Interactions

Energy Loss Mechanisms

Some examples of charged particles which might be encountered by a radiation protection technologist include alpha particles, beta particles and some types of mesons (in the vicinity of high energy nuclear particle accelerators). They are all considered directly ionizing radiations. The major rule which applies here is that CHARGED PARTICLES GIVE UP THEIR ENERGY, TO AN ABSORBER, CONTINUOUSLY ALONG THE PATH OF TRAVEL. Thus, a charged particle will always leave an absorber with less energy than it entered with. This behavior is NOT shown by uncharged, indirectly ionizing radiations as discussed earlier. A gamma ray can penetrate a thick lead shield and still have the same energy it had at the point of entry.

Actually there are over a dozen ways in which charged particles interact and deposit energy in matter. In practice, at the energies that the technologist usually works with, there are three major mechanisms which account for almost all of the energy deposited: ionization, excitation and bremsstrahlung. The process of IONIZATION involves the complete removal of orbital atomic electrons as a result of the Coulomb force between the charged particle and the orbiting electrons (which, of course, are themselves negatively charged). This process removes charge from a neutral atom and so it becomes an ion. The combination of the removed electron and the residual positive ion is called an ion pair. This mechanism adds electric charges to the absorber. In many materials the amount of energy needed to produce an ion pair is about 30 to 40 eV. This small amount of energy is named the W value. For air, the internationally official W value is 33.9 eV per ion pair.

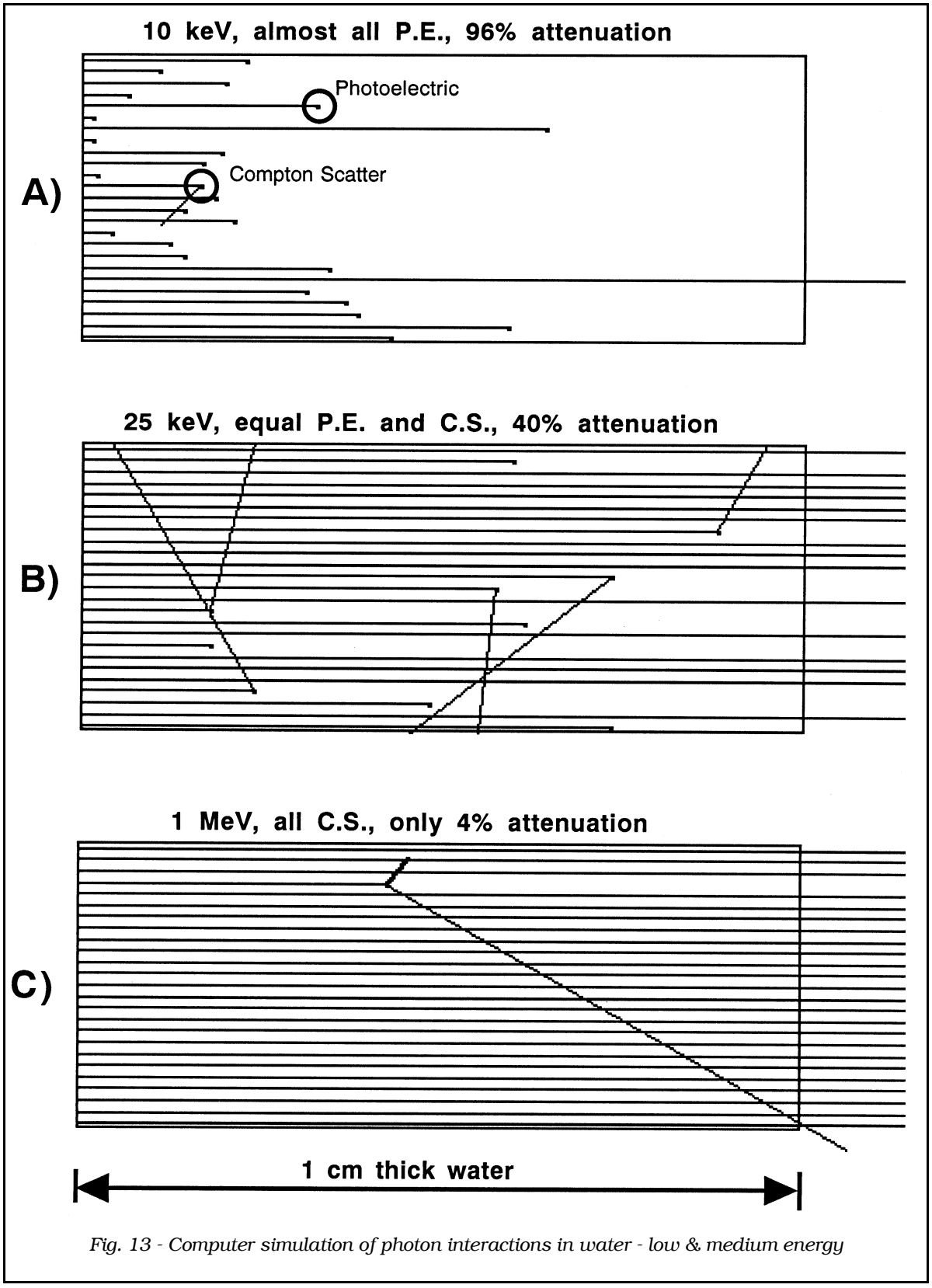


Fig. 13 - Computer simulation of photon interactions in water - low & medium energy

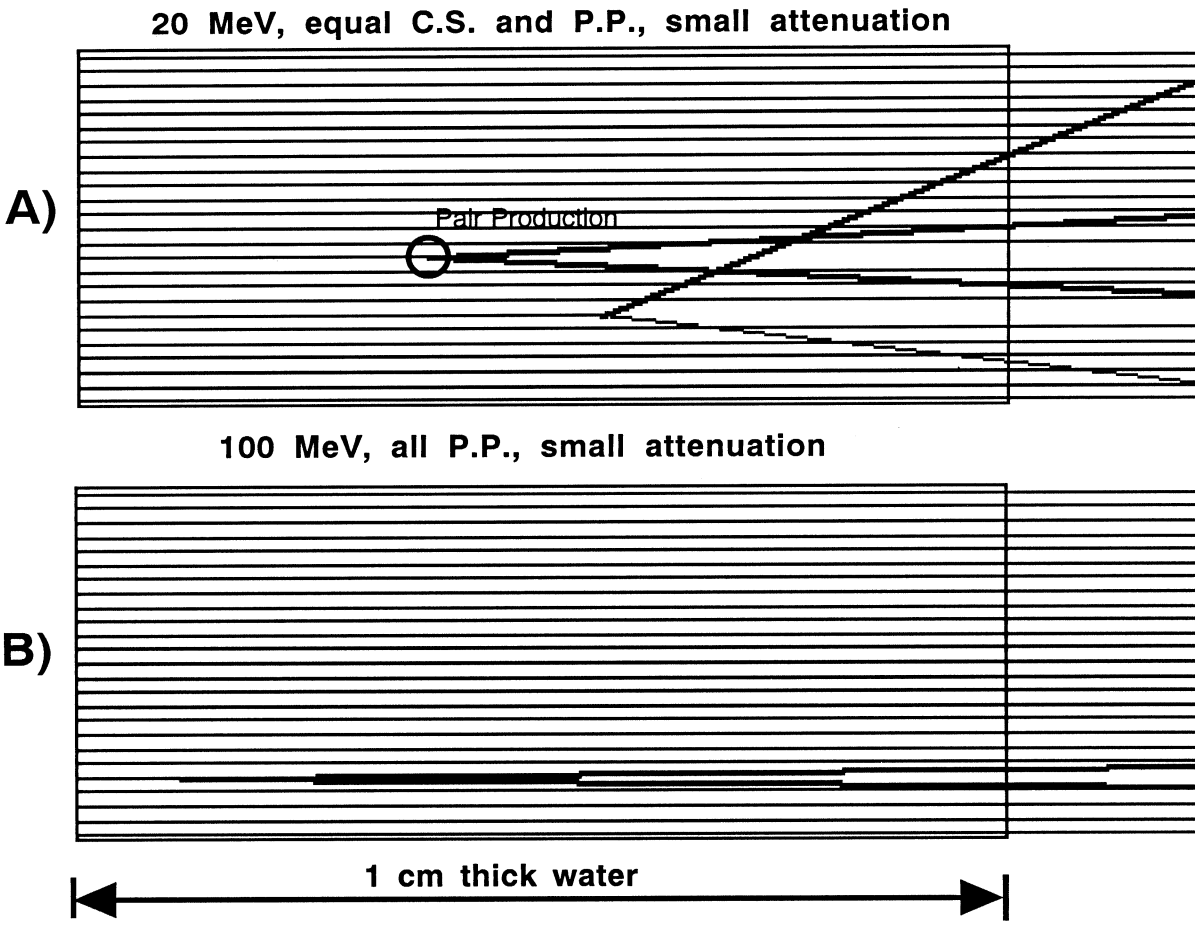


Fig. 14 - High energy computer simulation in water

Older texts quote a value of 33.7 eV/ion pair for W. For years, that was the accepted value. In 1979, the International Commission on Radiation Units and Measurements reexamined the basis for the international standard. It found that corrections for humidity in the air in which W had been measured had not been done correctly by some investigators. When the proper corrections were applied to the earlier data, a slight increase in W value resulted. The current internationally recommended value is 33.85 ± 0.15 eV/ion pair in dry air.

The energy loss mechanism termed EXCITATION is also a consequence of the Coulomb force between the charged particle and atomic electrons. In this case, in contrast to ionization, insufficient energy is transferred to the orbiting electron to break the electrical binding force so the electron merely jumps up to a higher atomic energy level rather than leaving the atom entirely. Note that in the case of excitation, the electron is still bound to the atom so that the electrical neutrality of the atom is not disturbed. This process does NOT lead to the formation of ion pairs and no free charge is released into the absorber.

The last of the three major atomic processes of energy removal is named BREMSSTRAHLUNG. It is a word of German extraction which translates literally as "braking radiation," that is, radiation generated when the charged particle puts on the brakes. The causative agent is again the Coulomb force which produces a deflection in the path of the charged particle. This change in direction is, in the physics sense, a negative acceleration (a de-celeration if you prefer) because the velocity vector changes with the directional change and the speed of the particle is reduced due to the energy loss. The radiation emitted by the particle, the bremsstrahlung, is electromagnetic in nature. It usually has an energy that puts it in the x-ray region of the overall electromagnetic spectrum. In fact, the major portion of energy radiated by most x-ray tubes is due to bremsstrahlung. A technical analysis of the process shows that the intensity of bremsstrahlung is proportional to the energy of the charged particle and also proportional to the atomic number, Z, of the absorber. Finally, the bremsstrahlung is inversely proportional to the square of the mass of the charged particle. (At a given energy, a proton will produce about 2000^2 times less bremsstrahlung than an electron). These results will be important in shielding considerations to be dealt with in Chapter 11. Also, some calculational examples will be given there.

Generally, all three processes described are occurring simultaneously as a beam of charged particles passes through matter. In the case of human tissue as the absorber, ionization and excitation account for about 99% of the energy deposited while bremsstrahlung accounts for the remaining 1%. Figure 15 summarizes the three microscopic energy loss mechanisms described.

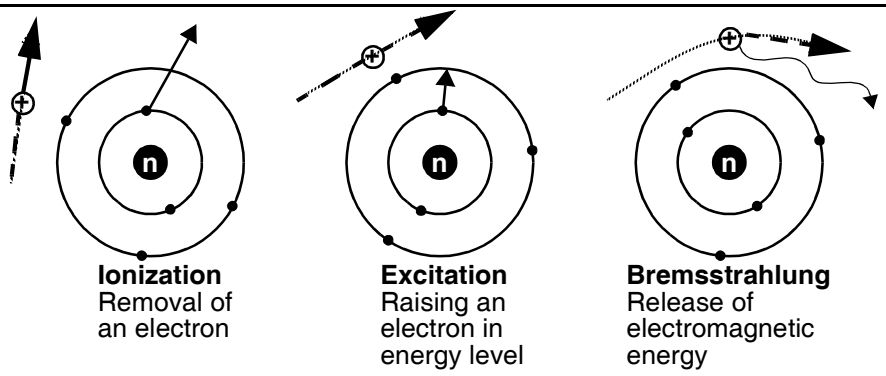


Fig. 15 - Charged particle energy loss mechanisms

Stopping Power

There are several concepts which are useful in dealing with charged particles at the macroscopic or real world level in contrast to the microscopic or atomic level. The first of these is stopping power, S, which is defined to be the average energy lost by a charged particle per unit distance of travel. It is sometimes measured in units of MeV/cm. Stopping power is a property of an absorber. It is analogous to the concept of the attenuation coefficients for gamma rays. It describes how effective the absorber is in removing energy from a beam of charged particles. The higher the S, the better the material acts as a shield. Microscopically, the interactions which result in energy

removal from the particle occur randomly along the path. By defining stopping power as the average energy loss, the effect of the random variations is then smoothed out.

Since the energy loss occurs by ionization, excitation and bremsstrahlung, each of which depends on the strength of the Coulomb force, it should be clear that S for a given particle-absorber situation will depend on the factors influencing the Coulomb force, F. Now F depends on the amount of the charge on the two entities involved and is inversely proportional to the square of their relative separation distance. Therefore, the stopping power depends on the charge of the charged particle and on the atomic number, Z, of the absorber which represents its charge. In calculating stopping power, the inverse distance squared factor might be thought of as representing the average distance of closest approach of the particles to the absorber atoms as they pass through. Thus, the stopping power depends on the density of the absorber. Finally, since low energy (slow moving) particles are subject to the Coulomb forces of an absorber atom longer than a high energy (fast moving) particle, more energy is transferred to the absorber and so S increases as particle energy decreases. The variation in relative value of S with energy, particle charge and absorber Z is illustrated by Figure 16. As predicted, alpha particles (charge = 2) have higher S than betas (charge = 1) and S for lead (Z = 82) exceeds the value in aluminum (Z = 13) at a given energy.

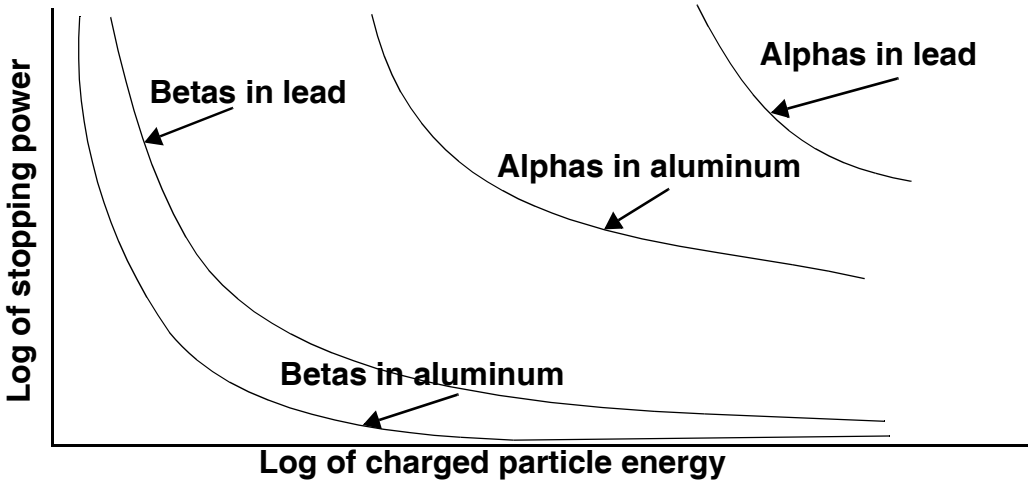


Fig. 16 - Stopping power versus particle energy

Since the stopping power depends on density, before a numerical value can be assigned in a given case, the density of the absorber must be known. As an example, the numerical values of S would be quite different for ice, water and steam, even though the absorber is the same chemical compound in all three cases. To get around this complication, use is often made of the mass stopping power, i.e., the stopping power per unit density. It is calculated simply by dividing the stopping power by the density. Thus:

Mass Stopping Power = S/ρ.

Ice, water and steam all have the same value of mass stopping power, S/ρ.

Specific Ionization

This macroscopic quantity is a close relative of stopping power. It is defined as the average number of ion pairs produced per unit distance of travel of a charged particle. It can be calculated from the stopping power and W value as follows:

Specific Ionization = S/W (ion pairs/cm).

As mentioned earlier, the W value in air (and in many gases used in radiation detectors) is 33.9 eV/ion pair. The specific ionization or density of ion pairs can be calculated for a gas-filled detector by dividing the stopping power (expressed in eV/cm) by 33.9 eV/ion pair. The concept is useful in the discussion of the operation of radiation detectors. Knowing the dimensions of a detector, the number of ion pairs produced by a particle passing on through can be calculated. This in turn allows the size of the detector signal to be calculated. (See Sample Problem 5)

Sample Problem 5

GIVEN:

A high energy proton passes through the 10 cm diameter of an air filled ion chamber.

S = 0.22 MeV/cm in this case.

FIND:

What charge would be collected in the chamber per proton?

SOLUTION:

We need the number of ion pairs deposited along the 10 cm path length. This is just the specific ionization (S/W) times the path length. Ion Pairs$_{deposited}$ = S/W (ion pairs/cm) x 10 (cm) = [0.22 MeV/cm x 10^6 eV/MeV / 33.9 eV/i.p.) x 10 cm = 6.5 x 10^4 i.p.

Each ion pair carries 1.6 x 10^{-19} coulombs of one sign.

So, the total charge deposited per proton is 6.5 x 10^4 i.p. x 1.6 x 10^{-19} coulombs = 1.0 x 10^{-14} C.

Range of a Charged Particle

The range, as used here, means the average depth of penetration of a charged particle into an absorber before it loses all its kinetic energy and stops. Note that photons (gamma and x-ray radiation) DO NOT HAVE A RANGE. Photons are characterized by their mean free path. Only charged particles travel a fixed, predictable distance in an absorber. If the stopping power is known, it is possible to compute the range of a particle. This is, however, more tricky than may be thought at first glance. If a charged particle has, for example, 1 MeV of energy and enters an absorber with an S of 0.5 MeV/cm, the particle will NOT travel an average of 2 cm. This is because of the fact that the value of S increases as the particle slows down. It will therefore travel less than 2 cm on the average. If the stopping power versus energy curve is known for this situation, it would then be possible to quite accurately calculate the range in that absorber.

If the stopping power is high, the particle will slow down rapidly and so the

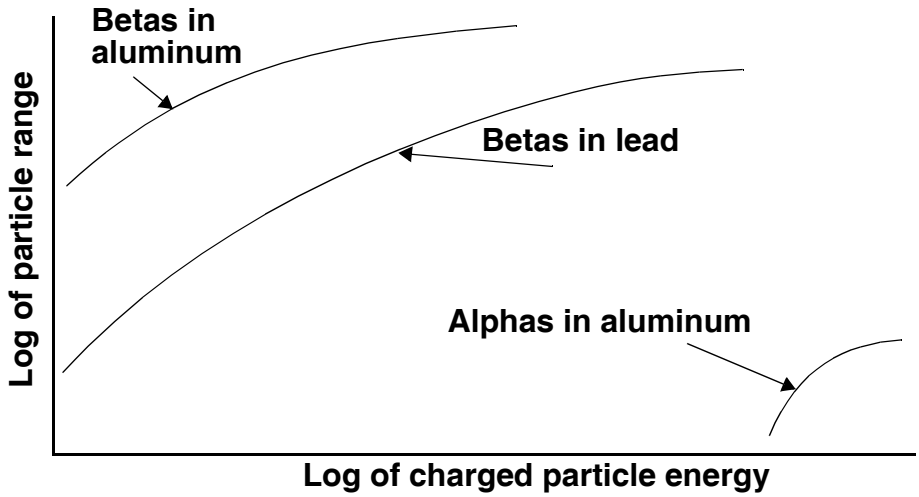


Fig. 17 - Range – Energy curves for charged particles

range will be small. The range is thus inversely related to S. It, too, will depend on all the factors influencing the Coulomb force, namely the charges on the two entities, the density of the absorber and the charged particle energy. Representative curves of the Range vs. Energy for some charged particles are shown in Figure 17. Note that these curves are the "mirror images" of the curves of S vs. E in Fig. 16.

The radiation protection technologist should be familiar with the approximate ranges of common charged particles in common materials. Generally speaking, alpha particles are stopped by a sheet of paper or aluminum foil while betas of ordinary energies will be stopped by a stout sheet of cardboard. Some handy rules of thumb for charged particle ranges are given in Figure 18.

1. Alpha particles up to 7.5 MeV are stopped in the dead layer of normal human skin.

2. Beta particles will penetrate about 4 meters in air per MeV of energy.

3. Beta particles will penetrate about 0.5 cm in soft tissue per MeV of energy.

4. Beta particles up to 70 keV are stopped in the dead layer of normal human skin.

Fig. 18 - Rules of thumb for charged particles

Before leaving this topic it bears repeating once more – charged particles have a definite predictable range. If a thickness of absorber greater than their range is placed in their path, 100% of all of the particles will be stopped.

In shielding problems (coming up in Chapter 11) it is sometimes necessary to calculate the thickness of a material needed to totally stop charged particles such as beta rays from an isotopic source. Of course, this involves merely determining the range of the most energetic beta in the proposed shield material. Since, in general, the range depends on the density of the material, it has been found to be convenient to specify

ranges of charged particles in terms of "density thickness" rather than ordinary, linear thickness. By using this "trick," the range then becomes independent of the density of the shield. Density thickness is found by taking the ordinary linear thickness and multiplying by the material density, i.e.,

Density Thickness (g/cm^2) = Thickness (cm) x density (g/cm^3).

Notice that the units of density thickness always have the dimensions of mass per unit area.

For the particular case of beta particles, an equation has been empirically derived that allows the range to be calculated in any material. It is valid for any beta emitter with energies below 2.5 MeV. The equation is:

Range (mg/cm^2) = 412 $E^{1.265 - 0.0954 \ln E}$

where E = E_{max} for the isotopic beta source, in units of MeV.

Linear Energy Transfer

The linear energy transfer, LET, is the final concept used to describe the interaction of a charged particle field with an absorber. The term is closely related to the stopping power, S. The chief difference is that the LET is concerned primarily with the energy left behind in the absorber while S focuses on the energy retained by the charged particle. The LET is defined as the average energy locally deposited in an absorber per unit distance of travel of a charged particle. As such, it is analogous to the energy absorption coefficient discussed earlier for gamma rays. It is important in radiation protection because, as will be seen in Chapter 5, the quality factor for a given radiation field is calculated from the LET. It can be measured in the same units as S, e.g., MeV/cm.

The concept of LET was introduced to allow a clearer description of particle energy vs. absorber energy in the theory of dosimetry. It will be shown in Chapter 5 that radiation dose is the energy deposited divided by the mass of the absorber. The LET is used to describe the energy deposition in an absorber. In practical radiation protection cases, the numerical value of the LET is virtually equal to S, the stopping power. LET includes all the energy lost by a particle <u>except</u> the energy released due to bremsstrahlung radiation. As mentioned above, bremsstrahlung typically only accounts for 1 or 2% of the energy lost by a charged particle in cases of interest in radiation protection technology.

Problem Set

1. Name a practical application of the sample reaction involving boron shown in Figure 1.

2. Why is elastic scattering the major contributor to dose when tissue is irradiated with fast neutrons?

3. What name is given to the neutron interaction which leaves an excited nucleus and reduces the kinetic energy of the particle?

4. How many "average" collisions with hydrogen nuclei would a 1 MeV neutron need to undergo before it becomes a "slow" neutron?

5. Define the term "mean free path" as applied to gamma rays.

6. The effective atomic numbers of tissue and air are both about 7.5. Compared to a gram of tissue, about how many interactions would occur in a gram of aluminum exposed to a beam of low energy photons? to medium energy photons? to high energy photons?

7. Explain why the Compton Scattering probability is Z independent per unit mass of absorber.

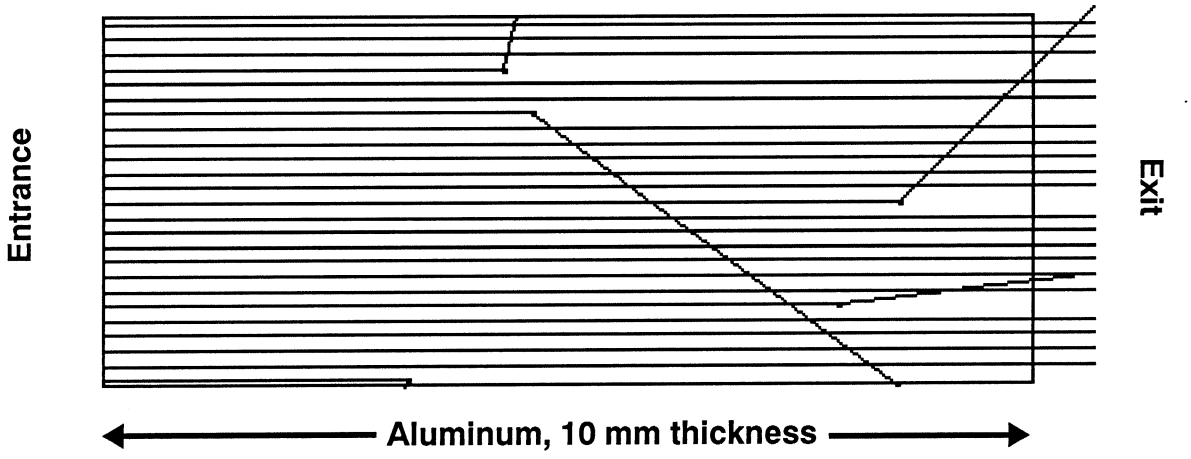
8. Assuming it is possible to create a proton anti-proton pair from a high energy photon, what would be the threshold energy required?

9. Calculate the wavelength of a 1 MeV gamma ray if the speed of light, c, is 3 X 10^{10} cm/sec.

10. Show that the two values (English system and metric system) for the Planck Constant in Figure 6 are equivalent.

11. What % of a photon beam would interact in passing through a thin 3 mm slice of absorber if the linear attenuation coefficient were 0.15/cm?

12. What energy photons are incident on the 10 mm thick aluminum slab shown below with their interactions?



13. Define the term "W value." What value does it have numerically?

14. Why is the range of a charged particle less than the quotient of the energy divided by the initial stopping power of the absorber?

15. Which properties of an absorber determine the value of the specific ionization?

16. Calculate the change in the amount of bremsstrahlung that would be produced if an aluminum target were substituted for an iron target.

17. Would the stopping power of a lead brick for 1 MeV particles be higher for alpha particles or "electron stripped" lithium ions with a +3 charge? Why?

18. Calculate the approximate range in air of the beta particles from P-32, Y-90 and C-14. Repeat the calculation for soft tissue. (Energies can be found in Appendix A-1).

19. Calculate the energy loss of a charged particle moving 2 cm through a counter gas with a specific ionization of 8,000 ion pairs/cm.

S-1. What energy does an epicadmium neutron have? What does the name refer to?

S-2. In Figure13 A) and 13 B), how closely does the computer simulation come to the theoretical value for the % attenuation? Why is it not perfectly correct?

S-3. Why is the W value significantly higher than the ionization potential for the atoms that make up air?

S-4. Calculate the thickness of aluminum (ρ = 2.7 g/cm^3) and acrylic plastic (ρ = 1.18 g/cm^3) needed to stop beta rays from ^{32}P.

Other Resources

1. "Nuclear Radiation Physics," Lapp and Andrews, Prentice- Hall, Inc., Englewood Cliffs, NJ, 1972 (Out of print but available at some libraries).

2. "Modern Physics" by Kenneth S. Krane, John Wiley & Sons., Inc., New York, 1995.

3. "Quantum Physics: Of Atoms, Molecules, Solids, Nuclei and Particles" by Robert Eisberg and Robert Resnick, John Wiley & Sons., Inc., New York, 1985.

4. "Radiological Physicists," Juan del Regato, American Institute of Physics, New York, 1985. (Contains detailed biographies of many historically important physicists mentioned in this text such as Roentgen, Curie, Rutherford, Compton and Fermi.)

4

Biological Effects of Radiation

Outline of this Chapter

Chapter Summary

With an understanding of how different radiations interact with matter, we turn now to the effects of these interactions when the "matter" is living tissue. The result of ionizing radiation striking water is the release of free radicals. If not removed from the system, by competing reactions, i.e., with chemical protective agents, hydroxyl radicals can combine to produce poisonous hydrogen peroxide or can denature critical molecules in the cell.

In the dose range of a few Gy to 10s of Gy, fatal damage can be inflicted on major organelles within cells. In particular, cells which are immature and undergoing rapid cell division are unusually sensitive to radiation injury. The fact that radiation affects living systems can in itself be useful in allowing scientists to determine radiation doses through biological measurement of tissues. This is the rapidly developing area of biological and physical biodosimetry, The relative biological effectiveness, caused by the large differences in LET among radiation types, means that different types of radiation produce different amounts of damage when depositing identical amounts of energy.

In humans, the blood system is one of the most sensitive to radiation. Dramatic drops in white cell count follow doses of only a few Sv. In the 10 to 50 Sv range, major damage occurs to the crypt cells buried in the small intestine wall, leading to death of this organ and the individual. Higher doses will interfere with the trigger signals from the lower brain stem to heart and diaphragm muscles. This is the Cerebrovascular Syndrome characterized by irregular heartbeat and breathing followed by coma and death.

If a survivable radiation dose is received, late effects are still possible months to years after exposure. Because radiation can alter DNA structure, the genetic code of germ cells can be affected leading to mutations in offspring. It requires about 1 Sv of dose to a population to double the natural, spontaneous mutation rate. Late effects are also possible in the exposed person. Radiation is a carcinogen. At sufficiently high doses, a number of cancer types occur more often than expected. An acute dose of 0.1 Sv (10 rem) will increase overall cancer mortality by about 3%. At low doses, radiation hormesis predicts beneficial effects on health, including increased life span and lower cancer risk. Human population studies are lending increasing support to this radical idea.

Treatment of the radiation accident victim depends on whether the dose was internal or external. For internally deposited radionuclides, several techniques have been successfully used to increase the elimination of the activity or prevent organ deposition. For external doses, treatment is focused on restoring the blood count to normal levels.

Radiation Effects on Water

Primary Reactions (Direct Action)

Water is basic to all living tissues. This chapter will begin with a discussion of radiation effects on water. In fact, water makes up 80% of all soft tissues in the human body. Thus, the chemical reactions that take place following the irradiation of pure water will be repeated when that water is incorporated as a part of a cell.

There are two categories of reactions which take place in pure water. In older texts, these were known as "direct action" and "indirect action." It is now clear that these terms implied more about the mechanisms than is justified by our current knowledge, so the terms primary reactions and secondary reactions have been introduced to replace the older terms.

Before describing the reactions, it is interesting to note that the primary and secondary reaction details have only been fully understood after extensive research. This is due to the extremely short time span over which the events occur. By combining the technologies of electron paramagnetic resonance and "flash" x-ray tubes (devices that can produce nanosecond bursts of high intensity x-radiation), the various chemical species produced in the reactions have finally been sorted out.

The primary reactions are responsible for much of the biological damage caused by high LET radiations. The direct passage of the "ray" will eject atomic electrons from the absorbing tissue which can ionize or otherwise alter key molecules in the tissue. The overall time span during which they are taking place is extremely short, only about 10^{-10} seconds. That is only a tenth of a nanosecond or one ten thousandth of a microsecond! During this interval after the passage of the "ray," a series of about eight different reactions occurs. As a result, two new chemical species are introduced in relatively large numbers. Chemists call them free radicals. A free radical is highly reactive, chemically. This is due to the presence of an unpaired electron in the free radical. In forming chemical compounds, electrons tend to "pair up" in such a way that an atomic suborbital shell is filled. One electron has its spin vector pointed up and the other has the spin vector pointing down. Thus, a free radical tries to combine chemically with other species so that its single unpaired electron can form a covalent bond with some other unpaired electron to complete the sub-shell. Note that a free radical is electrically neutral. It has an equal number of protons in the nucleus to balance the negative electrons. It is definitely not an ion.

The first of the two free radicals formed in irradiated water is a hydrogen radical. This merely consists of a proton and an electron, i.e., a hydrogen atom. Note that this is not what you would get in a tank of hydrogen gas. Hydrogen is a diatomic gas composed of molecules of H_2, formed by pairs of free radicals each of which can then pair its electron with the electron from the other. The second free radical formed in significant numbers in irradiated water is the hydroxyl radical, OH. Chemists often write the chemical symbol for a free radical with a small dot over the letters. This dot can be thought of as calling attention to the presence of the unpaired electron.

Secondary Reactions (Indirect Action)

The secondary reactions occur during the next 10^{-5} seconds after passage of the "ray" through the water. They are much less complex than the primary reactions, and are the main source of damage for low LET radiations. Only three secondary reactions occur with high probability. They are, simply, the three possible combinations of the two free radicals produced in the primary reactions; hydrogen + hydrogen, hydrogen + hydroxyl and hydroxyl + hydroxyl. These are shown in Figure 1.

$$\dot{H} + \dot{H} \rightarrow H_2 \text{ (gas)} \quad (1)$$

$$\dot{H} + \dot{O}H \rightarrow H_2O \text{ (water)} \quad (2)$$

$$\dot{O}H + \dot{O}H \rightarrow H_2O_2 \text{ (hydrogen peroxide)} \quad (3)$$

Fig. 1 - Secondary reactions in pure water

The first reaction leads to the formation of a molecule of hydrogen gas. Since a living cell has a considerable number of dissolved gas molecules normally present, this extra hydrogen is not a problem (as long as no one strikes a match down in the cell!!). The second reaction likewise produces a harmless product, water. The problem in a living biological system is the occurrence of the third reaction leading to hydrogen peroxide, a poison to the cells. In humans, about 2/3 of the injury produced by low LET radiation exposure to cellular DNA is traceable to the hydroxyl radical.

In addition, one of the H atoms of the hydrogen peroxide can be dropped readily to form a peroxide radical which then attacks other bio-organic molecules to form relatively stable organic peroxides. This process might lead to events which will prove fatal to the cell. For example, the organic peroxide formed by the peroxide radical attack might have formerly been some absolutely essential molecule such as a key enzyme. Its conversion to the peroxide form of that enzyme denatures the molecule and effectively removes it from the cell. When it is needed at some critical phase later in the cell cycle it will not be found. This represents an irreparable injury to the cell. If this third secondary reaction could somehow be eliminated, radiation would produce considerably less injury to living tissues. In fact, this is possible to some extent.

Radioprotectants

As just mentioned, much of the injury to living tissues is due to the formation of peroxides from the combining of hydroxyl radicals released by the radiation. If the hydroxyl free radicals could instead be encouraged to combine with a hydrogen free radical (reaction 2, Fig. 1), much less injury would be produced. This can be accomplished by chemically "flooding" the tissue with excess hydrogen free radicals or with compounds that readily donate hydrogen atoms or -SH groups. This is the subject of radioprotective chemicals, or the preferred term, radioprotectants. Although military interest in developing radioprotectants continued after the cold war, the possibility of nuclear terrorism on American soil caused an immediate upsurge in public interest regarding developing these pharmaceutical products. In response to this new culture,

the U.S. Food & Drug Administration (FDA) has recently established a new, fast track, protocol for accepting drugs used in nuclear or biochemical terrorism situations. Such drugs can be approved for human use without the normal need to show their value in human studies - animal studies will be sufficient.

In general, pharmaceuticals useful for treating radiation accident victims divide into two types - treatment for external radiation exposure and for internally deposited radioactive material. There are three different uses of drugs to treat external radiation exposure. The first use is the category of pre-irradiation protection - an attempt to reduce the amount of damage at the time of exposure. A pre-irradiation chemical protective agent has to be administered well ahead of the radiation exposure to allow time to disperse uniformly throughout the tissues at the cellular level. Recall that the secondary reactions are complete in only 10 microseconds following the energy deposition.

The second use of drugs to reduce external radiation injury involves repair of some of the damage at the molecular level in cells. These pharmaceuticals can be administered after radiation exposure. The third category of drugs useful for external radiation exposure attempts to "jump-start" the body into producing new stem cells or progenitor cells which are slightly differentiated and can mature into one of many different cell lines as needed. These compounds, again, are effective when administered post-irradiation.

In cases of internal uptake of radioactive substances, a number of approaches have been developed over the years. These are covered briefly at the end of this Chapter and in detail in Chapter 14, **Handling Nuclear Emergencies**. The pre-irradiation drugs for external irradiation (the first of the three categories) will now be discussed here. The second and third categories of post-irradiation treatments will be deferred until Chapter 14.

A number of pre-irradiation protectants have been extensively investigated over the years, many of them in connection with research done for the military services at Walter Reed Army Hospital. Historically, organic compounds such as cystene have been found which donate their hydrogens to neutralize hydroxyl free radicals. The usefulness of a protective agent is sometimes measured by the "dose reduction factor" (DRF), that is, the change that can be produced in the normal lethal dose ($LD_{50/30}$) for a test animal. The $LD_{50/30}$ is the value of the dose delivered to a group of animals such that 50% of them will survive for 30 days without treatment. Note that when discussing lethality, the assessment time for 50% survival is different between animals (where the time period is 30 days) and humans (where the time period is 60 days). This is due to the observed fact that acute deaths extend out to only 30 days in animal experiments but human deaths are complete only after 60 days. The results of a review of human data published in May, 2003 concluded that the best estimate of $LD_{50/60}$ in humans was 410 rads ± 150 rads (to 95% confidence), again, without medical treatment.

Cystene has been found to raise the $LD_{50/60}$ in humans by a factor of up to 1.7 times. Possibly the best compound found by the Walter Reed scientists is called Amifostine or WR-2721. The DRF is 2.7 for $LD_{50/30}$ in mice. This chemical is a form of cysteamine. When taken into the body, it enters the cells and goes after free radicals. As of 2004, Amifostine has been approved by the FDA for intravenous use during radiation treatment for certain cancers. It is not available over-the-counter to the

public as a “dirty bomb” pill. Unfortunately, these chemical protective agents are toxic for humans at concentrations needed for maximum protection. For example, Amifostine seriously lowers blood pressure and cystene causes severe nausea and vomiting. Further research may someday develop safer radioprotectants.

It is interesting to note that American astronauts carried radioprotective chemicals on the lunar space missions in case of solar flare activity and Russian field troops apparently carried them during the “Cold War” to reduce the biological effects of tactical nuclear weapons.

Radiation Effects on Cells

Cell Structure

Figure 2 illustrates some of the components that are normally part of typical animal cells. The diameters of cells are usually in the 5 to 50 micrometer (micron) size range. The whole cell is enveloped within the plasma membrane. It acts analogously

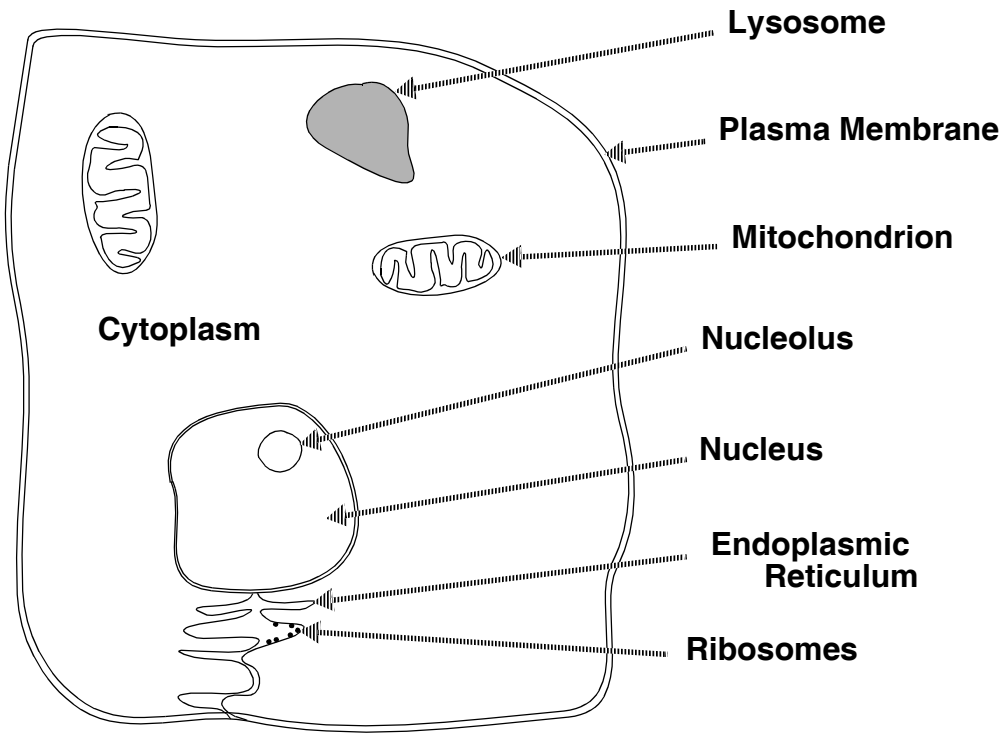


Fig. 2 - Structure of a typical animal cell

to the skin of a person. According to the fluid mosaic model, the membrane is composed of two layers of phospholipids, a phosphate group attached to fatty molecules (lipids) with chunks of proteins embedded like a mosaic. Some of the proteins extend all the way through the two-layer structure and act as two-way channels. The plasma

membrane separates the outside, extracellular material from the inside of the cell. By the complicated mechanism of active transport, the plasma membrane is able to bring outside ions and nutrients into the cytoplasm through the protein channels. The cell removes unwanted ions and waste products to the outside of the cell by passive transport.

The cell nucleus is analogous to the brain - it organizes the cell and controls the complex process of mitosis or cell division. The overall series of events leading to mitosis is called the cell cycle. It begins with a resting gap period, named "G_1" by biologists, that follows the previous division. Next, there is a period of up to 15 hours, called "S" for the synthesis phase, while the cell duplicates its DNA. Following another resting gap, "G_2," the cycle ends with the "M" phase (about 1 hour long) for mitosis.

The nucleus is the largest organelle within the cell and is itself contained within a double nuclear membrane. A smaller ball–shaped nucleolus can be seen inside the nucleus. It is the place where RNA is manufactured. Both DNA and RNA genetic material are housed inside the nucleus. During most of the cell cycle, fine threads are visible throughout the nuclear volume. At the time of cell division, they pull together and form into thick chromosomes holding the genes for the organism.

Forming a connecting pathway from the nucleus to the cell plasma membrane, the endoplasmic reticulum transports materials around through the cytoplasm and to the cell membrane. It also is the attachment site for many of the ribosomes. These tiny entities, only 0.025 microns in diameter, produce proteins, e.g., hormones, that are transported out of the cell and used by other tissues.

The mitochondria are small structures (a few microns long) which float around in the cytoplasm. They are the source of energy for cellular functions. This is accomplished via the compound adenosine triphosphate (ATP). The ATP is assembled within the mitochondrion through a series of oxidation and phosphorylation steps within the "compartments" shown in the figure. This just means that three phosphate groups are joined together with the adenosine. This process requires energy, i.e., it is an endothermic reaction. At a later point when the cell needs energy for some function, it disassembles the ATP (an exothermic reaction this time) and releases the energy. This process is somewhat analogous to the storage of electrical energy in an automobile battery while driving to work. After work, the reaction is reversed and electrical energy is drawn out of the battery to start the car.

The last structure to be considered is a lysosome. It acts as a recycling center in the cell. It consists of a membrane surrounding a digestive enzyme. Worn-out structures within the cell are passed through the membrane, dissolved into more basic components (e.g., amino acids) and then released into the cell cytoplasm.

Radiation Effects

Consider, next, the effects of ionizing radiation on some of the cell structures. The dose ranges given below apply to human cells. Beginning with the plasma membrane, it has been established that it takes about 3,000 to 5,000 rads of absorbed dose to rupture this structure. This represents a major injury to the cell. It allows the extracellular fluids to flood into the cell, and allows leakage out of ions and nutrients which the cell had brought inside with a considerable expenditure of energy. In some

cases, membrane rupture will lead to death of the cell. If this occurs, death will be analogous to drowning. At doses less than those needed to produce rupture, cells show an increase in permeability and some leakage of contents occurs.

It takes a few thousand rads to disrupt a mitochondrion. In this dose range, they appear swollen in size when viewed with a microscope. The internal "partitions" also appear to have broken off and are floating randomly around inside the structure. This means an immediate interruption in food production (ATP) for the cell. If the cell has adequate reserves of ATP it can repair the damage to the mitochondria and they resume ATP production. In general, the larger the radiation dose, the longer the delay period before ATP manufacturing resumes. If the cell does not have sufficient energy reserves, then that could lead to death by starvation.

The lysosomes are typically ruptured at a dose between 500 and 1,000 rads. This will release the digestive enzymes contained within the lysosome membrane. These enzymes will then attack various healthy structures within the cell and begin digesting them. Clearly, this could lead to cell death in a process analogous to suicide. Biologists call this process autolysis. It might be noted that a much higher radiation dose is needed to inactivate the actual digestive enzymes, so they are released full strength when the lysosome ruptures.

The cell nucleus, the repository of the cell's DNA, is the most radiosensitive structure in the whole cell. This has been repeatedly demonstrated in experiments where micro beams of radiation are focused on different cell structures. Those cells which have had the nucleus irradiated always show a lower survival rate than cells which have had other structures selectively irradiated. The major effects of radiation on the nucleus depend on the phase of the cell cycle when the radiation dose is received. For radiation delivered early in the cycle, before the DNA replicates, effects include the inhibition of DNA and RNA production. This means that the cell cannot prepare for cell division. Before dividing, the cell manufactures a complete duplicate set of chromosomes which carry all the information needed to produce fully functional daughter cells. Without DNA, duplicate chromosomes can't be made. If this process is delayed long enough the cell dies, analogous to death in childbirth. At low doses the DNA production is delayed only a short period. As the dose is increased, the delay period gets longer until death results. Due to the wide range of sensitivity for human cell nuclei, there is no typical dose range for this mechanism.

For radiation delivered any time during the cell cycle, the production of double-strand breaks in DNA is a major factor in cell survival. The double-strand breaks usually lead to aberrations which are lethal to the cell. These are discussed in detail below in the next section.

Human cells normally have a set of 46 chromosomes (a fact which was not known with certainty before 1956) which carry all the genetic code information for an individual. Each of our cells has around 6 billion pairs of the basic DNA coding molecules (nucleic acid bases, for the chemists in the audience). A single code entry consists of a triplet – a consecutive set of three base pairs which identifies one of the 20 necessary amino acids. The triplets are gathered together to form a gene. With publication of the human genome in 2001, and additional research, we now know that each human cell contains about 20,000 genes that carry all of our genetic information. Genes group together in lines to make up a chromosome. It was finally shown in 1997 that each chromosome contains a single DNA molecule. If fully unraveled and stretched out in a line, the human DNA molecule is 1 meter long!

A question sometimes asked is, “What actually kills the irradiated cells?” As was implied by the above information, no single overriding mechanism can be blamed for cell death. The combination of leaky plasma and lysosome membranes coupled with the introduction of lethal chromosome aberrations and delays in ATP, DNA and RNA production taken together is just too much injury for a cell to survive. See Sample Problem 1.

Sample Problem 1

GIVEN:

A collection of human cells is treated with 5 Sv of gamma rays.

FIND:

What types of injury would be expected on the cellular level?

SOLUTION:

A 5 Sv dose is too low to severely affect the plasma membrane or the mitochondria. Thus, the injury expected would be rupture of the lysosomes to release active digestive enzymes, introduction of lethal chromosome aberrations and a halt in DNA and RNA production for an extended time period. A number of the cells probably would not survive this damage.

Human Biodosimetry Techniques

The term biodosimetry refers to methods in which changes caused by exposure to ionizing radiation are directly measured in a living system to determine the radiation dose received. Under ideal conditions, the body maintains a dose history that we can access. When these techniques are fully developed in some future generation, it may be possible to eliminate radiation badges on workers. Dose records might be updated periodically by submitting a biological sample, perhaps a milliliter of blood.

There are two different types of methods used in biodosimetry. Biological based techniques use detection of biological tissue damage, usually at the cellular level. The second approach involves physical measurements of changes induced by the radiation in body tissues. In general, the biological measurements are more sensitive. They can detect lower radiation doses than the physical measurements. Some disadvantages to the biological methods include problems caused by the realization that most of the biological damage is not completely radiation specific. For example, the observed damage may have been caused by exposure to chemicals or by trauma or stress.

The physical measurement approaches have the advantage that they can be performed at times well after the exposure incident. The downside to this is that it will be unknown whether the dose measured was received at a particular time or is the result of cumulative exposures over a lifetime. Physical measurements, in contrast to biological, are much more radiation specific and can differentiate radiation doses to specific body areas, e.g., a particular arm or leg. Results are also found to not vary as much between individuals. A discussion now follows of some biological techniques. This will be followed by a look at the physical techniques.

As of 2011, several useful human biological biodosimetry methods are available. The most mature of these fall in the field of cytogenetics – the study of the structure and functioning of cells, especially the chromosomes from cell nuclei.

The oldest cytogenetic technique is chromosome aberration dosimetry in which cells from the radiation-exposed individual are counted to measure the number of aberrations present. This technique is quantitative and accepted by the legal system but is incapable of giving much information about exposures well in the past or for exposures delivered chronically over many months. It has proven very useful in a number of acute accident cases, e.g., Chernobyl, Goiania and Tokaimura. All of these serious radiation accidents are discussed at length in Chapter 14.

Chromosome aberrations are one of the clearest evidences of cell damage following irradiation in the sense that the damage is actually visible in a microscope. If the cells are appropriately stained when they are undergoing mitosis, it is possible to distinguish the individual chromosomes as diffuse 4 arm structures. Figure 3 indicates the appearance of a typical chromosome (lower right-hand corner) and its microscopic composition as presently understood. The exact nature of chromosomes is still under study. They are composed of approximately 25% DNA with the remainder being protein and fats. The structure of the DNA molecule is also indicated in Figure 3. It has an appearance similar to a ladder which has been twisted into a helix. The nucleic acid bases making up the genetic code form the "rungs" of the ladder and the "side rails" are long chains of sugar phosphate groups called strands. The genetic code information is duplicated in that a 'mirror image' of the code sequence runs along each strand of the DNA ladder.

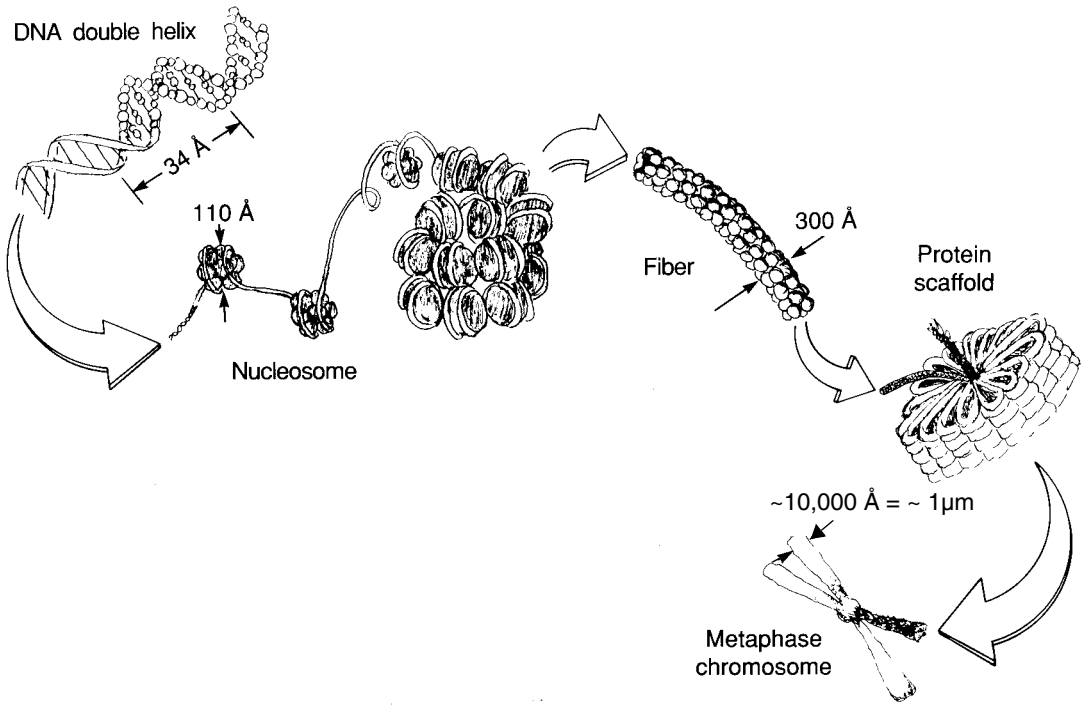


Fig. 3 - A typical chromosome

An aberration is any change from the normal shape of the chromosome. It takes about 100 rads to produce an average of 1 aberration per cell. In the typical mammalian cell, this 100 rad dose will cause 1,000 single and 30 double strand breaks and damage 1,000 to 2,000 nucleic acid bases. Certain types of aberrations appear to be favored when radiation is the causative agent. Recall that many other agents (e.g., chemicals and heat) can cause chromosome aberrations. Three of the more common radiation–caused aberrations are shown in Figure 4. Both rings and

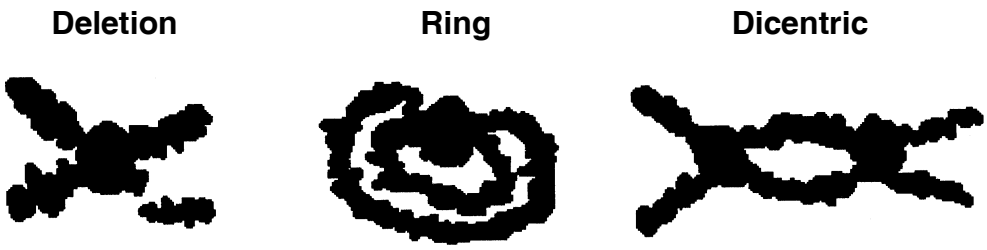


Fig. 4 - Radiation-caused aberrations

dicentrics are lethal to the cell. A deletion can lead to cancer if the "lost" section of the chromosome contains a suppressor gene. All three of these aberrations result from double-strand breaks in DNA.

Dicentrics, in particular, appear to be caused only by radiation or exposure to the chemical benzene. Careful measurements of the natural frequency of occurrence and the increases in frequency with radiation dose of the aberrations shown in Figure 4 show remarkable agreement in the data among the world's laboratories. This has resulted in a very useful practical application in radiation protection – dicentric chromosome aberration assay or DCA assay. When the assay is done using human lymphocytes (white blood cells) it is considered the "gold standard" for dosimetry purposes. The International Atomic Energy Agency has produced a procedures manual for the technique (see Other Resources at the end of this Chapter) and the International Standards Organization also issued a performance criteria standard in 2008. An intercomparison of several international labs showed their calibration graphs agreed to better than 99% confidence and that reported doses were accurate to ±15%. The range of doses measurable by DCA assay is approximately 5 to 500 rads.

In the event of an actual or alleged radiation exposure a blood sample can be examined by a cytogenetics laboratory specializing in this work and a rather precise estimate of dose can be made. The useful range of the technique depends on the number of cells counted. The Clinical Cytogenetics Laboratory at the University of Pittsburgh quotes a minimum sensitivity of 20 rem whole body for 200 cells counted and 10 rem for 500 cells counted. The upper dose limit of the test is about 500 rem. Low LET radiations, such as x and gamma rays, produce a linear-quadratic dose response in human white blood cells. High LET radiations produce a strict linear response with dose.

Extreme care must be exercised in sample collection and handling to get meaningful results. A highly skilled laboratory technician requires the expenditure of 2 working days of time to analyze 100 cells from an accident victim. Typically this

would give a result with 60% relative error. To reduce this to 20% relative error would require 3 full weeks of labor. The number of cells showing aberrations gradually decreases with time after an acute radiation dose. An indication of the time span is obtained by reference to the survivors of the Japanese nuclear detonations. Elevated chromosome aberration frequencies (for aberrations not lethal to the cell) were still observable after 40 years. A calculational example is given in Sample Problem 2.

Finally, chromosome aberration measurements give an indication of just how extensive biological repair really is. The energy required to produce breaks and other damage in DNA molecules potentially leading to an aberration has been measured. As mentioned above, a radiation dose of 100 rads will cause 1,000 single strand breaks, 30 double strand breaks and over 1,000 alterations to the nucleic acid bases. Thus, a typical cell can sustain over 2,000 individual pieces of damage, each one potentially an aberration, before an actual aberration appears. The cell has repaired all the other injury during the cell cycle. That represents a tremendous level of repair!

The repair mechanism for DNA was first discovered in bacteria. It has now been demonstrated to also occur in human cells. It is a four-step process. First, the damaged single DNA strand is cut above and below the damaged area. Next, this section is removed. Thirdly, a newly grown section of DNA with the genetic code taken from the opposite, normal strand, is moved into place. Lastly, this new section is reattached to the DNA molecule, completing the repair. Since the pattern for the replaced section of DNA is taken from the intact strand, the whole process leads to a very accurate repair of the damage, usually without alteration of the genetic code.

Recent research has shown the importance of "checkpoint genes" in the repair process. In particular, gene p53 acts as a gatekeeper during the first resting phase, G_1, of the cell cycle. The cycle is put on hold by p53 if it detects DNA damage and only allowed to continue after repair has been completed. If p53 is mutated, cancer risk increases.

Note that current radiation protection standards neglect biological repair (assume that there isn't any) so as to provide a "safety factor." Note also that considerable damage may be happening to the genetic code but it is not observable in a microscope. The above remarks were directed at chromosome aberrations only.

Sample Problem 2

GIVEN:

Measurements at several laboratories show the fraction of cells with dicentrics is given by $F = 0.001 + 6 \times 10^{-6} D^2 (rem^2)$.

FIND:

What radiation dose would be estimated for a worker with 25 dicentric cells out of 100 analyzed?

SOLUTION:

25/100 is a fraction, F, of 0.25. From the calibration formula,

$0.25 = 0.001 + 6 \times 10^{-6} D^2 (rem^2)$. So, $6 \times 10^{-6} D^2 (rem^2) = 0.249$.

Thus, $D^2 (rem^2) = 0.249/6 \times 10^{-6} = 4.15 \times 10^4$. Taking the square root of both sides gives D (rem) = 200 rem. (Only 1 significant figure allowed)

A competing test, also in the area of biological biodosimetry methods, is micronucleus assay. If the nuclei of cells are damaged by ionizing radiation and the cells divide, sometimes one of the daughter cells contains a small nucleus (the micronucleus, MN) along with a normal nucleus. The percentage of cells containing a MN is a measure of radiation dose. One advantage over the dicentric chromosome assay is that it doesn't require a highly trained cytogeneticist to perform the cell scans. The assay can be automated using commercially available flow cytometry equipment. With automation, large numbers of cells can be scanned rapidly, improving the sensitivity. Doses as low as 5 rads have been reported to produce a statistically significant increase of the fraction of cells containing a micronucleus.

Turning our attention now to physical methods of biodosimetry, the two most commonly analyzed tissues are teeth and fingernails. This involves the use of electron paramagnetic resonance, EPR. (When originally developed in the 1940s, the preferred term was electron spin resonance or ESR.) EPR is very similar to nuclear magnetic resonance, NMR, the physical technique behind MRI medical imaging. The difference is that EPR examines spinning atomic electrons in a target material whereas NMR (and MRI) examines spinning atomic nuclei. A radiation dose estimate is obtained by analyzing teeth from an exposed individual. A Russian research lab has done dosimetry on over 200 workers from the Chernobyl accident cleanup crew. Teeth were provided by dentists when removed from a worker for dental health reasons. (The workers were not asked to volunteer their healthy teeth!) The method has a claimed range of about 0.05 Sv to 1 Sv (5 - 100 rem). Figure 5 shows the Russian spectrometer setup for Chernobyl workers. The equipment was made in Germany and donated through a World Health Organization grant after the Chernobyl accident.

Since 2005, studies have been undertaken to develop a technique which can be applied to teeth *in vivo*, i.e., without first yanking them out of your mouth. It was felt

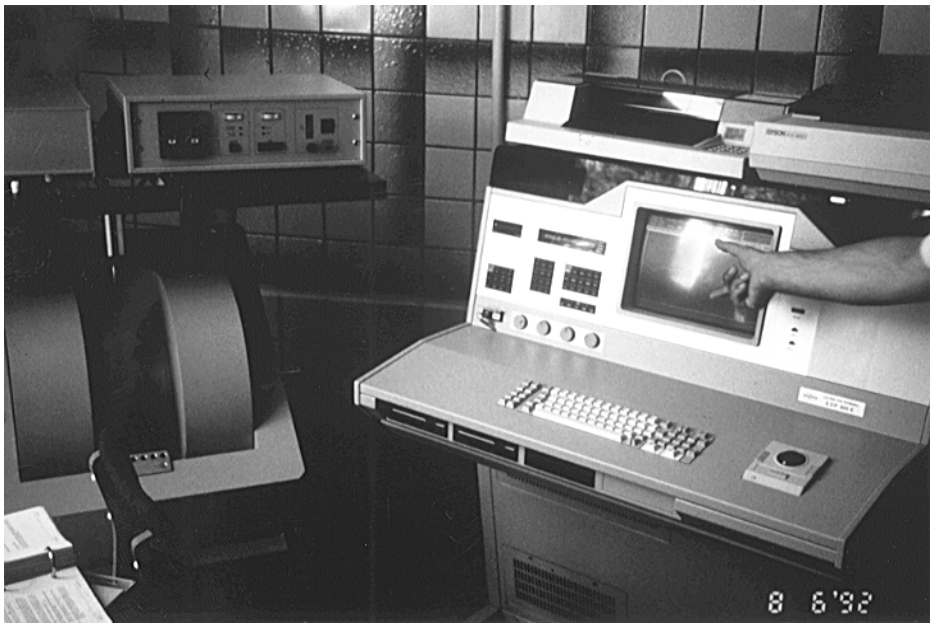


Fig. 5 - Russian ESR spectrometer lab for dosimetry of Chernobyl workers

Courtesy, Donald Philpotts

that using EPR techniques to perform triage (screening) following a mass casualty radiation event would be more acceptable to the public if the tests could be done *in vivo*! Currently (2011), the detection sensitivity is around 100 rem. Efforts to improve the sensitivity are underway.

Historically, EPR studies focused on teeth. More recently it was found that fingernails also produce an EPR signal following radiation exposure. Initially, results were very confusing and inconsistent. The problem was eventually traced to the fact that cutting off the fingernail sample induced mechanical stresses that also produced an EPR signal that competed with the radiation induced signal.

Finally, scientists with the French Atomic Energy Commission have demonstrated a physical biodosimetry method to measure fast neutron doses to humans. The technique detects the presence of phosphorus-32 as an activation product from natural sulphur in human hair. An accuracy of ±10 to 20% was reported over the dose range of 5 to 1500 rad. Only a gram or so of hair sample is needed.

Cell Radiosensitivity Theories

Ideally we would like to have a law analogous to Newton's Laws of Motion or Ohm's Law which would predict exactly the kind and amount of injury produced in a given cell following an exposure to radiation. Unfortunately, such a law remains yet to be discovered. The field of radiobiology is thus dependent on "rules of thumb" to estimate radiation effects. Simplistically, this is the underlying reason for the adoption of the ALARA philosophy. ALARA is the acronym for "As Low As Reasonably Achievable." This attitude is applied to radiation doses received by nuclear technicians because we have no accurate theory to predict what really happens at the low dose rates of 0.01 to 0.05 Sv (1 to 5 rem) per year encountered occupationally.

The oldest, and perhaps best, rule of thumb was developed by two French radiobiologists, Bergonie and Tribondeau, in 1906. It offers a prediction about the relative sensitivity of two different types of cells or tissues to radiation. The so-called Law of Bergonie and Tribondeau concluded that cells tend to be radiosensitive if they have three properties:

Cells have a high division rate.

Cells have a long dividing future.

Cells are of an unspecialized type.

The first condition can be determined by measuring the cell cycle time, i.e., the time between divisions. The second property refers to the fact that many cells go through phases in an overall life cycle. They begin by undergoing many cell divisions (childhood). They then enter a phase in which they stop active division and instead put together the internal structures to function in some usable capacity (adolescence). Finally, they enter the last phase where they function fully in the job assigned

(adulthood). Cells with a long dividing future would be those in the early immature phases where they are still dividing.

The last criterion, unspecialized, needs further comment. In the biological sense, this means a cell which is capable of specialization, at some future time, into one of several different "adult" cell types. An example might be one of the immature blood cells. Many types of blood cells are "born" unspecialized. Depending on the feedback signal received long after they are formed, they can choose to "grow up" and mature into lymphocytes, or different types of granulocytes. Probably the most unspecialized human cell is a fertilized ovum. From this single cell, daughter cells develop into such widely different mature cells as bone, brain, blood, and fingernails.

The generalization of the Law of Bergonie and Tribondeau is that tissues which are young and rapidly growing are most likely radiosensitive. A very practical application of the Law is given by NRC Regulatory Guide 8.13 which is titled **Instruction Concerning Prenatal Radiation Exposure**. This Guide requires that women of reproductive age be informed of the increased risk of injury of the human fetus from radiation exposure because such a tissue meets all the criteria of the Law of Bergonie and Tribondeau. The human fetus is particularly sensitive in the first few weeks of pregnancy when organs are forming. This is also a time period when the woman may not be aware of her pregnancy. Most radiation protection standards, including the 1991 U.S. Code of Federal Regulations Title 10, Part 20.1208, recommend that the dose to a developing embryo and fetus be kept below 0.5 rem during the entire nine months of gestation.

Relative Biological Effectiveness

Before leaving the cellular level and moving up to human organ systems, it should be noted that different types of radiation affect cells differently under the same irradiation conditions. This was first observed in the 1950s when radiobiologists began using neutron beams to irradiate various organisms. It soon was apparent that a rad of neutrons often produced greater injury than a rad of x-rays. This effect was eventually defined in terms of the Relative Biological Effectiveness or RBE, Figure 6.

$$\text{RBE} = \frac{\text{(Dose of 250 kVp x-rays to produce effect)}}{\text{(Dose of other radiation for same effect)}}$$

Fig. 6 - Relative biological effectiveness defined

Notice that the RBE compares the doses needed to produce the same biological effect rather than comparing effects at the same dose. The "reference radiation," 250 kVp x-rays, was chosen for historical reasons - most of the early radiobiology was done with that energy of x-radiation.

The term kVp stands for "peak kilovoltage." It refers to the fact that in most x-ray machines the potential difference applied across the tube is a sine wave. The "peak" value then is the maximum voltage, above ground potential (defined to be zero volts) reached by the applied voltage.

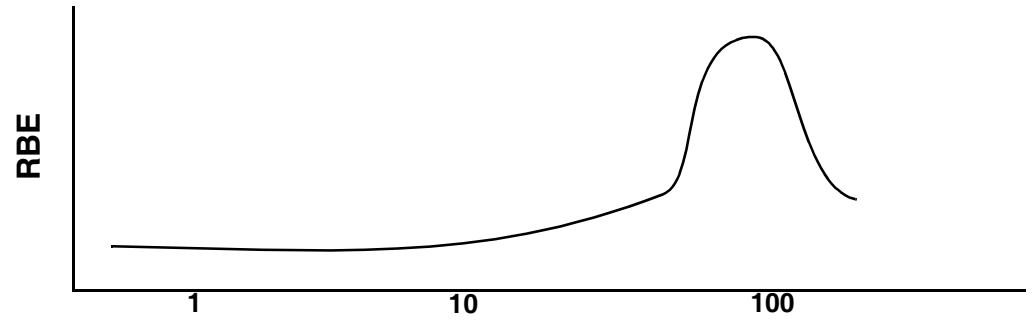


Fig. 7 - RBE versus the Linear Energy Transfer, LET

The reason that different radiations have different effects has been traced to the way the various radiations distribute their energy as they move through tissues. This is best described by the linear energy transfer, LET, discussed in Chapter 3. The rate at which energy is deposited per unit distance of travel determines how effective a given radiation is at injuring cells. The RBE slowly increases with increasing LET and then goes through a peak at an LET value of about 100 keV/micron for human tissues. (See Figure 7) Because of this behavior, the unit of absorbed dose, the rad, is not an acceptable measurement of radiation injury. One rad of fast neutrons produces more damage in humans than a rad of x-rays or beta rays. Thus, FOR RADIATION PROTECTION PURPOSES, it is necessary to invent a new quantity and units that will be directly proportional to radiation injury in people. This is done through the dose equivalent measured in sieverts (Sv) or rems and by use of the quality factor which takes into account the differing RBEs of different radiations. This concept is explored further in Chapter 5.

The RBE is now known to vary with dose, dose rate and with the choice of the "effect" being sought. RBE appears to become larger as either the dose or dose rate are reduced. This has some serious implications for low dose chronic exposures which characterize the radiation workplace. Sample Problem 3 shows a RBE example calculation.

Sample Problem 3

GIVEN:

A laboratory finds that plants must be exposed to 25 Sv of 250 kVp x-rays to kill them. The same effect is obtained when these plants are exposed with 23 Sv of P-32 beta rays.

FIND:

What RBE does P-32 exhibit for killing this plant species?

SOLUTION:

The RBE is merely the ratio of doses for the same effect. In this example, the RBE would be 25 Sv / 23 Sv = 1.1 for killing this species.

Another related development was the confusion over RBE human data for neutron exposures. For many years, neutron RBE values had been widely accepted based on the Japanese A-bomb survivors. They were calculated based on the assumption of a large neutron component of total dose in Hiroshima compared to a negligible neutron dose in Nagasaki. This was explained by the differences in construction of the two devices (See Chapter 6). In 1986, all this work was thrown out when it was learned that large errors had been made in the original tentative dosimetry of 1965. It was felt that both cities had about the same neutron component. A 1992 study of thermal neutron activation products produced more than one km from the bomb epicenter shows that the doses were actually 2 to 10 times higher than calculated by the 1986 dosimetry model for low energy neutrons. In April, 2003, the "final" study on the neutron and gamma ray doses received by the Japanese survivors was announced. It was called "DS02" for dosimetry study 2002. In the 15 years since the 1986 dosimetry, new calculations in the bomb yields and improved three-dimensional modeling of the radiations as they moved through the air enabled the confusion about the neutron doses to be resolved. More details on this subject will be given later in this Chapter.

The yield of the Hiroshima bomb is now placed at 16 kilo tons of TNT. The earlier dosimetry had been using 15 kT.

A final development is the realization that RBE varies significantly for low LET radiations of different type. Low energy x-rays are now known to have an RBE 200% to 300% higher than gamma ray emitters such as cobalt-60. This is the reason that the U.S. Nuclear Regulatory Commission settled on a quality factor for fast neutrons that is only half the value recommended by other organizations. This is covered in more detail in Chapter 5.

Finally, some closing words about recent research in predicting radiation risk. Three problem areas have emerged that call into question some basic assumptions of radiobiology theories. First, cells which are <u>not</u> in the direct path of a radiation track through tissue are called bystanders. It has now been shown that they can have their DNA mutated even though no energy was deposited directly in them. Next, it turns out that mutations and chromosome aberrations can suddenly appear many cell cycles AFTER the radiation dose was received. This is called genomic instability by biologists. Finally, the concept of "adaptive response" is being observed. When a low dose of radiation is delivered before a follow-up high dose, the damage from the high dose is less than would normally be seen. Clearly there still remain many challenges to be understood in the field of radiobiology.

Radiation Effects On Human Organs

Blood System

The four major types of cells in circulating blood are the erythrocytes (red blood cells), lymphocytes (white cells), granulocytes (another white cell type), and platelets. The technical name for the white cell family is leukocytes. The biological role of each of these cell types is indicated in Figure 8.

Erythrocytes - Oxygen transport to the cells

Lymphocytes - Generate antibodies to fight infection

Granulocytes - Fight infection by phagocytosis

Platelets - Blood clotting agent and vessel integrity

Fig. 8 - Human blood cell types

Following an acute, survivable dose of radiation, the general behavior over time for blood cells is a delay period followed by a drop in cell count. The count stays depressed for a period and then recovers to normal levels. The initial delay period is caused by the fact that mature, circulating blood cells are resistant to radiation. On the other hand, the young developing cells are sensitive. Thus, when the mature cells die off, there are no replacements "standing by" so the drop in count occurs.

It is observed that 95% of the mass of erythrocytes or RBCs is hemoglobin. This is a substance which allows the cell to transport oxygen to tissues. Mature RBCs do not have any cell nucleus. Based on the Law of Bergonie and Tribondeau they should be relatively resistant to radiation. This is indeed the case. It requires a dose of over 10 Sv (1,000 rem) to cause any serious effect on the number of circulating erythrocytes. The depletion of red blood cells (i.e., anemia) is NOT an effect of survivable doses of radiation.

In general the leukocytes (white cells) have a major role in the body's infection fighting system. As shown above, lymphocytes employ a form of "chemical - biological warfare" in which they produce chemical antibodies that poison invaders. The granulocytes make use of "hand to hand combat" whereby they engulf, smother and digest the invader.

Lymphocytes are spherically shaped cells with a large nucleus. Scientists in the sixties realized that there are more than one type in the human body. The two main varieties have been designated the B type and the T type. Probably as a result of their large nucleus, lymphocytes are very sensitive to radiation. Some biologists feel that they may be the single most sensitive cell in the body. A radiation dose of a few hundred rem to the whole body will produce a severe drop in circulating lymphocytes in a matter of hours. The body will be without the protection of the lymphocytes (a key player in the body's immune system) for 4 to 5 weeks following such an acute dose. Recovery of the count to normal levels takes about 7 weeks.

As a spin-off from the extensive research efforts to combat the AIDS virus, scientists have made major breakthroughs in understanding the functioning of the human autoimmune system - the way our bodies fight infection. It is now known that there are at least four distinct kinds of T lymphocytes:

Helper T Cells identify foreign invaders

Killer T Cells attack and destroy invaders

Suppressor T Cells shut the system down after battle

Memory T Cells allow future response to the same invader

Macrophages, members of the granulocyte family, are constantly circulating throughout the body, alert for anything they don't recognize. When invading viruses comes near, some are attacked and eaten by the macrophages. This releases the unique genetic code (antigen) of that particular virus which is subsequently recognized by a helper T cell trained to read that particular antigen. The helper T sends out the alert in the form of a chemical signal, interleukin - one of a family of proteins called lymphokines, discovered around 1985. The interleukin causes the body to mass produce killer T cells and B cells which are coded to seek out and destroy any cell showing the antigen of the invading virus. The B cells produce specific antibodies to attack the invader's antigen. Eventually, as the invader is overwhelmed, the helper T cells release two other lymphokines which activate the suppressor T cells and memory T cells. The suppressor T cells shut down production of helper T and B cells and the memory T cells continue to circulate for years, coded to respond immediately if that identical virus should ever be sighted again. Thus, the person is "immune" to reinfection from that disease in the future.

Some radiobiology books quote the sensitivity of lymphocytes to be "about 25 to 35 rem." This requires further comment. The implication is that if a worker is exposed to 40 rem on the job, it will be possible to "prove it" from a blood count of lymphocytes. This is just not the case. The 25 - 35 rem limit was measured in germ free strains of laboratory animals maintained in a sterile environment from birth and then subjected to a radiation dose after repeated measurements have given the same lymphocyte count. Under these conditions, the 35 rem will produce a just detectable drop, statistically, in the lymphocyte count. The situation in the real world is that workers are not germ free and they don't usually work in a perfectly sterile location. Depending on the state of general health and recent exposure or absence of exposure to persons with infectious disease, a "normal" lymphocyte count can be anywhere from 2,000 to 5,000 per cubic mm of blood. After strenuous exercise, the number may be even higher. Thus, an exposure of a worker to 40 rem will produce a drop in lymphocytes which will be very small compared to the normal variations of 200% to 300% in the count, and so would not be observable.

Granulocytes, another leukocyte, are characterized by lobed nuclei and a "grainy" cytoplasm that looks like it contains suspended granules, hence their name. Mature granulocytes have a life span of only one day. A radiation dose of a few sieverts (a few hundred rem), acutely delivered, will produce a severe drop in about 3 - 5 days, thus, also interfering with this second body defense system against infection. The normal count is about 5,000 per cubic mm. After about 5 weeks, the depressed count begins recovering, with full recovery at about 7 weeks after the irradiation. Granulocytes are sensitive to radiation. Under the carefully controlled laboratory conditions mentioned earlier it is possible to see a drop in the count following about 50 rem of radiation. Once again, this would NOT be observable under normal occupational conditions.

Platelets are the last of the blood cells to be discussed. They are chunks of cytoplasm and completely lack a nucleus. As indicated earlier in Figure 8, they play an important role in the body in the formation of blood clots (hemostasis). In addition they are an essential component in the overall system which keeps the blood from

leaking between the "joints" in the blood vessels (remember that the tubular vessels are made by sticking rounded cells together side by side, like an adobe brick wall). The normal platelet count is about 200,000 per cubic mm. If the count falls below about 20,000 per cubic mm, the blood vessels become sieves, and the blood abruptly leaks out through the walls. Such massive hemorrhaging is, of course, fatal. The critical period for this effect is 2 to 3 weeks post-irradiation, as the platelet count has fallen to a minimum during that time. Recovery to normal levels again takes about 7 weeks following a dose of several hundred rem.

A brief summary of some of the properties of blood cells is given in Figure 9. It

Cell Type	Normal #/mm^3	Time to Minimum	Relative Radiosensitivity
Erythrocytes	5 x 10^6	3 weeks	Very low
Lymphocytes	2 to 5k	1 day	Very High
Granulocytes	5 x 10^3	7 days	High
Platelets	2 x 10^5	2-3 weeks	Medium

Fig. 9 - Response of blood cells to radiation

might be noted that the $LD_{50/60}$ acute dose would produce a severe response in the blood system. In humans, as mentioned early in this Chapter, the best current estimate for $LD_{50/60}$ is about 410 rem whole body, with no medical intervention. By isolating a person in a sterile environment for care and giving antibiotics as needed, acute doses at twice the $LD_{50/60}$ can be survived!

Gastrointestinal System

The mouth and esophagus are radioresistant. Doses above the maximum survivable dose are needed to produce ulceration in these organs. This is consistent with another general rule of thumb – all muscle and connective tissues of the body are radioresistant.

The stomach is more sensitive. At doses in the neighborhood of a few Sv (few hundred rem), the chief and parietal cells which are embedded in the lining reduce or completely stop their glandular secretions. These include HCl and pepsinogen which aid in food digestion. Following a delay period, these cells again resume production at normal levels.

The small intestine is an important organ from two points of view. It is the most radiosensitive organ in the GI tract. It also is the organ which, at least at the present time, determines whether a person will survive an acute massive whole body radiation dose. The drawing in Figure 10 indicates some of the anatomical features of the small intestine. The small finger-like projections are called villi. Each villus is surrounded by depressions called crypts. Deep in the bottom of the crypt is a special cell which produces lining (epithelial) cells as offspring. These cells move slowly up the side of the crypt. When they reach the surface, they are fully mature and move up onto the

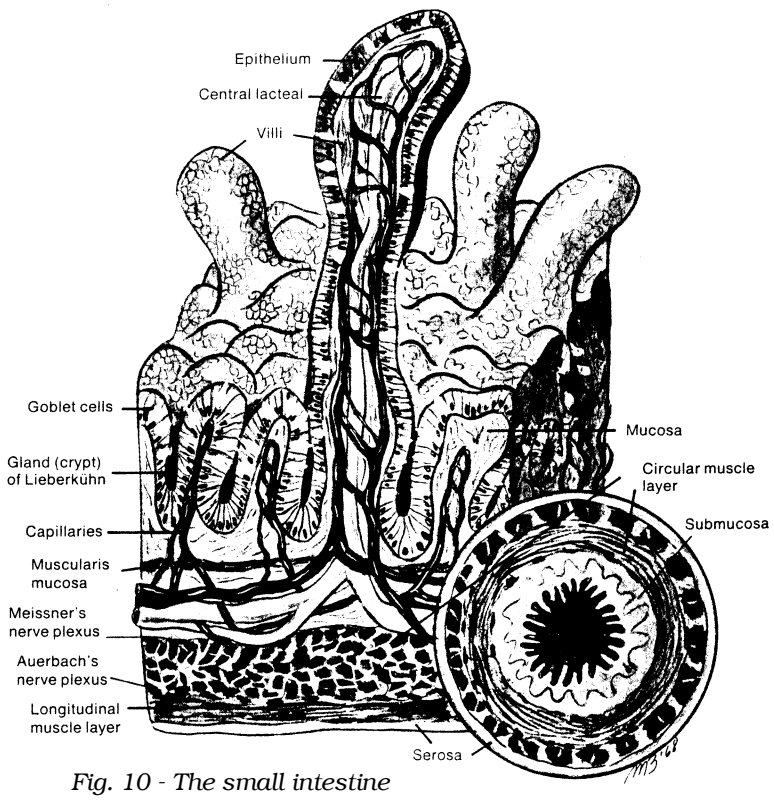


Fig. 10 - The small intestine

Textbook of Physiology, 17th Ed., Schottelius and Schottelius, C.V. Mosby Co., 1973

villus. They protect the minute, fragile capillary bed housed within the villus, and are permeable to food nutrients passing through into the capillaries. Due to the friction of the food passing through and to the peristaltic muscle contractions in the wall, the lining cells on the tips of the villi are worn away. These cells are replaced by new cells moving up from the crypts.

Consider, now, the effects of irradiation on this "assembly line" progression of cells. According to the Law of Bergonie and Tribondeau the most radiosensitive point in the process is the crypt cell which is undergoing rapid and continuous mitosis to produce cells which are not yet fully specialized. A dose of a few Sv (few hundred rem) to the gut will cause the crypt cells to temporarily cease dividing. Mature lining cells will be unaffected. In a few hours the mature lining cells will have sloughed off but there will no longer be new replacement cells moving into position. This will cause the villus to retract, over a period of 5 to 10 days, and expose the delicate capillary bed. The capillary bed will be ruptured which allows blood plasma and electrolytes to leak into the GI tract. Equally important will be the movement "backwards" of intestinal bacteria directly into the bloodstream through the open capillaries. This will rapidly spread infection throughout the body at a time when its immune system has been drastically affected by loss of leukocytes.

At doses below 10 Sv (1,000 rem) the crypt cells will recover in about a week. This will allow rebuilding of the villi and enclosed capillary bed. The person should recover. At doses over 10 sieverts the crypt cells do not survive. In humans, death would be expected within 6 to 10 days. Due to the present impossibility of surgically

replacing an entire small intestine with its uncountable millions of capillary connections, the dose of 10 Sv (1,000 rem) acute whole body represents the maximum survivable dose for humans.

The maximum survivable dose should not be confused with $LD_{50/60}$. That term refers to the dose to a group of persons which would result in 50% of them surviving after 60 days, WITHOUT MEDICAL TREATMENT. In other words, it would apply under nuclear warfare battlefield conditions or a nuclear terrorist attack in a remote area of the world. Under normal occupational conditions or accident conditions it is reasonable to expect that effective medical treatment would be able to be mobilized in a relatively short time interval. With treatment occurring within a few hours post-exposure, humans can survive 1,000 rem of radiation delivered acutely. Even in the confused aftermath of the Chernobyl reactor explosion, it was possible to assemble an international team of medical experts to perform bone marrow transplants within the time such a procedure might still be beneficial to the patient. This procedure and it's usefulness with radiation accident patients is covered in more detail in Chapter 14 on **Handling Nuclear Emergencies**.

The large intestine is about as radiosensitive as the stomach. Doses over 10 sieverts are needed to produce ulcers in this organ.

Central Nervous System

This organ system consists chiefly of muscle and connective tissues. Based on the Bergonie and Tribondeau rule of thumb, the CNS system should be resistant to radiation exposure. This is the case. It requires doses over about 50 Sv (5,000 rem) to produce effects known as the "cerebrovascular syndrome." Symptoms such as severe nausea and vomiting appear within minutes. Loss of muscle control, seizures and lapsing into a coma follow shortly thereafter. At lethal doses, the parasympathetic nervous system is affected. This system normally originates and sends the electrical signals needed to fire the contractions in the heart muscle to cause heartbeats and in the diaphragm muscles which force lung inflation. At high doses, these signals become irregular or stop altogether for periods of time. This leads to death of the exposed individual in a few hours. One possible mechanism for death in this high dose range is a sharp increase in pressure on the brain due to a buildup of fluid within the skull. Since death in this extreme dose range is so early, the victim's body does not have time to exhibit either the gastrointestinal or blood system effects that were discussed earlier.

Historically, a few individuals were victims of the cerebrovascular syndrome as a result of nuclear criticality accidents associated with weapons development. However, diligent efforts worldwide to create radiological safe workplaces, and with the demise of the "cold war" with its need for continual production of nuclear weapons, it was felt, with some justification, that no one would ever again experience the full CNS effects. Unfortunately, this has not proven to be the case. After more than two decades without a criticality accident fatality, an accident at a Russian facility in June of 1997 (see Chapter 14 description) delivered a lethal radiation dose of 50 Sv to a scientist. He began experiencing the cerebrovascular syndrome 30 minutes later and died 64 hours later.

Reproductive System

In human males, mature sperm are radioresistant. However, consistent with the Law of Bergonie and Tribondeau, production of spermatogenesis (immature sperm cells) is reduced or disrupted at low doses. A dose to the gonads (sex organs) of 2.5 Sv (250 rem) will produce a temporary sterile period of one year. To produce permanent sterility in males, a dose between 5 and 6 Sv is needed.

In the human female, a dose of about 1.7 Sv (170 rem) to the ovaries will produce a 1 to 3 year sterile period. A dose between 3 - 6 Sv is needed to produce permanent sterility, depending on age at the time of irradiation.

Conclusions

As indicated earlier in this chapter, the different radiations produce different effects per rad due to the RBE effect. In the case of human tissues, alpha particles will normally be stopped in the dead protective layer of skin cells and, therefore, not produce external exposure. Beta particles have a lower stopping power and thus, will penetrate several mm into tissues exposed externally. This will have the greatest effect on two tissues - the active growing layer (dermis) of the skin and the lens of the eye. Similar results would be expected for very low energy gamma rays. Doses delivered to surface tissues are recorded as “shallow dose” on a personnel dosimetry report. Higher energy x- and gamma rays would produce most of their dose in deeper lying body organs. It is necessary that the photons travel some distance through the tissues before enough interactions have taken place to produce the secondary electrons (photo, Compton and pair production electrons) that deposit the dose. The example of erythema, the reddening produced in irradiated skin, illustrates this well. A dose of 2.7 Sv (270 rem) of low energy x-rays will produce erythema but it takes a dose of 10 Sv of high energy (1.25 MeV average) cobalt-60 radiation to produce the same effect. The lower attenuation coefficient means a larger “buildup thickness” before significant energy is deposited. Thus, the cobalt gamma rays leave a smaller fraction of their energy at the depth of the dermis than the x-rays, so a higher dose is needed. The dose produced at depth by high energy radiations is listed as the “deep dose” in personnel dosimetry reports.

Fast neutrons produce both a surface and depth dose (shallow + deep). Neutrons are also noted for their effectiveness in producing cataracts or clouding in the lens of the eye (a process called cataractogenesis). Some recent (and disputed) experimental evidence suggests the RBE might be as high as 100 for chronic low level fast neutron exposures that lead to a cataract.

Whole Body Effects In Humans

Acute Effects

Acute, as used here, refers to biological effects that are evident over a period of up to a few months following a radiation dose which was also received over a period of

Range →	Subclinical Range 0 - 1 Sv	Therapeutic Range			Lethal Range	
		1 to 2 Sv	2 to 6 Sv	6 to 10 Sv	10 to 50 Sv	Over 50 Sv
Vomiting Incidence	NONE	5% to 50%	3 Sv, 100%	100%	100%	100%
Delay Time	– – – –	3 hr	2 hr	1 hr	30 min	
Leading Organ	NONE	Bone Marrow			GI Tract	Central Nervous System
Characteristic Signs	NONE	Moderate leukopenia	Severe leukopenia, hemorrhage, infection, purpura, epilation > 3 Sv		Diarrhea, fever, electrolyte loss	Convulsions tremor, ataxia
Therapy	Reassurance	Blood monitoring	Blood transfusion Antibiotics	Marrow transplant? Growth factors?	Maintain electrolytes	Sedatives
Prognosis	Excellent	Excellent	Good	Guarded	Hopeless	
Incidence of Death	NONE	NONE	0 to 80%	80 to 90%	90 to 100%	

Fig. 11 - Immediate clinical effects of acute radiation

up to a few hours. The whole body effects are summarized in Figure 11.

This chart lists various dose ranges across the top and then lists effects and comments in the vertical columns. The term "Subclinical" means that any effects would be so small that they would not be detected by a routine medical examination. It does not mean that there are no effects. Effects such as increased risk of some form of cancer or genetic effects in offspring may not be evident for many years. It does mean that a medical examination would not be able to prove or disprove an alleged radiation dose in a person exposed to less than 100 rem whole body.

The next dose range, the "Therapeutic Range," covers acute doses of 1 to 10 Sv (100 - 1,000 rem). Of the Chernobyl accident responders, 203 received a dose of 1 Sv or higher. The most critical problem in this range will be maintenance of minimum levels of circulating blood cells. Since many of the blood cells originate from stem cells in the bone marrow, the effects seen in this range are often referred to as the "bone

marrow syndrome." Under "Signs," leukopenia means a drop in the leukocyte (white cell) count. Purpura denotes small splotchy red or purple spots on the skin caused by rupture of a capillary with subsequent leakage of a small amount of blood under the skin layers. Epilation indicates the loss of hair. Generally, hair will regrow within about a month, although the texture of the new hair will be coarser and the color lighter.

In the next dose range, 10 to 50 Sv (1 to 5 thousand rem), the most critical organ is the small intestine. The effects in this range are, therefore, called the "Gastrointestinal Syndrome." Fever is caused by bacterial invasion of the bloodstream. The electrolyte balance is upset by leakage of blood plasma into the GI tract through the broken villi. A number of the early responding emergency personnel at the Chernobyl accident suffered death after exhibiting the gastrointestinal syndrome. One victim of the 1999 Japanese criticality accident was exposed to 17 Sv which was also in the GI syndrome range.

The lethal dose range, over 50 Sv (5,000 rem), exhibits the "CNS Syndrome" or "Cerebrovascular Syndrome" due to the effect on the parasympathetic nervous system. Ataxia refers to loss of muscular coordination. Death usually occurs within hours. Persons who have been exposed in this range through accidents have fallen unconscious within a few minutes and died without regaining consciousness.

In interpreting the summary table (Figure 11), it is important to recognize that PSYCHOLOGICAL STRESS was not taken into account. This factor can greatly alter some of the numerical results presented in the chart. An actual occurrence may serve to illustrate this. A group of police officers were examining a recovered stolen car for evidence. In the course of the search, they pried open the vehicle's trunk. Inside they found a cardboard box with a "Caution - Radioactive Materials" label attached. In a short time they began feeling nauseated, at which point they hurried to a local hospital. Upon arrival at the emergency room they proceeded to vomit. Reference to the biological effects chart would indicate exposure to at least 300 rem. When health physicists arrived at the scene, they soon determined that the box was completely empty and there was no radioactive contamination present.

Human Late Effects – Genetic

Late effects are those which exhibit themselves a period of years after an acute exposure. The incidence is generally dependent on the radiation dose, dose rate, age at the time of irradiation and state of health. The effects are usually grouped into two categories - genetic effects which show up in the offspring of exposed persons (and are discussed in this section) and somatic effects which occur in the exposed person and will be discussed in the next section.

There is no question that radiation is a mutagen, that is, causes genetic mutations. Mutations are caused by a change in the composition of a gene. This can happen through loss of DNA, through gaining extra DNA, or by a rearrangement of the order of the DNA coding units within a gene. Radiation can change genes through ionization interactions that deposit enough energy locally to break one or both of the strands that are characteristic of the DNA molecule. Or, free radicals can be released by radiation which will directly attack the DNA molecule to change the genetic code.

The actual risk of mutations is difficult to estimate, particularly for the low doses and dose rates encountered in the occupational environment. Most of the reliable data was obtained under conditions of several hundred rem doses delivered acutely. It is not clear whether mutation risks at lower doses are dose proportional (i.e., the linear hypothesis holds) or whether the risk is greatly reduced at low doses and rates (the threshold hypothesis).

It is clear that genetic effects are stochastic in nature. This merely means that the chance of genetic injury depends on the radiation dose. But the severity of the injury is not dose dependent. In contrast, a deterministic effect is one which has a threshold dose and the severity of the observed effect increases with dose. An example would be a cataract in the eye caused by radiation exposure or an x-ray skin burn.

An early source of numerical estimates of genetic risk was the "Mega Mouse Project," so named because of the use of over 7 million mice over the course of the project. This work was conducted at Oak Ridge National Lab in Tennessee. Mice were continually exposed to radiation from gamma ray sources placed around their cages. A number of mice were followed over many generations of offspring and a variety of mutations observed and correlated with irradiation conditions. Two conclusions reached were that it took about 0.4 sievert (40 rem) to get a mutation and that the mutation rate seemed directly proportional to dose.

Before discussing the numerical estimates for humans it is important to be aware of what a radiobiologist means by the term "mutation." Members of the general public often have their concept of mutations shaped by late night television reruns of old monster movies. This is not a campaign to stamp out old horror movies, but the misconceptions encouraged have probably done a disservice in making non-emotional discussions of genetic effects of radiation more difficult for persons in the radiation protection field. The fact is that radiation exposure does NOT produce mutations which are unusual or have never been seen before. What radiation does is to INCREASE THE RATE of mutations which already occur naturally in the population.

Geneticists have identified four main types of genetic mutations that occur in humans and animals. Dominant mutations are caused by a single gene from one of the parents of the offspring. A recessive mutation can result only if the same altered gene is furnished by both parents. Chromosomal rearrangements lead to mutations by mixing up the order of the genetic code. Finally, the multifactorial type of mutation is the result of multiple genes being involved along with the probable influence of environmental factors such as eating habits.

The U.N. World Health Organization has identified well over 200 different conditions or diseases that are genetically transmitted from parent to offspring. Most of these are survivable, and many are very common. In fact, the gross malformation mutations (extra appendages or major organs missing, etc.) usually are not live-born. The rather strict "quality assurance" standards within the human body usually lead to a spontaneous abortion at an early point in the pregnancy. The table in Figure 12 is a short listing of some of the conditions considered a genetic mutation by biologists.

One of the common concepts used to describe the numerical relative risk of genetic effects of radiation is the Doubling Dose. This is defined as "the radiation dose which, if delivered to a large population, would produce an additional number of mutations equal to the number of spontaneous (natural) mutations." If this were the case, then the total number of mutations would be:

Congenital Cataracts	**Low Gamma Globulin**
Anemia	**Steroid Imbalance**
Muscular Dystrophy	**Albinoism**
Down's Syndrome	**Diabetes**
Epilepsy	**Asthma**

Fig. 12 - Some examples of genetic mutations

Total Number = Natural + Radiation Induced

= 2 x Natural.

Thus, the number of mutations would be doubled in the population. One reason for the popularity of the Doubling Dose concept is that it includes a "built-in" perspective. It compares the amount of radiation needed to produce some effect with a pre-existing natural phenomenon – the spontaneous mutation rate. This spontaneous rate is the equilibrium rate in the population of mutations produced by all of the various environmental mutagenic agents acting over millions of years. Each newborn child which exhibits one of the "genetically transmitted conditions" previously referred to is a spontaneous mutant, biologically.

It would be reasonable to assume that the natural mutation rate in the U.S. population is a readily available measurement. This is not the case. Based on a British Columbia study of over 750,000 live births, the U.N. Scientific Committee on the Effects of Atomic Radiation, UNSCEAR, concluded (in their 1977 report) that the spontaneous mutation rate in North America was 10.5%. That corresponds to approximately one new birth in ten. The problem recently recognized is that they only followed up the study population for the first 25 years of life. Geneticists now know that both dominant mutations and multifactorial mutations often do not show up until much later than 25 years. Thus, these components were greatly underestimated. The BEIR V Committee, in their 1990 report, used the currently accepted figures shown in Figure 13. As can be seen, the 10.5% figure was low by a factor of more than 10

Mutation Type	# per million Live Births
Dominant	**10,000**
Recessive	**2,900**
Chromosomal Rearrangements	**4,400**
Multifactorial	**1,230,000**
TOTAL	**1,247,300**

Fig. 13 - The natural spontaneous mutation rate

times! On the average, each live-born child will experience 1¼ significant diseases, that had a genetic origin, over their lifetime.

Returning to the actual numerical values, it is difficult to obtain a reasonable estimate of the Doubling Dose. This is a result of the fact that the many variables present in a study of living human populations cause statistical fluctuations which are large compared to the anticipated changes in mutation rates caused by exposure to radiation doses up to tens of rem. These days, there are few groups of persons chronically exposed to those doses as a result of effective radiation protection programs conducted by persons such as yourselves. Even the persons who perform maintenance on nuclear power reactors have a dose much smaller than this range. In spite of the difficulties, some estimates of Doubling Dose have been made over the years by the Biological Effects of Ionizing Radiation (BEIR) Committee and the UNSCEAR. In their 2006 report, the BEIR VII Committee updated their calculations and estimated the human Doubling Dose for chronic irradiation at 82 ± 29 rads. Finally, studies summarizing a 40-year-long follow-up of the survivors of the Japanese detonations, taking into account the revised dose estimates published in 1981 for the Hiroshima and Nagasaki devices, arrived at an average Doubling Dose for the Japanese survivors of 156 rem acutely delivered.

While the doubling dose represents a relative risk of genetic injury (the risk is compared to the natural mutation rate), it is also possible to arrive at absolute risk estimates. An absolute risk is expressed as the risk per unit radiation dose. The ICRP assumes a value of 0.6 per person per rem of exposure as the absolute radiation risk of genetic injury to a person's offspring.

Much of the human evidence for radiation effects comes from studies of the survivors of Hiroshima and Nagasaki. For many years, results of these studies all depended on radiation dose estimates developed at Oak Ridge in 1965. These estimates gave the neutron and gamma ray dose equivalents from the nuclear detonations at various distances from ground zero. According to the dosimetry, Hiroshima survivors were exposed to a much larger fraction of neutrons than the Nagasaki survivors. The differences in the radiation fields for the two cities allowed scientists to calculate neutron RBE values for humans. Then, it came as a shock when two groups independently announced, in 1981, that the original 1965 dose estimates were sharply in error!.

New estimates were prepared by Livermore National Laboratory and by Oak Ridge National Laboratory. The results were published in 1987 and designated "DS86." They show that the neutron doses in Hiroshima were about ten times less than previously thought and the gamma doses actually higher than the previous estimate. The 1965 values were in error because of two factors. Accurate computer models of the neutron and gamma ray output of atomic bombs were not available in 1965. Instead, the Little Boy output was approximated by making physical measurements near an unshielded nuclear reactor sitting atop a metal tower in the Nevada desert. Unfortunately, the dry Nevada air was a poor approximation of the humid morning air of seacoast Hiroshima. The moisture in the Japanese air attenuated the neutron spectrum by about 10 times more than the dry air near the test reactor. Scientists had to sort back through several decades of previously published radiation studies and had to recalculate the doses to take account of the DS86 changes.

As discussed earlier in this Chapter, the "final" dosimetry study of the Japanese survivors was released in 2003. It was designated "DS02," and revised many neutron and gamma dose estimates from the DS86 study. The DS02 estimates no longer have discrepancies in the neutron components. This should allow more accurate RBE estimates to be prepared for neutrons.

Human Late Effects - Somatic

The somatic effects of radiation, i.e., results produced in the exposed individual, also show late effects. For discussion purposes these are often divided into three categories - life shortening, leukemia, and other cancers. The reduction in expected life span appears to be the result of accelerated aging. This is quite different than a shortened life due to contracting a fatal disease. The effect is well documented in animal experiments. Early radiologists who received large doses likewise showed reduced life spans. If the somewhat dubious extrapolation is made for humans at low doses, the result is an estimated loss of life expectancy of a few days per rem of radiation dose.

The knowledge that radiation causes cancer dates back to the nineteenth century. High rates of lung cancer in the Czech Republic uranium miners were diagnosed in 1879. However, current occupational levels of radiation exposure carry very low risk, and background radiation even less. It has been calculated that on the average, within a person's body, ten million cells a minute are struck by ionizing radiation. Clearly, not all radiation causes cancer!

Cancer is considered a stochastic effect by biologists. Thus, the chance of getting cancer from radiation is dependent on the radiation dose, but the severity of the cancer is not dose dependent. Finally, not all cancers are caused by radiation. Many other carcinogens are present in the workplace and the environment. Cancers for which the radiation risk and the mortality risk are high include the digestive system, lung, leukemia and female breast. An example for which the radiation risk is high but mortality risk low is thyroid cancer. This means that the thyroid gland is very sensitive to radiation but the resulting cancer develops slowly and can be very successfully treated with surgery or radioactive iodine.

Discoveries regarding the origins of cancer seem to make the news almost daily. Although progress has accelerated recently, scientists are still puzzling out many mysteries in this regard. Cancer refers to a disease in which the body's cells have lost the ability to regulate their orderly growth. A tumor can take up residence in some body organ and then through rapid cell division, attack the tissues of the host organ. Small pieces can break off and transport themselves (metastasize) via the blood vessels or lymph channels to remote sites and begin malignant growth in their new home. It now appears that a cancer can develop only following a multi-step process – initiation, promotion and progression. The initiation step involves activating a recently discovered class of genes called oncogenes. Apparently, oncogenes are normal genes in somatic cells which have mutated. Radiation produced double-strand breaks in DNA appear to play a major role here. The oncogenes play a critical role in the signals that cells send and receive to stay in regulation. Initiation was shown in 1986 to necessarily involve many cells. As of 1999, over 50 different human oncogenes have been recognized.

The second step, promotion, is thought to not involve the mutation of genes. Some promotional agents which "turn on" the initiated cancer interrupt the communication pathways between cells. Others significantly increase the level of free radicals which like to attack DNA. It has been learned that the cancer promoting agent must be present over a long time period to successfully result in a viable tumor. To complete the story on oncogenes, it should be mentioned that a second family has been discovered. These are the tumor-suppressor genes, or suppressor genes. They are apparently able to cancel the effect of oncogenes and prevent initiation of a cancer. Their presence is the reason that many individuals with an inherited tendency to certain forms of cancer never develop them during their lifetime. It is thought that normal cells naturally contain a suppressor gene. At this time, a few suppressor genes have been traced to specific locations on human chromosomes.

Leukemia, a cancer of the blood, and its relationship to radiation dose is well studied. One reason is the relatively short delay time (known as the latent period) between radiation dose and the onset of the disease. In most forms of solid tumors, the latent period may be from 20 to 50 years. In the Japanese survivors, the peak incidence of new cases of leukemia occurred around 7 years after exposure and returned to natural levels after 20 years. Death from leukemia in this group is best described by the "linear-quadratic" model. This just means that the mortality rate is proportional to dose at low doses (the linear part) and proportional to the dose squared at high doses (the quadratic term). Elevated leukemia risk has been seen in some groups of persons receiving medical x-rays. However, a 1991 study of over 1,000 leukemia patients failed to show any statistical connection between the leukemia and diagnostic x-ray exposure.

At the present time, one of the best summaries of human data for radiation protection purposes is **Health Risks From Exposure to Low Levels of Ionizing Radiation** from the U.S. National Academy of Sciences BEIR VII Committee. This report was issued in 2006 and was based exclusively on the Japanese A-bomb survivor data. It incorporated the DS02 A-bomb dosimetry data. They estimated the risk of dying of leukemia following an acute, one-time radiation exposure to 0.1 Sv (10 rem) per 100,000 persons over the rest of the exposed person's lifetime. For men, the value is approximately 100 excess deaths and for women the value is approximately 72 excess deaths.

To put this into perspective, pick a city of 100,000 population (e.g., Berkeley, CA, Waco, TX or Columbia, SC). Deliver 0.1 Sv (10 rem) to the entire city. Based on the BEIR VII report, without any extra radiation beyond background levels, about 560 of those 100,000 residents will ultimately die of leukemia (from "natural causes"). Thus, the 0.1 Sv radiation dose will increase the leukemia mortality rate by 15% (extra 86/560 x 100%) where the leukemia radiation risk was averaged between the men and women in the exposed city.

Based on the Law of Bergonie and Tribondeau, children should be more likely to contract radiation-caused leukemia than adults. In several reported studies, the leukemia risk to fetuses exposed during abdominal x-ray procedures on the mother has been measured. The risk is about 40% higher than in non-irradiated fetuses. The doses delivered are in the 0.01 to 0.1 Sv range (1-10 rem). The studies have been subjected to considerable criticism by other scientists. In fact, in a 1970 study of 1,000 fetuses exposed *in utero* by the atomic bombings of Japan, not a single leukemia case resulted. However, a more recent 1988 study of 1630 fetuses exposed to the Japanese

bombs showed two childhood leukemia cases. The revised dosimetry for the Japanese survivors now also tends to support the increased childhood leukemia risk. The 1990 BEIR V Committee concluded "that susceptibility to the carcinogenic effects of irradiation is high during prenatal life."

The risks for other types of cancer are roughly comparable in size to the leukemia risks. Gender specific values taken from the 2006 BEIR VII report are summarized in Figure 14. These are all expressed as EXCESS cancer deaths expected for 100,000 persons over their remaining lifetimes following 0.1 Sv (10 rem) of exposure. Without an added radiation dose, U.S. mortality figures predict 20,360 lifetime cancer deaths in a population of 100,000 persons, i.e., one in five!

Excess lifetime cancer deaths per 100,000 persons exposed to 0.1 Sv

CANCER TYPE	EXCESS RISK-Males	EXCESS RISK-Females
Leukemia, adult	100	72
Lung	140	270
Stomach + colon	95	71
Breast, female	- - -	73
Total for all solid cancers	480	740
TOTALS for all cancers	580	812

Fig. 14 - Excess cancer mortality estimates (BEIR VII, 2006)

Effects Of Chronic Irradiation Conditions

All of the information presented above was for the case of the radiation being delivered over a short time interval. In many cases (e.g., the occupational work environment) radiation exposures may be steadily accumulated by an individual over a period of many years. Under these conditions, the biological repair mechanisms have an opportunity to work. Experimenters have demonstrated that the dose received by a cell over the cell cycle time is more important than the dose rate. Thus, low dose rates mean that a given cell will not receive a lethal dose in one cell cycle, and survival of the tissue is assured. In early work with cells, biologists showed that dividing a given dose into fractions which were delivered with a resting period between could reduce the overall effect of the radiation by as much as eight times. That a reduction of effects is the case for humans as well is easily demonstrated. A radiation worker receiving 50 millisieverts (5 rem) per year over a 50 year working lifetime would receive 2.5 Sv (250 rem) total. No effects would be seen on a clinical examination. Yet if the same worker received the 2.5 Sv acutely in an accident, a number of clinical effects would be evident such as depressed blood counts and possible vomiting.

Actual human experience with chronic exposure situations at low dose rates often leads to conflicting interpretation among the experts. The 1976 "Mancuso Study" purported to show elevated cancer rates in Hanford radiation workers. Others argue that this same study showed lowered cancer rates. The differences are dependent on the "control group" selected for comparison. Persons participating in the Smokey atomic tests at the Nevada Test Site are alleged to show higher than expected leukemia

rates. Yet a 1973 Los Alamos study of 25 workers with body burdens of from 2.5 to 30 times the maximum permissible legal level of plutonium followed the workers for 27 years. They concluded that, "to date, none of the medical findings in the group can be attributed definitely to internally deposited Pu. We conclude that the body has protective mechanisms which are effective in discriminating against these materials following some types of occupational exposure."

In recognition of the existence of biological repair mechanisms, the National Council on Radiation Protection and Measurements (NCRP), recommended in their 1980 Report # 64, the establishment of a "Dose Rate Effectiveness Factor," DREF. This would be a numerical constant by which radiation doses recorded on personnel radiation histories would be REDUCED before entry on the history. The DREF could only be used for accumulated doses below 20 rem and dose rates less than 5 rem per year. Under these conditions, the NCRP recommended a DREF of between 2 and 10 for low LET radiations like x- and gamma rays. See Sample Problem 4 for an example. In 1997, the NCRP settled on an endorsement of a DREF of 2. Their Report #126 on cancer risk used this value. Also, the ICRP now uses a DREF of 2. Hopefully, at some future date, the lawmakers will recognize the validity of the DREF concept.

Sample Problem 4

GIVEN:

Tom Smith has a 4 year radiation worker history with a lifetime dose of 0.046 Sv.

FIND:

If a DREF of 5 were agreed upon at some future date, how would Tom's dose history be rewritten?

SOLUTION:

A 4.6 rem dose meets both criteria – less than 20 rem total and less than 5 rem per year. Thus, the entire dose can be divided by 5. His new dose history would show 0.046 Sv / 5 = 0.0092 Sv. The reduction of 0.0368 Sv or 3.68 rem is accounted for by the biological repair that functions at low dose rates.

Developing Human Embryo/Fetus

As would be predicted by the rule of thumb of Bergonie and Tribondeau, the earliest human stages of a developing life are characterized by unusually high radiation sensitivity. Radiation doses delivered during pregnancy can produce spontaneous abortion, malformed organs, mental retardation, growth retardation or teratogenic effects. The types of injury produced depend on the radiation dose and are particularly related to the time during pregnancy at which radiation exposure occurs.

Prenatal death is observed when the radiation is received during the preimplantation phase. This would be during the first 2 weeks after conception. Production of deformed organs and limbs may be observed for exposures delivered 2 to 6 weeks after conception. This is the organogenesis phase in human development. After

6 weeks, radiation exposure appears to produce stunted growth and possible mental retardation. The peak for mental retardation seems to be from 8 to 15 weeks after conception. Some of the Japanese A-bomb survivors that were irradiated in utero exhibited reduced head diameter, reduced height and weight and lowered IQ compared to the control population. Abnormalities in physiological development are given the special name teratogenic effects. Such birth defects are a known result of irradiation *in utero*. Other causes of teratogenic effects are certain medicines, chemicals (e.g., PCB, mercury), infections (e.g., German measles, syphilis) and certain diseases (e.g., diabetes, iodine deficiency).

Most of the information on embryonic and fetal effects of radiation was obtained from rat and mouse experiments. Data from the Japanese survivors and from a limited number of patients undergoing medical radiation therapy tend to support the animal conclusions. Statistically, it is difficult to prove radiation caused a particular effect in humans as the numbers are so small and the effects observed are those which regularly occur naturally. For example, malformations evident at birth are normally observed in 6 percent of all human births.

Prospective parents are sometimes very concerned if the woman receives medical or occupational radiation during her pregnancy. Some guidelines have been developed for this situation. It is suggested that no more than 0.10 rem be delivered to the embryo/fetus during any single month of pregnancy. If this is exceeded, some thought should be given to a possible therapeutic abortion.

Most of our limited knowledge of radiation effects during pregnancy are from women exposed to external radiation. In the case of internally deposited radionuclides in the mother, the information is even less reliable. The situation is complicated by uncertainties in what fraction of the radionuclide crosses the placenta to the developing child, by the small size of fetal organs so that a substantial fraction of the radiation energy is deposited outside the organ of deposition and by differences in uptake between the mother's organs and the fetal organs. (For example, iodine concentrates higher in the fetus than in the mother.) NCRP Report Number 128 is devoted entirely to the issue of dose to the embryo/fetus following uptake of radioactive material by the mother and includes data for a large number of specific radioactive elements.

Radiation Hormesis

One of the most exciting developments in the entire field of radiation protection is still receiving much attention by radiobiologists and health physics researchers. This is the idea of radiation hormesis and the accompanying concept of "radiation deficiency." Professor T. D. Luckey of the University of Missouri, Columbia, is credited with making hormesis a household word among radiation protection practitioners. In a major review article published in **Health Physics**, December 1982, Dr. Luckey stated, "Extensive literature indicates that minute doses of ionizing radiation benefit animal growth and development, fecundity [ability to produce offspring], health and longevity. Specific improvements appear in neurologic function, growth rate and survival of young, wound healing, immune competence, and resistance to infection,

radiation morbidity [radiation sickness], and tumor induction and growth." He then proceeded to summarize over eight decades of research published in the open literature which supports a hormetic response to ionizing radiation.

The word hormesis, taken from Greek "to excite or stimulate," was coined in 1942 by Southam and Ehrlich, who discovered that a certain substance stimulated the growth of fungi at low concentrations but suppressed growth in high concentrations. Today, such a behavior is not considered unusual. In pharmacology, a vast array of chemical substances used to treat various human conditions exhibit a hormetic behavior – that is, at a low dose level many drugs produce a beneficial effect while, at high dosages, they are harmful or even fatal. In current usage, "radiation hormesis" is used to describe the production of any physiological effect of radiation that is observed at low radiation doses which cannot be expected based on an extrapolation downward from the toxic effects at high doses. As of this writing, there are three such recognizable effects:

a) increased life span
b) increased growth and fertility
c) reduction in cancer incidence.

Radiation deficiency refers to the fact that radiation may be a necessary ingredient for health, like vitamins. Thus, if an organism is prevented from receiving the minimum radiation dose needed, it may show signs of ill health as a result of the deficiency. A summary of work relative to hormetic effects observed at the cellular level is presented first. Then, summaries of immune system effects and some human findings that demonstrate radiation hormesis will be presented.

Normally, in cells which continually replace themselves (such as human blood cells), an organism achieves a very close balance between the production rate of new cells and the rate of loss due to death or damage of old cells. Cancer is an example of such a system out of control. The balance between production and loss can be upset by ionizing radiation. The typical response of cells to a continuous, low level radiation field is an "adaptation" in which the output of new cells increases to compensate for higher losses due to the radiation damage. The cycle time (time between cell divisions) is shortened and more cell divisions occur before the cells mature. These effects have been demonstrated in mammals for intestinal lining cells, bone marrow cells, sperm cells and liver cells. The radiation dose rates are typically a few sieverts per day.

When animals are exposed to low levels of ionizing radiation, they frequently show a rise in their white cell (lymphocyte) count. As mentioned earlier in this chapter, B lymphocytes are able to cause production of antibodies which attack and wipe out invading viruses. Elevated antibody levels are seen in irradiated animals. Thus, lightly irradiated animals show fewer infections than the control animals. The role of the T lymphocytes was also covered earlier. In the case of a malignant tumor which has become established in the body, suppressor T cells are able to protect the growing cancer by fending off the helper T cells. But suppressor T cells are highly sensitive to radiation – a low dose will wipe them out while sparing the helper T cells which can then gain the upper hand in attacking the tumor.

A number of population studies appear to demonstrate radiation hormesis in humans. Such effects have been seen for both high LET radiation (alpha particles) and low LET radiations (beta, gamma, and x-ray).

Many plutonium workers in U.S. Department of Energy facilities

have significant body burdens (plutonium deposited internally) and have been medically studied for many years, often as long as 25 - 35 years following initial inhalation uptake. As of 1987, 620 workers from Rocky Flats, Los Alamos and Hanford carried measured body burdens exceeding 10% of the allowed level for a radiation worker. Based on U.S mortality rates, adjusted for age of the workers, there should have been 16 lung cancer deaths expected through 1985 for this group. In fact, there were only three lung cancer deaths in this group.

Data from numerous studies of lung cancer rates in persons exposed to low level concentrations of radon gas appear to show a hormetic response for this type of internal alpha exposure. Some examples include:

a) Southern Finland - this area of the country has elevated indoor radon concentrations. Levels average three times higher than the normal level nationwide. The lung cancer rate for women, between 1955 and 1974, was 8% lower in the southern area compared to the rate for Finland as a whole. (Women from a couple of decades ago were purposely chosen as the subjects to reduce the influence of cigarette smoking on the results.)

b) Guangdong Province, People's Republic of China - This area has a region with deposits of monazite, a thorium ore. The radon levels average 2.4 times higher than in a similar, adjacent region of the province used for a control population. The population is very stable; 91% of the inhabitants have lived there for six or more generations, and a third of them have been there for over 16 generations! The lung cancer rate in the high radon region is 15% lower than the rate in the control population.

c) Cumberland County, PA - This area in southeastern Pennsylvania has indoor radon levels which are nine times higher than the average for the United States. The lung cancer rate for women, between 1950 and 1969, was 13% lower than the national average.

Human population studies of radiation hormesis have also been reported for external, gamma radiation exposure at low levels. Data from a five-year study in India, published in 1990, show that total cancer incidence, total cancer deaths, leukemia incidence, female breast cancer and lung cancer incidence are all strongly correlated to the inverse of the background radiation rate. In other words, cities with a higher background radiation level definitely have lower values for the cancer rates mentioned. India is a particularly good location for this type of research. The natural rate of cancer incidence in the country is only one-fourth that of the United States. Presumably the lower level of industrialization translates into fewer carcinogens in the environment. With such a low natural cancer rate, it should be much easier to see the effect of some agent, like radiation, on that rate. Figure 15 shows the overall cancer rate in the Indian cities examined versus the background rate. For comparison, the linear hypothesis predicted by UNSCEAR 1988 risk projections is also shown.

The Guangdong Province high radiation background region has also been studied for external radiation hormesis. The average annual external background dose rates were 196 mrem and 72 mrem in the high radiation and control regions, respectively. A total population of 41,000 subjects was studied. The observed cancer incidence rates in 1975 were 35 per 10^5 persons in the high background region and 66 per 10^5 persons in the control (low background) region.

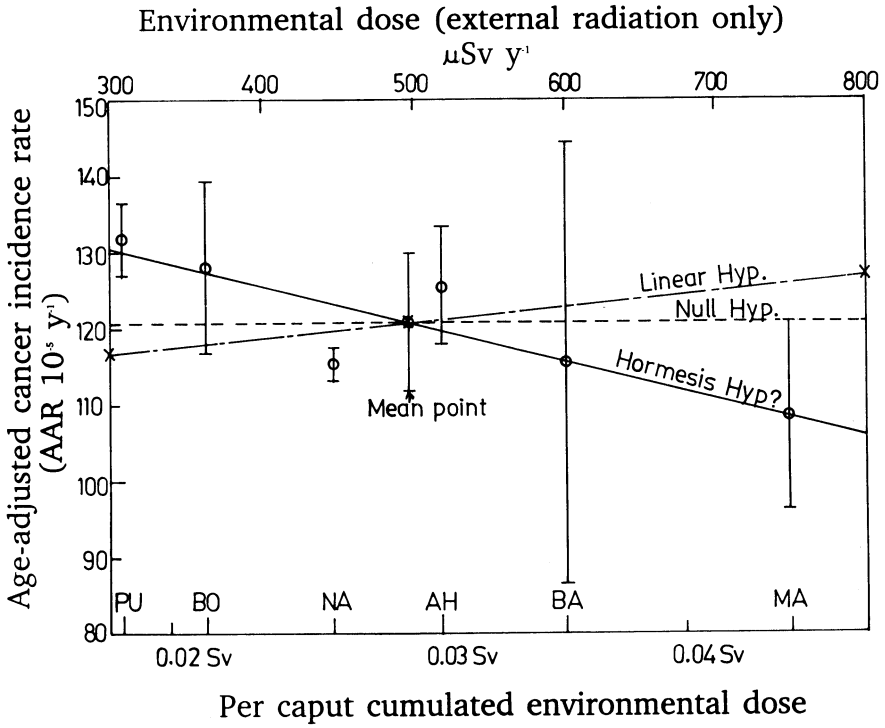


Fig. 15 - Cancer risk vs. background radiation rate in several cities in India

The Japanese survivors again provide useful data. In a 1989 review, the 28,855 persons exposed in the 5 mSv to 50 mSv range (500 mrem - 5 rem) had 108 fewer cancer deaths than expected from the controls. Note that all of these studies, although interesting and suggestive, DO NOT PROVE OR DISPROVE THE HORMESIS THEORY. Population studies are merely clues for scientists to follow-up on. Until the hormesis mechanism is better understood, it will not be possible to say that low levels of radiation unquestionably show hormetic benefits.

How does all of this information fit together? Perhaps a figure from some of Dr. Luckey's work is helpful. Figure 16 shows a plot of radiation response, compared to un-irradiated control mammals, as the dose increases from below background up to very high, lethal levels. ZEP stands for the zero equivalent point which is the upper dividing line between beneficial (hormetic) and harmful effects. It stands at around 0.2 Sv (20 rem) for acute exposures. In the dose region above natural background but below the ZEP, radiation is producing positive health effects. Dr. Luckey estimates that the optimum (point of greatest benefit) dose rate for humans is around 20 to 100 mSv per year (2 rem - 10 rem).

One of the frequently suggested possibilities for the mysterious mechanism that leads to radiation hormesis is that the organism's immune system is triggered by ionizing radiation and thus, through increased readiness, is better able to ward off viruses and/or other infectious agents. Chinese scientists have been focusing particular attention on the effects of low level radiation on the immune system, trying to explain the results of the Guangdong Province study. Residents of the

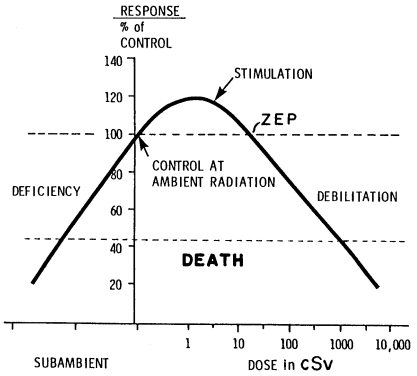


Fig. 16 - Radiation dose-response curve with hormesis

Courtesy T.D. Luckey

high background radiation region have measurably higher levels of T and B lymphocytes and their cells have a higher ability to produce DNA on demand. In animal experiments, continuous low level irradiation produced an increase in the animal's ability to form antibodies and their thymus gland (source of T cells) showed increased activity. Typical cumulative doses for these experimental results ranged from 2 to 100 rem total.

Japanese A-bomb survivors who had received acute doses calculated to be in the 1 to 100 rem range showed higher levels of interferon production compared to the control group. Interferon activates killer T cells, increases the antibody output of B cells and aids macrophages in cell digestion. Interferon is a protein in the recently discovered lymphokine family which is used by the immune system cells to communicate with each other.

Whole Body Radiation Risk

Most of the activities of life carry some risk of injury or death. In an industrialized society, even such mundane actions as breathing the air or having a drink of water carry risks. It is dangerous to work (12,000 deaths on the job in the United States per year) and to stay home (40% of all fatal accidents occur there)!

One way to discuss radiation risk is to compare radiation work with other occupations. Data are readily available on accident rates and fatality rates in various workplaces. Based on figures from the National Safety Council, the loss of life expectancy (shortening of your normal expected life span) at work averages 74 days for all occupations in the United States. Radiation workers, exposed to 5 mSv (0.5 rem) per year, will lose an average of only 40 days. The trade industry is lowest at 30 days. Other occupations, with their loss of life expectancy, are manufacturing (43 days), service industries (47 days), transportation and public utilities (164 days), farming (277 days) and mining (328 days). Dr. Bernard Cohen from the University of Pittsburgh has compiled data on a multitude of common actions that result in risk. He summarizes the data in terms of the average loss of life expectancy from each cause. Figure 17 shows his results.

Another way to discuss radiation risk is to compare the risk of death from a

radiation exposure to the risk of other fatalities. ASSUMING NO RADIATION HORMESIS, an acute exposure to 10 mrem whole body carries a lifetime risk of death by cancer of 1 chance in one million. This same risk is taken if you ride a bicycle for 10 miles, drive 300 miles in your car, smoke 1 1/2 cigarettes or eat 100 charcoal broiled steaks (presumably not all during the same meal).

Dr. Cohen has proposed an additional method for evaluating the risks of radiation exposure. This method involves calculating the amount of money to save a life, assuming various risks in life can be reduced by such expenditures. For example, he points out that "over 5 million childhood deaths could be averted each year at a cost ranging from $50 per life saved from measles in Gambia and Cameroon to $210 per life saved by a combination of immunizations in Indonesia." On the other hand, in the U.S.A., about $200,000 is spent per life saved from cancer by screening programs. This is the same amount that would be needed per life saved by

Cause	Days Lost	Cause	Days Lost
Being unmarried - male	3500	Drowning	41
Cigarette smoking - male	2250	Job with radiation exposure	40
Heart disease	2100	Falls	39
Being unmarried - female	1600	Accidents to pedestrians	37
Being 30% overweight	1300	Safest jobs - accidents	30
Being a coal miner	1100	Fire - burns	27
Cancer	980	Generation of energy	24
20% overweight	900	Illicit drugs (U.S. average)	18
Less than an 8th Grade education	850	Poison (solid, liquid)	17
Cigarette smoking - female	800	Suffocation	13
Low socioeconomic status	700	Firearms accidents	11
Stroke	520	Natural radiation (BEIR)	8
Living in unfavorable state	500	Medical x-rays	6
Army in Vietnam	400	Poisonous gases	7
Cigar smoking	330	Coffee	6
Dangerous job - accidents	300	Oral contraceptives	5
Pipe smoking	220	Accidents to pedalcycles	5
Increasing food intake 100 cal/day	210	All catastrophes combined	3.5
Motor vehicle accidents	207	Diet drinks	2
Pneumonia - influenza	141	Reactor accidents - UCS	2*
Alcohol (U.S. average)	130	Reactor accidents - Rasmussen	0.02*
Accidents in home	95	Radiation from nuclear industry	0.02*
Suicide	95	PAP test	-4
Diabetes	95	Smoke alarm in home	-10
Being murdered (homicide)	90	Air bags in car	-50
Legal drug misuse	90	Mobile coronary care units	-125
Average job - accidents	74	Safety improvements 1966-76	-110

*These items assume that all U.S. power is nuclear. UCS is Union of Concerned Scientists, the most prominent group of nuclear critics.

Fig. 17 - Loss of life expectancy due to various causes

expenditures on improved highway safety, according to U.S. Department of Transportation figures. Moving into the radiation risk side of this issue, it turns out that the U.S. Department of Energy is currently spending between $200 million and $300 million per life saved in radioactive waste management activities at government sites. In the nuclear power reactor field, the Nuclear Regulatory Commission calculates that required tighter safety regulations cost the nuclear utilities about $2.5 billion per life saved. Presumably, this vast difference in the "value of a human life" is due to the unrealistic perception of the hazard of nuclear radiation on the part of some members of the public.

Dose-Effect Models For Radiation Risk

Dose-effect models are a useful way to picture radiation risk. In the past, much argument took place over whether radiation effects required a minimum dose (the threshold hypothesis) or whether small doses produced small risk of injury, i.e., the linear hypothesis. Recent work with human data suggests that some effects clearly show a threshold (e.g., skin cancer incidence, acute radiation sickness, depletion of white blood cells). The 1990 report of the BEIR V Committee concluded that for doses up to 4 Sv (400 rem) cancer mortality follows a linear relationship except for leukemia which follows a linear-quadratic relationship. A linear model means that some radiation-produced effect, e.g., cancer mortality, is calculated from an equation of the form

$$\text{Amount of effect seen} = c\,D$$

where c is a constant of proportionality and D is the dose. A linear-quadratic model means that the radiation effect is calculated from the equation

$$\text{Amount of effect seen} = c\,D + kD^2$$

where c and k are proportionality constants and D is the dose. The BEIR V Committee didn't consider dose-effect models for acute doses less than 0.1 Sv (10 rem), due to the lack of reasonable human data in this low dose range.

Most of the human data used by the BEIR Committee came from studies of the Japanese atomic bomb survivors. The number of survivors which have been followed medically for over 4 decades is 93,000 persons. As of 1982, the following conclusions were reached in terms of those survivors exposed in the lowest dose range:

1) All excess leukemia deaths were in the exposure range above 50 rads.

2) The leukemia death rate was lower than the control group for doses below 50 rads.

3) Excess deaths from solid tumors were all in the exposure range above 10 rads.

4) The mortality rate for solid tumors was lower than the controls for doses less than 10 rads.

The shapes of the various models are shown in Figure 18. Note that in the case of radiation hormesis, the curve first dips into the "beneficial effects" region before reversing and moving up into the harmful effects region. The part of this curve to the left of the lowest point represents the condition of "radiation deficiency."

Researchers in the field of Health Physics continue to argue whether the Linear-Nonthreshold (LNT) dose-response theory is the best one to use for regulatory

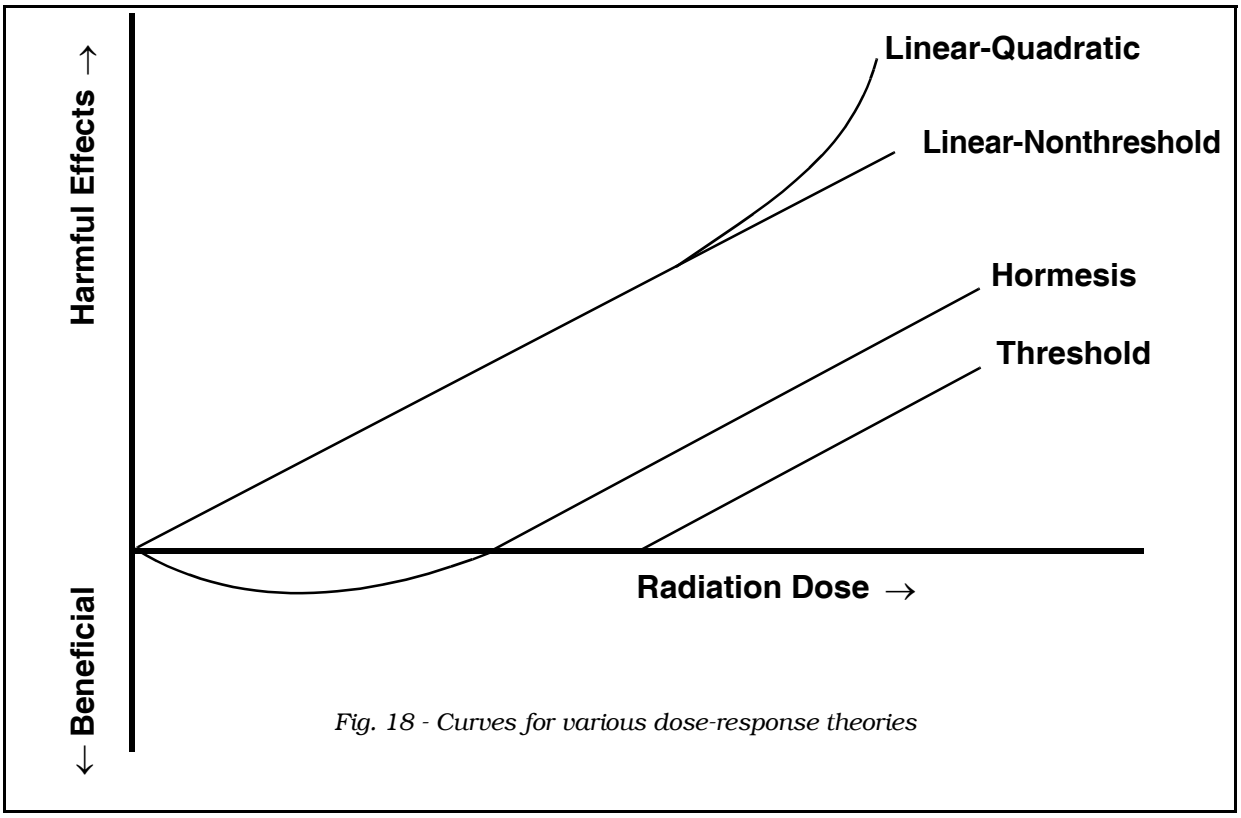


Fig. 18 - Curves for various dose-response theories

purposes. The NCRP undertook a comprehensive review of this subject and issued their findings in 2001 in NCRP Report No.136. They conclude that there is not enough scientific data to reject the LNT Theory and that the bioeffects of low doses near background levels are so insignificant that we'll probably never know!

In 2005, the French Academy of Sciences rejected the Linear-Nonthreshold theory for all cases in which a person is exposed to a radiation dose of less than 100 mSv (10 rem). They noted that about 40% of laboratory studies of radiation effects on cell cultures and animals show radiobiological hormesis.

Post-Irradiation Treatment Of Radiation Injury

Treatment depends on whether internal or external radiation is the primary concern. In the case of internal uptake of radioactive materials, chemicals have been successfully used in three ways. As a blocking agent, the stable element of the radioisotope is administered just before or shortly after uptake. This then causes the body cells to "compete" for the radionuclide atoms. Since the cells can't recognize the stable from the radioactive atoms, they only "choose" a small fraction of the radioactive species if most of the available atoms are in the stable form. As an example, a 130 mg potassium iodide tablet taken within 2 hours of a radio iodine uptake will reduce the thyroid dose by up to 90%. Another method to treat internal uptake is to speed up the normal biological clearance mechanisms in the body. The force feeding of liquids or use of diuretics under a physician's care can speed up excretion of soluble radioactive

contaminants. Finally, chelating agents have been used successfully for heavy transuranic elements such as plutonium and americium. The chelate forms a soluble metal ion complex with the heavy atom and is removed via the kidneys. These techniques and the drugs used are covered more fully in Chapter 14.

The treatment of radiation syndrome following external irradiation is governed by the time course of the symptoms. In the survivable dose range, most treatment focuses on the blood system. Use of antibiotics reduces chances of infection. Transfusions of platelets will prevent the massive hemorrhaging which is fatal. Bone marrow transplants can be used if the dose received is less than the 10 Sv (1,000 rem) maximum survivable dose, although the medical experience gained in caring for the Chernobyl victims has thrown doubt on the usefulness of this technique. Out of 13 transplants, only one was credited with saving a life. As a practical matter, experts now recommend this treatment only for acute doses between 800 and 1000 rem. Use of antibiotics and a sterile environment can prevent death for doses below about 800 rem. Recent experience treating human victims of high dose accidents with white blood cell growth factors manufactured by recombinant genetic engineering shows great promise. The topic of pharmaceuticals for external radiation treatment is covered in more detail in Chapter 14.

Radiobiological Basis For ALARA

One of the more practical applications of radiobiology is in standards setting. What is a safe dose? Unfortunately, there is no comprehensive theory of radiobiology which can answer that question at present. Instead, extrapolations are made from the results of experiments conducted at high doses and high rates. In radiation protection, we assume the worst and most conservative case - the linear hypothesis. Many years ago the philosophy was introduced that radiation doses should be as low as possible. This did not catch on because it became impossible to agree on what was possible. (If funds were available, it would be possible to build lead domes over our cities to reduce cosmic ray exposure). Still, the intent was meritorious. In radiation protection, the 50 mSv (5 rem) per year limit should not be treated as a goal to be achieved but as an upper limit that should not be approached under reasonable circumstances. In 1971, the Atomic Energy Commission restated the principle that exposures be kept “As Low As Reasonably Achievable, economic and social factors being taken into account,” i.e., ALARA! This necessitates a risk-benefit analysis. Ultimately, the need for ALARA is due to the current limitations in radiobiology. It is conceivable that at some future date, a “safe dose” can be defined and ALARA replaced. Until then, the 1973 ICRP statement summarizes the situation: “Whilst the values proposed for maximum permissible doses are such as to involve a risk which is small compared to the other hazards of life, nevertheless, in view of the incomplete evidence on which the values are based, coupled with the knowledge that certain radiation effects are irreversible and cumulative, it is strongly recommended that every effort be made to reduce exposure to all types of ionizing radiation to the lowest possible level.”

Problem Set

1. Describe a free radical. Why is it so reactive chemically?

2. Why is the release of OH free radicals in tissue exposed to ionizing radiation potentially harmful? What might be done to reduce this effect?

3. Why is the radiosensitivity indicator "$LD_{50/30}$" changed to "$LD_{50/60}$" when discussing human radiation effects?

4. Describe each of the structures commonly found in cells. What is the function of each?

5. How does a cell make use of the molecule ATP as an energy supply?

6. List the various cell structures in order of increasing radiosensitivity. What would be the probable result of exposing cells to 1,000 rads?

7. What is meant by the term "chromosome aberration dosimetry?"

8. How is the Law of Bergonie and Tribondeau used to predict cell radiosensitivity? What "practical" radiation protection application does it have? Why does the text refer to it as a "rule of thumb?"

9. Calculate the dose of x-rays that would produce the same effect as 3 rad of fast neutrons if the neutrons have an RBE of 10.

10. Discuss the probable result on the human blood system of an exposure to 650 rem under acute, whole body conditions. What steps might be taken to reduce the effects of these changes in the blood count?

11. Name the four major components that make up human blood.

12. What is the recommended radiation exposure limit for a developing fetus?

13. Describe the probable effects of an acute dose of 950 rem to the small intestine. How would your answer differ if the dose were delivered uniformly over a one-year period?

14. What would be the probable result of a person receiving an acute dose of 16,000 rem of penetrating radiation to the head and neck?

15. What organ in the gastrointestinal tract has the highest radiosensitivity?

16. What is the difference between $LD_{50/60}$ and what the text calls the maximum survivable dose? What are the numerical values of each for humans?

17. Why is a larger dose of 1 MeV gamma rays needed than 150 keV gamma rays to produce skin erythema?

18. Name three different late effects of radiation. For each effect chosen, discuss the possible late effect consequences of an acute dose of 200 rem.

19. Assuming you are representative of the U.S. average, what is the % chance that you will die of cancer? If you receive a single dose of 0.1 Sv, according to BEIR VII estimates, what will be your % chance of dying of cancer? What would the radiation hormesis theory predict in this case? (Sex specific baseline cancer mortality risk in the USA is 22,810 per 100,000 males and 18,030 per 100,000 females.)

20. List some possible radiation produced effects following a human exposure of 50 rem, acute whole body, delivered during pregnancy.

21. Under the same conditions as Problem 20 just above, which effect would be most likely if the radiation were received in the 1st week of pregnancy? The 10th week of pregnancy?

22. Define the term "radiation hormesis."

23. Suggest a possible mechanism for radiation hormesis, i.e., what might be an underlying biological basis for hormesis?

24. What precautions should be observed in the use of radioprotective chemicals for reducing the effects of external radiation exposure?

25. What is the mechanism by which a blocking agent works?

26. Justify the statement, "If we had a comprehensive theory of human radiobiology we could discard ALARA."

S-1. What role do lymphokines play in the body's immune response?

S-2. What is a macrophage and what is its function?

S-3. What are antibodies and where do they come from?

S-4. Radiation produced changes in the DNA bases can be repaired to prevent genetic mutations to future generations of cells. What role do checkpoint genes play in this process?

Other Resources

1. "Radiobiology for the Radiologist," Eric J. Hall, Sixth Edition, Lippincott Williams & Wilkins, Philadelphia, PA, 2005.

2. "Radiation Biology: A Handbook for Teachers and Students, "IAEA Training

Course Series No. 42, Vienna, 2010. Available free at www-pub.iaea.org/MTCD/publications/PDF/TCS-42_web.pdf

3. "Cytogenetic Analysis for Radiation Dose Assessment," IAEA Technical Reports Series No. 405, Vienna, 2001. Viewable at www-pub.iaea.org/MTCD/publications/PDF/TRS405_scr.pdf

4. "The Relative Biological Effectiveness of Radiations of Different Quality," NCRP Report No. 104, Bethesda, MD 1990.

5. "Radionuclide Exposure of the Embryo/Fetus," NCRP Report No. 128, Bethesda, MD, 1998

6. "Evaluation of the Linear-nonthreshold Model for Ionizing Radiation, NCRP Report No. 136, Bethesda, MD 2001.

5

Radiation Quantities and Dosimetry Calculations

Outline of this Chapter

Chapter Summary

Although some of the traditional quantities and units used to measure radioactivity and radiation fields have survived, new recommendations in the 1990s by international organizations and the U.S. NRC have introduced a multitude of new terms. Activity measures the decay rate of a source. It is now expressed in Bq. The familiar exposure, in R, has been laid to rest. It has been replaced by absorbed dose (in Gy), which measures energy actually deposited or by dose equivalent (in Sv) which includes a biological weighting factor. For radiation protection purposes, the relevant external field quantity is the effective dose equivalent which finally allows for those cases when a worker receives only partial body or non-uniform irradiation. In the United States, the 1990s was also the decade where, for the first time, licensees were required to record and limit internal doses from deposited radioactivity. This required the introduction of the committed dose equivalent and CEDE concept. These quantities give the cumulative dose over a 50 year period following the uptake by a worker. Finally, before permanently banishing the roentgen, a useful rem to roentgen conversion factor is introduced to allow the transfer of valuable calibration data obtained under the old system to be used with the new system.

The radiation protection technologist is often confronted with the need to calculate dose rates for a variety of practical situations. Several common cases are dealt with in this chapter. Many times, real gamma ray sources can be approximated by a point source. This allows the use of the simple inverse square law to convert dose rate readings at one location to other distances from the source. A basic working equation is presented to estimate the dose equivalent rate (Sv/hr) at a specified distance from a known activity of a gamma emitter. A supplemental section introduces Bragg-Gray theory which is a method of converting air ionization measurements into tissue absorbed dose rates. Next, some conversion factors are presented to enable neutron field measurements to be translated into dose equivalent rates. In day-to-day practice, this usually means measurement of the fast and the thermal neutron fluxes which are then multiplied by conversion factors and added. Finally, the problem of skin being radioactively contaminated is dealt with by presenting, in Appendix A-2, a useful chart of radioisotopic conversion factors to calculate skin dose. Information is referenced for calculating doses in cases of radiation accidents involving wounds.

Radiation Quantities and Units

Introduction

The interaction effects of a radiation field on some absorber cannot be measured without defining a meaningful set of radiation quantities and units. In technology, a QUANTITY is some physically measurable variable such as length, volume or electrical current. A UNIT, on the other hand, is an agreed upon amount of the quantity which forms the basis of a measurement system. Units appropriate to the quantities mentioned just above could be meter, gallon and ampere. Due to the rapid development of the field of radiation protection technology over the last few decades, a number of basic quantities and units have been introduced and then later discarded in favor of new ones as understanding of both the physical and biological interactions increased. Some of this historical development will be summarized in this chapter.

In 1975, the 15th General Conference on Weights and Measures adopted some new names for certain basic units in radiation protection technology. These new units are consistent with the "metric system" or International System of Units (SI System) developed by the International Committee for Weights and Measures. The International Commission on Radiation Units and Measurements, ICRU, decided on a ten year "phase-in" time period for conversion to the new units. As of 1985, they no longer recognize the old units. In January of 1991, the U.S. Dept. of Commerce published a "Final Rule" requiring federal agencies to use only the SI system after September 1992. In February 1992, the U.S. Nuclear Regulatory Agency issued a proposed "Policy Statement" concurring with this mandate except in reporting accidents and emergency messages. However, even though we are now in the 21st century, little progress appears to have been made toward this lofty objective!

The chief advantage of the SI system is that the "conversion factors" are all unity (1.0). This is offset by the disadvantages that these new units aren't as familiar and that the numerical sizes are different – which leads to confusion. Both the new and old units will be discussed in this chapter, with the main emphasis on the new units. One further complication results from the fact that, just as the USA was finally adopting the "new" radiation quantities accepted by the International Commission on Radiological Protection in their 1977 report, the ICRP changed the names and definitions again! Their 1990 report (ICRP Publication 60) changed long-standing occupational dose limits and introduced new radiation quantities. Most countries have adopted the recommendations of the ICRP so, the latest of them, published in 2007 as ICRP Report 103, will be briefly covered in this text as well.

Activity

The first basic quantity to be dealt with is Activity. As shown in Figure 1, this is merely the rate of radioactive decay of a sample. It represents the number of atoms decaying per unit time and is a measure of the mass of radioactive material in a sample. The old unit for this quantity was based on the measured disintegration rate of

ACTIVITY = Sample disintegration rate
Old Unit: 1 curie = 1 Ci = 3.7 x 10^{10} disintegrations per second = 37 GBq
SI Unit: 1 becquerel = 1 Bq = 1 disintegration per second = 2.7 x 10^{-11} Ci
2007 ICRP Quantity/Unit: Same as above in SI units
Recommended Symbol: A

Fig. 1 - Activity - quantity and units

one gram of the isotope radium-226. It was called the curie, abbreviated Ci, in honor of the discoverer of radium. The agreed upon value for one curie was, as shown, 3.7 X 10^{10} disintegrations per second (dps). Recalling that all conversion factors are 1 in the SI System, the basic unit of activity in the new system is 1 dps. This unit is given the special name becquerel, abbreviated Bq, to honor the discoverer of natural radioactivity. See Sample Problem 1.

Sample Problem 1

GIVEN:
An old calibration certificate for a Cs-137 source claims an activity of 1.3 mCi on today's date, 17 years ago.
FIND:
What is the current activity in Bq and GBq?
SOLUTION:
The radioactive decay correction is $e^{-0.693\ t/T}$ where t = 17 years and T = 30.0 years from Appendix A-1. Thus, today's activity is $e^{-.392}$ x 1.3 mCi = 0.675 x 1.3 mCi = 0.878 mCi. From Fig. 1, 1 mCi = 10^{-3} x 3.7 x 10^{10} d/sec. Therefore, the present activity is 0.878 mCi x 3.7 x 10^{7} d/mCi-sec = 3.25 x 10^{7} d/sec. But 1 d/sec = 1 Bq so the present activity is 3.25 x 10^{7} Bq. Since a GBq = 10^{9} Bq then the present activity is also 3.25 x 10^{7-9} = 3.25 x 10^{-2} GBq.

Antoine Henri Becquerel announced the discovery of spontaneous radioactive decay from uranium in March of 1896. He also reported that the radioactive emissions were charged particles since they, unlike x-rays, were deflected by a magnetic field. The 1903 Nobel Prize in physics was given to Becquerel for this work.

Note that the activity, alone, is not a direct measure of the hazard of a radioactive sample. The relative hazard also depends on the types of radiation emitted in the decays and on the number of such emissions per decay. As an example, working 1 meter away from a 4 terabecquerel (108 Ci) source of cobalt-60 for 8 hours would deliver a whole body dose of about 11.3 Sv (1130 rem) to a person – a lethal dose. Work performed under the same conditions with a 4 TBq (108 Ci) source of hydrogen-3 would produce no effect at all. The low 0.018 MeV maximum beta would be totally absorbed by the air between the worker and the source.

Another quantity that is related to activity is Specific Activity. This is a measure of the concentration of radioactivity. It is commonly expressed in terms of the

activity per unit mass, e.g., Bq/gm or kg. Occasionally, the term is applied to the activity per unit volume, e.g., Bq/cubic meter.

Exposure

The first radiation field quantity to be discussed is exposure. Although no longer with us officially, the quantity played such an important role historically that it deserves coverage. This quantity measured the ability of PHOTONS to produce ionization in AIR. It was finally defined as the sum of the charges of one sign (+ or -) produced by photon irradiation per unit mass of air. The traditional unit of exposure was the roentgen, abbreviated R. The unit is defined in Figure 2.

EXPOSURE = Ability of photons to ionize air
Old Unit: 1 roentgen = 1 R = 2.58 x 10^{-4} Coul/kg_{air}
SI Unit: None purposely chosen. Use is discouraged.
2007 ICRP Quantity/Unit: None purposely chosen. Use is discouraged.
Recommended symbol: X

Fig. 2 - Exposure, the quantity and the unit

Note that there is no new SI unit defined for exposure. This was done intentionally to discourage further use of the quantity after 1985. The three limitations, to be discussed shortly, on the use of the roentgen should clarify the reasons for this action by the International Commission on Radiation Units and Measurements.

The exposure, measured in R, was adopted officially in 1928. It was introduced to replace a quantity that had been unofficially in wide use to measure the output of an x-ray machine. That quantity was the "erythema dose." It was measured by timing the irradiation period necessary to just produce visible reddening on the skin of the hand or arm placed in the beam. This method of determining x-ray intensity had been in use for some twenty-five years. During those years, much unwarranted tissue damage had been produced, particularly to medical practitioners. By converting to an air measurement instead of using human tissue, this unnecessary radiation exposure source was eliminated.

For the sake of historical completeness, the original definition of the roentgen is included here. "The unit of dose is that quantity of roentgen radiation which when secondary electrons are fully utilized and the wall effect of the chamber is avoided produces in 1 cc of atmospheric air at 0° C and 760 mm Hg pressure such a degree of conductivity that one electrostatic unit is measured under saturation conditions." Note that the original definition used the word dose. Under current usage, "dose" is reserved exclusively for measuring energy deposited in matter. This situation was not officially clarified until 1954.

There are three major limitations on the use of the roentgen. These are listed in Figure 3. The limit on the energy is due to the necessity of collecting all of the charge released by photons in the mass of air. At energies above 3 MeV, it becomes impossible to collect all the ion pairs formed by the secondary electrons because their range in air is too great.

The roentgen applies only to photons.
The roentgen applies only in air.
The roentgen is defined only for E < 3MeV.

Fig. 3 - Limitations on the use of the roentgen

It should be realized that none of the three limitations mentioned were of significance in 1928. At that time, virtually all of the radiation sources available for occupational exposure were x-ray machines. Neutrons weren't discovered until four years later, and the multitude of exotic particles produced by high energy nuclear accelerators were not even dreamed of. Even the 3 MeV limit was not a problem. Based on engineering principles, it appeared that potential differences larger than a few hundred thousand volts could never be produced due to the breakdown of the insulators and leakage of charge into the air. In the 21st century, a 3 MeV limit is totally unrealistic. The Large Hadron Collider (LHC) at the European Organization for Nuclear Research, in Geneva, reached a collision energy of 7 TeV (7×10^{12} eV), over 2 million times higher than can be measured by the roentgen. Finally, from the point of view of radiation protection, it is not enough to be able to measure the effects of photons in air. It is necessary to know their effects in human tissue. Due to the differences in composition, tissue interacts differently with radiation than air. For these reasons, a new quantity and unit were introduced to get around the roentgen's limitations. Before discussing that development, another unit and concept related to exposure should be mentioned.

The Specific Exposure Rate Constant gave the exposure rate (e.g., R/hr) at a specified unit distance from a specified activity of a photon-emitting radionuclide. This quantity was useful for estimating the external hazard from a source. The usual symbol was Γ, a capital Greek gamma. It was frequently measured in roentgens per hour at 1 meter from a 1 curie source. If the photon energies emitted by a source are within the energy range from 50 keV to 3 MeV, a convenient rule of thumb will give the correct specific exposure rate constant to within a factor of ± 20%. This rule is given in Figure 4.

$$\Gamma \text{ (R-m}^2\text{/hr-Ci)} = 0.5\ E\ [\pm 20\%]$$

where Γ = Specific Exposure Rate Constant
E = Total photon energy per disintegration (MeV)

Fig. 4 - The Specific Exposure Rate Constant

Even though the roentgen is no longer a recognized unit, the rule of thumb just given is still very useful in operational radiation protection technology. Over a wide range of gamma ray energies, the rem unit is approximately equal to the old roentgen unit numerically. (The exact relationship will be discussed later in this chapter.) Thus, FOR PURPOSES OF ESTIMATION, the dose equivalent rate of a gamma source can be predicted with the equation of Figure 4.

Absorbed Dose

The quantity Absorbed Dose was adopted officially in 1953 as the replacement for exposure. This quantity measures the energy which is actually deposited in some given mass. It is officially defined as the quotient of the deposited energy by the mass. It was originally defined with the unit rad. (The rad "stands for" Roentgen Absorbed Dose). One rad was the dose delivered to anything which received 100 ergs of energy deposited per gram of material. The new SI system unit is the joule per kilogram. In radiation protection work, the "common unit" acknowledged by the SI system is the gray, abbreviated Gy. The internationally agreed upon symbol for absorbed dose is D. These units are defined and interrelated as shown by Figure 5.

ABSORBED DOSE = Energy actually deposited in matter per unit mass
Old Unit: 1 rad = 100 ergs of deposited energy per gram of absorber = 0.01 Gy
SI Unit: 1 gray = 1 Gy = 1 joule per kg (= 100 rads)
2007 ICRP Quantity/Unit: Same as above in SI units
Recommended Symbol: D

Fig. 5 - Absorbed Dose, quantity and units

In 1940, Mayneord suggested that radiation doses should be measured in terms of the energy deposited in a gram of tissue, rather than in roentgens. He proposed the name gram-Roentgen for this new unit. It eventually became the rad as indicated above.

The gray was named after Louis H. Gray, a British physicist with interests in radiation biology and in ionization chambers. Dr. Gray and William Bragg developed the defining theory of ion chamber design. (See supplemental section "Bragg-Gray Theory" later in this Chapter.)

In contrast to the roentgen, the rad or gray can be used to measure all ionizing radiations at all energies and in all absorbers including human tissue. It was heralded as the "universal unit" when it was adopted. Unfortunately, that turned out to be overly optimistic. It did not take into account the RBE of different radiations.

To be strictly correct, it is necessary to specify the absorber when expressing an absorbed dose because the definition does not name the medium (in contrast to the case for exposure which can only be measured in air). In radiation protection technology the absorber is always understood to be soft human tissue unless otherwise specified. However, the preferred term of usage would be "tissue rad" or "tissue gray." In the case of air, the actual physical relationship is such that an exposure of 1 R would produce an absorbed dose of 0.87 air rads. If soft tissue were substituted for air at a point where the exposure to the air had just been measured to be 1 R, the tissue would receive an absorbed dose of 0.95 to 0.96 tissue rads for commonly encountered photon energies. Thus, the former U.S. regulatory assumption of equivalence of 1 R and 1 tissue rad of photons was in error by only 4-5%. See Sample Problem 2.

Soon after the absorbed dose came into use, a problem arose in the field of radiation protection. This was the realization that neutrons seemed to produce more injury per rad than x-rays. Note that the SAME AMOUNT OF ENERGY is deposited in

Sample Problem 2

GIVEN:

An air filled ion chamber shows an exposure of 3.2 R/hr 1 m from a source.

FIND:

What absorbed dose rates, in Gy/hr would be expected for air and for tissue under these same conditions?

SOLUTION:

Since 1 R = 0.87 air rads, the absorbed dose rate in air would be 3.2 R/hr x 0.87 air rad/R = 2.8 air rads/hr. But 1 Gy = 100 rads so the air would receive an absorbed dose rate of 2.8 rads/hr x 1 Gy/100 rads = 0.028 Gy/hr. Similarly, the tissue absorbed dose rate would be about 3.2 R/hr x 0.95 rad/R x 1 Gy/100 rads = 0.030 tissue Gy/hr.

both cases. The difference is in the linear energy transfer of the two radiations. Fast neutrons have an LET that is about 100 times higher than medium energy photons. This results in a pattern of energy deposition along the radiation path which is more damaging to cells. As discussed in the last chapter, biologists use the term Relative Biological Effectiveness to describe this effect. The result is that "a rad is no longer a rad" in radiation protection. In other words, the relative amount of injury to an irradiated person depends both on the energy deposited (rads) and the type of radiation (RBE). Although the attempt was made for a period of time to modify personnel doses by multiplying the dose in rads times the RBE, it became clear that this was not an ideal solution. The RBEs are known to be variables in the sense that repeating an experiment in radiobiology usually gives an answer that is slightly different from the previous experiment. This is due to the many variables introduced by using a complex living organism in experiments. The problem, then, with using an RBE is that the value changes from month to month. Also, the RBE is inherently tied into making comparisons with 250 kVp x-rays. To get around these problems, the radiation protection community proposed the adoption of a new term, the quality factor which is defined independent of x-rays and has a fixed, unchanging value from year to year. The recommended symbol for quality factor is Q. It can be thought of as a modifying factor by which the absorbed dose at a point can be multiplied to determine the risk of biological injury corresponding to the irradiation conditions. Officially Q is defined as a function of LET. Figure 6 shows the ICRP values assigned to Q.

It is possible to calculate a value for Q for a radiation field if the LET distribution is known. The equation that is used is:

$$Q = [A/LET] \times (1 - e^{-B \times LET^{2.03}})$$

where $A = 6000$ **(keV/μ)**

$B = 4.6 \times 10^{-5}$ **(μ²/keV²)**

and the average LET for the field is expressed in keV/μ. The values calculated are correct to within 3% for low LET values and to within 10% for high LET fields.

LET (keV per micrometer)	Q
3.5 and less	1
7	2
23	5
53	10
175 and over	20

Fig. 6 - Quality Factor versus LET

Dose Equivalent [and Equivalent Dose]

Modifying the absorbed dose by the quality factor produces a new quantity, the dose equivalent. The formal definition is that "the dose equivalent, H_T, is the product of the absorbed dose in tissue, the quality factor, and all other necessary modifying factors." Note that the dose equivalent is measured in tissue. Figure 7, shows this relationship and the new and old units. Even though both a gray and a sievert are defined as 1 joule of deposited energy per kilogram, note that the Sv definition includes the modifying factor Q so that the numerical value of absorbed dose and dose equivalent will not be equal unless Q has the value of 1. Note also that 1 Sv = 100 rem in the old system, and that Q is a dimensionless quantity without units.

Rolf M. Sievert was the Swedish scientist honored by the unit for dose equivalent. He was active in the fields of radiation shielding theory and ion chamber theory. For over three decades, he was a member of the International Commission on Radiological Protection, serving as chairperson from 1956 through 1962.

DOSE EQUIVALENT = Biologically weighted absorbed dose
Defining equation: $H_T = D\,Q$
where H_T is the Dose Equivalent in tissue T
D is the Absorbed Dose
Q is the Quality Factor
Old Unit: 1 rem = 100 ergs/gram = (0.01 Sv)
SI Unit: 1 sievert = 1 Sv = 1 joule/kg = (100 rem)
2007 ICRP Quantity: EQUIVALENT DOSE
Defining equation: H_T (Sv) = $\Sigma\, w_R\, D$ (Gy)
where w_R is the radiation weighting factor (replaces Q)
H_T is the Equivalent Dose in tissue T
Recommended Symbol: H_T

Fig. 7 - Dose Equivalent definition and units

As indicated in Figure 7, the 2007 ICRP recommendations threw out the familiar dose equivalent and replaced it with a new term, Equivalent Dose. Besides the word swap, the main change is the elimination of the quality factor. Q has been replaced by w_R named the radiation weighting factor, hopefully not to be confused with the tissue weighting factor, w_T, of earlier ICRP recommendations or the U.S. NRC and U.S. DOE defined weighting factor, w_T, which is defined identically with ICRP but has different numerical values! The need for a new radiation weighting factor was explained by ICRP as the necessity for averaging the absorbed dose over a tissue or organ rather than at a point as resulted from the definition of Q. (Hopefully the reader is less confused than the author at this point!)

As of 2011, the 2007 ICRP recommendations (ICRP Publication 103) are the latest word on quantities and units internationally. They are widely adopted throughout the world, except for the United States. The U.S. NRC is currently satisfied with the terms introduced in the 1994 revision to Title 10, Code of Federal Regulations, Part 20, known affectionately as "10 CFR 20."

The dose equivalent was formally adopted in 1968. It is an ADMINISTRATIVE QUANTITY in that it includes the factor, Q, that is not directly measurable in terms of other physics defined quantities. The old unit for dose equivalent was the rem, an acronym for roentgen equivalent man (or roentgen equivalent mouse or mammal, depending on who tells the story). The new SI unit is sievert, abbreviated Sv.

In order to report measurements in sieverts or rem it is necessary to assign an appropriate value to Q or w_R. As indicated above, these factors depend on the LET distribution of the radiation field. Unfortunately, we are not blessed in radiation protection technology with a "Q meter," a portable instrument that gives the value of Q at some measurement point. However, all is not lost. The regulatory agencies realize that practical measurements of dose equivalent (or equivalent dose) must be able to be made. For practical situations, conservative "average" values of Q and w_R have been determined for common radiation fields and are allowed to be used when the LET is unknown (i.e., almost always).

Up until 1986, the international commissions, NCRP and the NRC all agreed on approximate Q values. Then, the joint ICRU/ICRP task force concluded that enough new biological data was at hand to allow a recalculation of approximate Q values for common field conditions. Not to be outdone, the NCRP followed suit in a 1987 document (Report 91) by recommending a new set of approximate Q values. In 1990, the ICRP abandoned the term Q in favor of w_R. In 1991, new 10 CFR 20 regulations went into effect for U.S. NRC licensees. These regulations continued the old NRC Q values completely unchanged. Figure 8 compares the various sets. Technologists practicing in the United States are legally required to use the "NRC" column values. Technologists in the rest of the world should use the last column, the "ICRP" values for protection calculations. The ICRP values are from their 2007 Report 103. See Sample Problem 3 for an example.

One additional clarification is needed before moving on. For purposes of limiting radiation worker doses in the United States, the actual quantity used for regulatory purposes is the Deep-dose Equivalent, $H_p(10)$. This is merely the dose equivalent at 10 mm tissue depth caused by external exposure. It is the "dose" that is listed as the deep dose on a radiation badge report and was called the "whole body dose" in the good old days! The corresponding former "skin dose" is now officially called the Shallow Dose Equivalent, $H_p(0.07)$.

Organization	NRC	ICRU	NCRP	2007 ICRP
Weighting factor used	Q	Q	Q	w_R
x- and gamma rays	1	1	1	1
Beta rays, except ^3H	1	1	1	1
Tritium beta rays	1	2	1	1
Thermal neutrons	2		5	2.5
Fast neutrons	10	25	20	20 → 7
Relativistic neutrons	3.5		7	7 → 2.5
Hi energy protons	10		1	2
Alpha particles	20	25	20	20

Fig. 8 - Recommended Radiation Weighting Factors

Sample Problem 3

GIVEN:

A technologist is exposed during October to 0.01 Gy of gammas, 2 rads of tritium betas and 0.03 Gy of fast neutrons.

FIND:

What monthly dose equivalent was received if this was a U.S. worker? What equivalent dose if performed in a country accepting ICRP recommendations?

SOLUTION:

In the U.S., the dose equivalent would be H_T = 0.01 Gy x 1 + 0.02 Gy x 1 + 0.03 Gy x 10 = 0.01 + 0.02 + 0.3 = 0.33 Sv using NRC Q factors.

In an ICRP country, the equivalent dose would be H_T = 0.01 Gy x 1 + 0.02 Gy x 1 + 0.03 Gy x 20 = 0.63 Sv.

Effective Dose Equivalent [and Effective Dose]

Under old U.S. standards, no distinction was made between true "whole body irradiation" and irradiation of only a portion of a worker's body mass. Obviously, the risk of harmful effects is higher if the entire body is irradiated. To fix up this apparent discrepancy, the NCRP in 1967 laid the groundwork for a concept later defined formally by ICRP as the Effective Dose Equivalent. (See Figure 9.) The quantity is used to assess the risk to a worker from both uniform whole body and non-uniform partial body exposures. To do this, it makes use of weighting factors, w_T, which take into account the reduced risk of cancer mortality and genetic effects when only some body organs receive a dose. The formal definition is that "the effective dose equivalent (H_E), is the sum of the weighted dose equivalents for irradiated tissues or organs." In 1990, the ICRP also redefined this quantity. They now call it simply the Effective Dose, E, defined in Figure 9. ICRP calls w_T the tissue weighting factor (with different numerical values than NCRP, NRC and DOE) and H_T is the equivalent dose. The table in Figure

EFFECTIVE DOSE EQUIVALENT = Whole body equivalent of partial exposure
Defining equation: H_E (Sv) = Σ w_T H_T (Sv)
where w_T is the NRC weighting factor from Figure 10
H_T is the Dose Equivalent in exposed tissue T
Old Unit: rem
SI Unit: sievert
Recommended Symbol: H_E
2007 ICRP Quantity: EFFECTIVE DOSE
Defining equation: E (Sv) = Σ w_T H_T (Sv)
where w_T is the ICRP tissue weighting factor from Figure 10
H_T is the Equivalent Dose in exposed tissue T
Recommended Symbol: E

Fig. 9 - Effective Dose Equivalent and Effective Dose

10 shows the various currently recommended values for the tissue weighting factors w_T. Sample Problem 4 shows a calculation involving the weighting factors.

It is probably prudent to point out that the differences between the ICRP weighting factors and the NRC/DOE/NCRP factors are actually larger than implied by Figure 10. The weighting factors multiplied by total life-time fatal cancer risk give the probability of dying by a cancer starting in the respective organs. But the more recent ICRP factors use a cancer risk of 5 x 10^{-4} /rem while the other organization's factors were based on a lifetime fatal cancer risk of 1 x 10^{-4} /rem.

Tissue	NRC / DOE / NCRP	Tissue	2007 ICRP
Gonads	0.25	Gonads	0.08
Breast	0.15	Breast	0.12
Red Marrow	0.12	Red Marrow	0.12
Lung	0.12	Lung	0.12
Thyroid	0.03	Thyroid	0.04
Bone Surfaces	0.03	Bone Surfaces	0.01
Remainder (5 organs)	0.30 (0.06 each)	Remainder	0.12
		Colon	0.12
		Stomach	0.12
		Bladder	0.04
		Liver	0.04
		Esophagus	0.04
		Skin	0.01
		Salivary Glands	0.01
		Brain	0.01

Fig. 10 - Tissue Weighting factors

Sample Problem 4

GIVEN:

In the U.S., an average diagnostic x-ray study of the thoracic spine delivers a dose of 0.115 Gy to a patient's thyroid gland and 0.040 Gy to the red marrow.

FIND:

What is the effective dose equivalent from this procedure? Why is it less than 0.115 plus 0.040 Sv?

SOLUTION:

From above, $H_E = \Sigma w_T H_T$ where the weighting factors are from the NRC column in Figure 10. In this case, $w_T = 0.03$ for thyroid and 0.12 for red marrow. Thus, $H_E = \Sigma (H_{E,\text{thyroid}} + H_{E,\text{red marrow}}) = 0.115$ Gy x 1 Sv/Gy x 0.03 + 0.040 Gy x 1 Sv/Gy x 0.12 = 0.0035 Sv + 0.0048 Sv = 0.0083 Sv. This is less than the delivered dose because only part of the whole body was exposed, i.e., this procedure puts the patient at the same risk as 0.0083 Sv delivered "whole body."

Committed Dose Equivalent Family

A final dosimetry quantity needs to be introduced here. This is the concept of a committed dose equivalent which applies to the case of radioactivity internally deposited in a worker. Since 1989 the U.S. DOE has required the recording and limiting of internal dose for a worker along with external doses. Only since 1991 has this been a requirement for NRC licensees (who actually had until Jan. 1, 1994 to fully convert over to this new 10 CFR 20 system). Under most regulations, the quantity Committed Dose Equivalent is given the symbol $H_{T,50}$ and it represents the total cumulative dose delivered to an organ or tissue of the worker for a 50 year time period beginning with the instant of uptake of a radioactive material into the body. See Figure 11. If the particular radioactive material has a short half-life or it is cleared rapidly by the body, the $H_{T,50}$ will be approximately equal to the annual internal dose equivalent for the year of intake. On the other hand, if the half-life is long or it clears slowly, then $H_{T,50}$ is larger than the annual internal dose because it includes dose that will still be delivered to the worker for many future years.

The alert reader may note that something is still missing! As will be discussed in detail in Chapter 9, radionuclides taken into the human body rarely distribute uniformly. Certain organs tend to have much higher concentrations. Thus, it should be clear that, typically, internal doses are once again "partial body" doses. We have a repeat of the problem dealt with in the previous section. The proper way to record partial body doses is to use a weighting factor to report only part of the actual dose delivered to an organ. Remember, this is to make the risk of partial body exposure consistent with whole body exposure risk and not give full weight to radiation exposures that do not irradiate the body uniformly from head to toe.

This brings us to the Committed Effective Dose Equivalent, CEDE, or $H_{E,50}$ (See Figure 11). Here, w_T is the same NRC weighting factor of Figure 10 and $H_{T,50}$ is the Committed Dose Equivalent to an organ, "T." CEDE represents the radiation risk

COMMITTED DOSE EQUIVALENT = 50 year organ dose from internal emitters
Old Unit: rem
SI Unit: sievert
Recommended Symbol: $H_{T,50}$
COMMITTED EFFECTIVE DOSE EQUIVALENT = 50 year body dose from internal emitters
Defining equation: $H_{E,50} = \Sigma w_T H_{T,50}$
Recommended Symbol: $H_{E,50}$ or "CEDE"
2007 ICRP Quantity: COMMITTED EQUIVALENT DOSE
SI Unit: Sv
Recommended Symbol: H_T (50)
2007 ICRP Quantity: COMMITTED EFFECTIVE DOSE
Defining equation: E (50) = $\Sigma w_T H_T$ (50)
SI Unit: Sv
Recommended Symbol: E (50)

Fig. 11 - Committed Dose Equivalent family

to a worker from internal radioactivity that is equivalent in risk to a uniform whole body external exposure of the same numerical size. When added to the deep dose equivalent, H_d or DDE, the sum becomes the Total Effective Dose Equivalent, TEDE. Analogously, the sum of the DDE and an organ committed dose equivalent is the total organ dose equivalent, TODE. These quantities will be discussed in detail in Chapter 9 on internal dosimetry.

For the sake of non-U.S. readers, ICRP Publication 60 (1990) has made slight changes in these quantities as well. They define the Committed Equivalent Dose and the Committed Effective Dose in a similar way to the U.S. definitions by substituting the equivalent dose in place of the dose equivalent in the equations. These quantities are also included in Figure 11.

Roentgen / rem Conversion Factors

Although the roentgen is no longer with us officially, it is still very ingrained in some organizations. Many instrument calibration procedures are still in effect which employ measurements in R. Expensive laboratory secondary dosimeters are often calibrated in exposure rather than absorbed dose, particularly in medical dosimetry applications. Even though exposure is no longer a recognized quantity, it is possible to convert exposure measurements into dose equivalent values. The conversion factor can be calculated using computer models of photon interactions in mathematical phantoms that closely approximate human composition and dimensions. The results of such calculations are shown in Figure 12. Two sets of data are included. The NRC values would be applicable in the United States. The NRC values convert exposure values in R to Deep-dose Equivalent in rem. The ICRP adopted a slightly different set

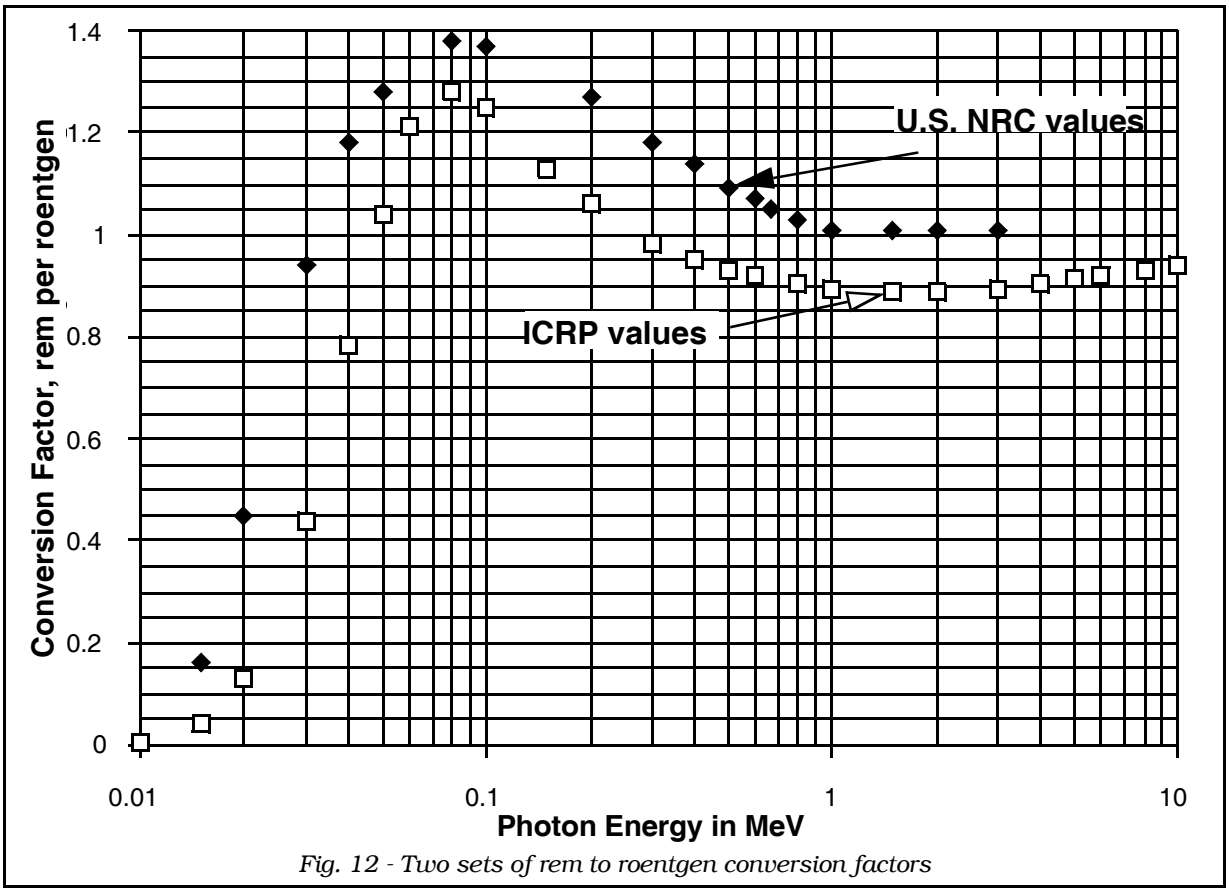


Fig. 12 - Two sets of rem to roentgen conversion factors

which would be used in countries that accept ICRP recommendations.

Sample Problem 5 shows how to use the conversion factors. Finally, Figure 13 is an overall summary review of the multitude of quantities and units introduced in this chapter. It lists the names of the quantities, the units to be used with each, the

Sample Problem 5

GIVEN:

A Cs-137 source used to calibrate personnel badges has a NIST traceable exposure rate calibration. A decay correction to today gives a current exposure rate of 132 mR/hr at 1 meter.

FIND:

What is the correct deep dose equivalent delivered to a badge placed at 1 meter for a 5 minute exposure?

SOLUTION:

From Appendix A-1, the gamma ray energy is 0.662 MeV (listed under the Ba-137m daughter). From Fig. 12, the conversion from R to deep dose equivalent is 1.05 rem/R at 0.662 MeV. Thus, the dose to the badge is

H_D = 132 mR/hr x 1 hr/60 min x 5 min x 1.05 mrem/mR = 11.55 millirem.

organizations which have adopted that quantity and finally a brief description of what exactly is being measured.

Quantity	Units	Used by	Measures
Activity	Bq, Ci	NRC, ICRP	Disintegration rate
Exposure	R	Obsolete	Air ionization by photons
Absorbed Dose	Gy, rad	NRC, ICRP	Energy deposited in mass
Dose Equivalent	Sv, rem	NRC	Biologically weighted dose
Equivalent Dose	Sv	ICRP	Biologically weighted dose
Deep-dose Equiv.	Sv, rem	NRC	Tissue dose at 1 cm
Effective Dose Eq.	Sv, rem	NRC	Partial body dose
Effective Dose	Sv	ICRP	Partial body dose
Committed D. Equiv.	Sv, rem	NRC	Organ dose from internal emitter
CEDE	Sv, rem	NRC	Body dose from internal emitter
TEDE	Sv, rem	NRC	Total of internal + external D.E. Comm.
TODE	Sv, rem	NRC	Total organ dose, internal + external
Equiv. Dose	Sv	ICRP	Organ dose from internal emitter
Comm. Effective Dose	Sv	ICRP	Body dose from internal emitter

Fig. 13 - Summary of quantities & units in radiation protection

Radiation Dose Calculations

Point Gamma Ray Sources

Once the dose rate at some distance from a point gamma ray source is known, the dose rate at other distances is easy to calculate. This is because the radiation intensity follows an inverse square law falloff with distance, a direct result of geometry. Imagine a spherical surface with a 1 meter radius drawn around a point source. Assuming the radiation is emitted uniformly in all directions (i.e., is isotropic) then all points on the spherical surface will have the same radiation dose rate. If the distance away from the source is now doubled by increasing the 1 meter radius sphere to 2 meters radius on a new spherical surface, the SAME NUMBER OF EMITTED GAMMA RAYS PER SECOND FROM THE SOURCE will now be spread over the new larger sphere. Recalling that the formula for the surface area of a sphere is $4\pi r^2$, the surface area of the new sphere will be 2^2 or 4 times larger. Thus, the dose rate at a point on the new sphere's surface will be only one fourth as large as before. Mathematically, this is expressed by saying that the radiation intensity is inversely proportional to the square of the distance from the source, i.e., the inverse square law.

Before dealing with the calculations, it is useful to consider what is actually meant by a "point source." Mathematically, a point is the intersection of two lines and

so it has no size or mass whatsoever. Clearly, no practical radiation source would meet this criterion. In practice, many real sources can be approximated by a point source for purposes of dose calculations. This is valuable because the inverse square law then allows simple calculations of dose rates at various distances from the source. A rule of thumb can be used to decide whether a particular source can be treated as a point.

> RULE OF THUMB: AS LONG AS THE DISTANCE AWAY FROM THE SOURCE IS AT LEAST THREE TIMES THE LONGEST DIMENSION OF THE SOURCE, THEN INVERSE SQUARE LAW CALCULATIONS WILL GIVE THE CORRECT ANSWER TO WITHIN ONE PERCENT.

For example, in the case of a long pipe carrying radioactive fluid, the largest dimension would be the length. The actual dose rate at a point three times the length away located along the perpendicular bisector is only 0.8% different from that calculated using inverse square law.

> **For the advanced reader, it may be helpful to show more quantitatively how this rule of thumb was arrived at. Figure 14 below is a table showing the % error from the inverse square point source approximation compared to the actual dose equivalent rate close to some common "non point" sources. The "actual" dose rates were computed with MicroShield 4, a personal computer based system for calculating dose rates from a large number of user input geometries using Gaussian numerical quadrature integration and energy dependent buildup factors.**

The basic working equation for point gamma ray sources is obtained by combining the rule of thumb for the specific exposure rate constant (Fig. 4) with the inverse square law. The old system form is given first, followed by the SI system form, in Figure 15. This is still ONLY A RULE OF THUMB. It is valid to within ± 20% only over the energy range from 80 keV to 2 MeV due to the energy dependence of the rem/R conversion factor of Fig. 12.

Further comment on the expression for Γ is in order. The factor of

Distance as % of longest dimension	Sphere	Cylinder	Line
50	41.3	16.1	-22.6
100	4.5	1.8%	-7.9%
150	1.7%	0.6%	-4.1%
200	-2.1%	0.0%	-2.1%
300	0.0%	-0.2%	-0.8%
400	-0.8%	-0.8%	-0.8%
500	0.0%	0.0%	0.0%

Fig. 14 - % error in using inverse square law close to a "large" source

Exposure Rate (R/hr) = X/t

$= \Gamma A / r^2$

$= 0.5\ A\ E / r^2$ [±20%]

where A = Source activity in Ci

E = Total photon energy (MeV/disintegration)

r = Distance from point source (m)

Γ = Specific Exposure Rate Constant = 0.5E (Figure 4)

and

Dose Equivalent Rate (Sv/hr) = H_T/t

$= \Gamma A / r^2$

$= 0.15\ A\ E / r^2$ [±20%]

where A = Source activity in TBq

E = Total photon energy (MeV/disintegration)

r = Distance from point source (m)

Γ = Specific Exposure Rate Constant

Fig. 15 - Exposure & dose equivalent rate from a point gamma source

0.15 for Γ in the SI System was obtained by multiplying together the linear absorption coefficient for air, the rem/R conversion factor and several physical constants to convert the energy deposited in air to R/dis. The results of this computation are shown graphically in Figure 16. Here, the SI System conversion factor Γ is plotted vs. gamma ray energy. As can be seen, the

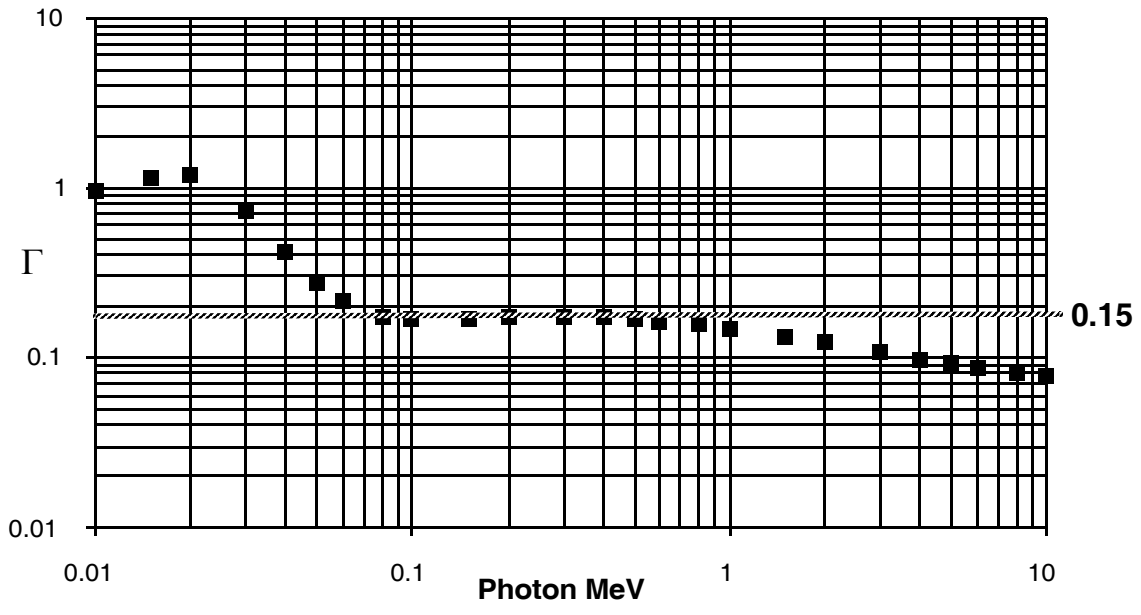


Fig. 16 - SI System dose equivalent rate constant, Γ, vs. photon energy

0.15 is a "compromise" value in that between gamma energies of 80 keV and 2 MeV the factor varies between 0.12 and 0.18, i.e., 0.15 ± 20%.

Sample problem 6 demonstrates the use of the point gamma source equation. Note that while 100% of the Co-60 decays produce a pair of gamma rays (at 1.17 and 1.33 MeV), only 85% of the decays of Cs-137 result in a 0.662 MeV gamma ray. From Appendix A-1 it can be seen that 100 Cs-137 disintegrations produces 95 Ba-137m radioactive daughters. In the next Appendix A-1 table entry, it is seen that 90% of the Ba-137m daughter disintegrations leads to a 662 keV gamma ray. Hence, for the original 100 Cs disintegrations, we get 95 Ba-137m atoms disintegrating to yield 85 gamma rays (90% X 95 disintegrations).

For the "old timers" in the reading audience, the point gamma source equation is a close relative of the familiar "6CE rule." This rule states that the exposure rate at one foot from a gamma emitter is equal to the number of curies, C, times the gamma energy in MeV. Then, $6CE/r^2$ is the exposure rate at any distance, r, in feet from the source. By changing the distance to meters, the "6" is converted to "0.5" for exposure rate or "0.15" in the case of dose equivalent rate.

Although many practical situations requiring a knowledge of the dose equivalent rate can be solved using the Figure 15 rule of thumb approximations just discussed, there are occasions when higher accuracy is needed. A common example would be the need to calculate the dose equivalent rate at some distance from a

Sample Problem 6

GIVEN:

Two sealed gamma ray sources of 0.40 Ci of Cs-137 and 0.40 Ci of Co-60.

FIND:

What is 1) the total dose equivalent rate and 2) the total exposure rate at r = 2 meters from the two sources?

SOLUTION:

For part 1) of the question, from Fig. 1, the source activities are A = 0.40 Ci x 37 GBq/Ci x 1 TBq/1000 GBq = 0.0148 TBq each. For Cs-137, E = 85%/disintegration x 0.662 MeV = 0.56 MeV/dis. For Co-60, E = 100%/disintegration x (1.17 + 1.33) MeV = 2.5 MeV/dis.

Thus, for Cs-137, H_T/t (Sv/hr) = 0.15 A E / r^2 = 0.15 x 0.0148 TBq x 0.56 MeV/dis ÷ $(2\ m)^2$ = 3.1 x 10^{-4} Sv/hr. For Co-60, DE rate = 0.15 x 0.0148 TBq x 2.5 MeV/dis ÷ $(2\ m)^2$ = 1.4 x 10^{-3} Sv/hr. The total DE rate = 1.7 x 10^{-3} Sv/hr = 1.7 mSv/hr.

For part 2), the exposure rate is given by X/t (R/hr) = 0.5 A E/ r^2 where the A is in curies now. For Cs-137, X/t = (0.5 x 0.4 x 0.56)/2^2 = 0.028 R/hr. And for Co-60, X/t = (0.5 x 0.4 x 2.5)/2^2 = 0.13 R/hr. Finally, the total exposure rate at this distance = 0.16 R/hr. This result is slightly smaller numerically than the part 1) result since the rem/R conversion factor is slightly larger than 1 for the average of these gamma energies (see Fig. 12).

sealed gamma ray source being used to calibrate portable radiation safety survey meters to meet regulatory requirements. In this case, the ±20% error in the dose equivalent rate is already twice the uncertainty allowed for the entire procedure (error in the calibration source plus error in the instrument response). When more accuracy is needed, the technician must revert to tables of published values for the actual measured results of specific exposure rate constant determinations for particular gamma ray sources. Values are available in several publications, including the **Radiological Health Handbook**. Caution should be observed when using some of the published results, particularly the older studies, as some of these neglected to include low energy x-ray emissions from the respective sources. A limited list of gamma rate constants, for both the old and new unit systems, is shown in Figure 17. The SI system values were obtained using the NRC values for the rem to R conversion factor. These values do include all photon emissions from the sources.

Radionuclide	Ave Energy (keV)	Γ (R-m^2/hr-Ci)	Γ (μSv-m^2/hr-Bq)
Iodine 125	28.5	0.145	3.0×10^{-8}
Americium 241	59.5	0.0129	4.0×10^{-9}
Cobalt 57	123.5	0.0965	3.0×10^{-8}
Cesium 137	662	0.323	8.0×10^{-8}
Radium 226	800	0.824	2.0×10^{-7}
Cobalt 60	1250	1.32	3.1×10^{-7}
Sodium 24	2061	1.85	4.4×10^{-7}

Fig. 17 - Gamma rate constants for selected radionuclides

Bragg-Gray Theory

There are times in radiation protection when it is necessary to know the absorbed dose in tissue rather than the exposure rate in air. The Bragg-Gray Equation provides a technique for calculating the tissue absorbed dose from a measurement of the exposure in air. It should probably be emphasized again that the passage of a photon through an absorber does not deposit energy directly. Dose results when photo electrons, Compton electrons and electron-positron pairs are released in the absorber. These charged particles then cause ionization and excitation that results in an absorbed dose.

Consider a ping-pong sized ball of solid frozen air (see Figure 18, A). The charged particles released by photons will have a range of a few mm. Next, picture a small hollowed out cavity in the center (Figure 18, B) filled with air in the gas state. Bragg and Gray stated that if the solid air wall around the cavity was thicker than the range of the released charged

Fig. 18 - Frozen air masses for Bragg-Gray discussion

particles and if the cavity were small enough then the number of electrons per unit volume at points within the solid air wall would be unaffected by the cavity. Since the cavity and the wall are identical in chemical composition (both air) then the energy deposited PER UNIT MASS will be equal in the wall and the cavity and thus:

Dose to the wall = Dose to the cavity.

Now the dose to the gas in the cavity is directly proportional to the number of ion pairs formed and inversely proportional to the mass of the gas. Figure 19 shows the actual calculation.

$$\text{Dose}_{\text{cavity}}\ (\text{rad}) \propto \frac{N(\text{ion pairs}) \times W(\text{erg/ion pair})}{m(\text{gram})} \propto \frac{N\ W}{m}$$

Fig. 19 - Dose to the cavity gas

Making use of Figure 2 from this chapter, the exposure of the air filled cavity, i.e., the charge per unit mass, can be expressed as shown here in Figure 20.

$$\text{Exposure, X} \propto \frac{N(\text{ion pairs}) \times e(\text{Coul/ion pair})}{m(\text{gram})} \propto \frac{N\ e}{m}$$

Fig. 20 - Exposure to the cavity gas

Thus, as shown in Figure 21, the dose to the air in the cavity is directly proportional to the exposure measured in roentgens of the cavity gas, which in turn is now proportional to the dose to the frozen air wall.

The final step in this process is to make the equation useful for practical dosimetry. To do this, the frozen air wall is usually replaced with a tissue equivalent plastic wall. Since the wall composition is now different, the

$$\text{Dose}_{\text{cavity air}}\ (\text{rad}) \propto \frac{X\ W}{e}$$

$$\text{So, Dose}_{\text{wall}} = \text{Dose}_{\text{cavity air}} \propto \text{Exposure}_{\text{cavity air}}$$

Fig. 21 - Relationship between cavity and wall doses

energy deposited per unit mass by the electrons released by photon interactions will be larger or smaller than in the frozen air depending on the relative MASS STOPPING POWER between the frozen air and the plastic. Recall from Chapter 3 that the stopping power of a material measures the energy deposited per unit distance. Thus, the mass stopping power measures the energy deposited per unit density thickness. The mass stopping power ratio will show the relative rates of energy deposition in the two materials. This fact was expressed by Bragg and Gray in the form shown in Figure 22.

$$\text{Dose}_{\text{plastic wall}} / \text{Dose}_{\text{air wall}} =$$
$$\text{Mass Stopping Power}_{\text{plastic wall}} / \text{Mass Stopping Power}_{\text{air wall}}$$
$$= s$$
$$\text{So, Dose}_{\text{plastic wall}} = s \times \text{Dose}_{\text{air wall}}$$

Fig. 22 - Dose ratio for different wall materials

In Figure 22, the symbol s represents the ratio of the mass stopping powers in the two materials. Combining this with the previous figure gives the famous Bragg-Gray Equation, Figure 23.

$$\text{Dose}_{\text{plastic wall}}\ (\text{tissue rads}) \propto \frac{s\ W\ X}{e}$$

Fig. 23 - Final form of the Bragg-Gray Equation

Through the use of this equation, an exposure or exposure rate measurement in air can be converted into an accurate dose or dose rate in tissue. This equation is widely used in the design of ionization chamber radiation detectors, especially for medical dosimetry in hospital radiology departments.

Neutron Dosimetry

As mentioned in Chapter 3, neutron monitoring poses one of the most difficult situations for a radiation protection technologist. As a result of the fact that neutron survey instruments which read directly in dose equivalent rates (mrem/hr) are both expensive and bulky to handle, occasionally in radiation protection we need to convert a neutron flux reading from a “fast/slow” neutron counter into a dose equivalent rate. The fast/slow neutron survey meter itself will be discussed in Chapter 7. It is a portable instrument which has two configurations for operation. In one, it reads slow neutron flux (neutrons per square cm per second) and in the other configuration it reads the fast neutron flux. Before dealing with the practical problem of interpreting

the meter readings, some additional background information on neutron dosimetry is in order.

As a result of the energy dependence of the various reactions by which neutrons deposit energy in human tissues, and due to the fact that the biological injury depends on the LET of the radiation, it is necessary to know the energy distribution of a neutron field to correctly determine the dose to a person exposed to that field. This means that in addition to a knowledge of the numbers of neutrons per unit area and time (the flux) the energies of each of those neutrons must also be known.

If the flux and energy distribution can be measured, it is then possible to estimate the average dose equivalent rate (mrem/hr) for tissue as the dose is usually uniformly distributed for neutrons. This is because the dimensions of a person are generally small compared to the neutron mean free path (average distance of travel between collisions) and because most dose producing reactions lead to charged particle releases which travel only a mm or two in tissue, hence causing "local deposition" of energy. Doses calculated in this way were sometimes called a "first collision dose." A more modern term is Kerma, an acronym for Kinetic Energy Released per unit MAss. Kerma is a precisely defined term used for describing energy deposition by charged particles released by indirectly ionizing radiations (e.g., neutrons and photons).

Returning to the practical radiation protection problem, it is necessary to come up with a "conversion factor" to relate the measured fluxes to dose equivalent rates. As mentioned above, this conversion factor will depend strongly on the neutron energy. That is, some energy neutrons are more damaging than other energies. As a result of careful measurements and with the aid of computer models, the ICRP has established values for the required conversion factor vs. neutron energy. Based on this data, the Nuclear Regulatory Commission has adopted a table of conversion factors which are to be used for radiation protection purposes "if there exists sufficient information to estimate with reasonable accuracy the approximate distribution in energy of the neutrons." This table is published in the Code of Federal Regulations, Title 10, Part 20.1004 (abbreviated 10 CFR 20.1004). Using data from the table, calculated flux to dose equivalent rate factors for some selected energies are given in Figure 24.

For practical purposes, a radiation protection technologist ordinarily uses only two of the entries in this table, entries which properly correspond to the "fast flux" and "slow flux" on the fast/slow survey instrument.

Clearly, the first table entry, "thermal," is the proper one for the slow flux conversion. That is, 272 slow neutrons per square cm per second will produce a dose equivalent rate of 1 mrem/hour or 0.01 mSv/hr. The choice of a proper conversion factor for the fast neutron flux is more difficult. Even if the primary neutron energy is known, scattering from surroundings and attenuation by shielding and air will rapidly cause the overall average neutron energy to degrade to lower energies. On the other hand, only the highest energy neutrons have reasonable probability of penetrating shielding. These are also the most damaging to tissue, as is seen from Fig. 24. It takes 272 thermal neutrons to cause the injury to tissue produced by only 7 neutrons of 10 MeV. In other words, the 10 MeV fast neutron is 39 TIMES MORE INJURIOUS TO TISSUE PER NEUTRON (272÷7) THAN A THERMAL NEUTRON. Finally, since the

Ave Neutron Energy (in MeV)	Q	Ave Flux in n/cm^2-sec to deliver: 1 mrem/hr	1 mSv/hr
Thermal	2	272	27200
0.0001	2	233	23300
0.001	2	272	27200
0.01	2.5	281	28100
0.1	7.5	47	47200
0.5	11	11	10800
1.0	11	7.5	750
2.5	9	8	806
5.0	8	6	639
10	6.5	7	667
100	4	6	556

Fig. 24 - US NRC values for flux to dose equivalent rate conversions

meter response of the fast/slow instrument falls off (gets less sensitive) as neutron energy increases, it is prudent to adopt a conservative assumption about the "fast flux." This is done by assuming the worst case, namely, all neutrons detected with the instrument in the fast configuration are assumed to be greater than 10 MeV and thus, have a conversion factor of 1 mrem/hr per 6 n/cm^2-sec, the last table entry. The numerical calculation in Sample Problem 7 should clarify the use of these factors.

Note that the conversion factors already include the quality factor. This is evident from the units used - rem instead of rad. Additional information on neutron field measurements will be presented in later chapters.

Sample Problem 7

GIVEN:

A technologist measures a neutron field having a fast flux of 20 and a slow flux of 900 n/cm^2-sec.

FIND:

What is the total deep-dose equivalent rate due to neutrons at that location?

SOLUTION:

Fast H_D/t = 20 n/cm^2-sec x (1 mrem/hr per 6 n/cm^2-sec)
= 3.3 mrem/hr.

Slow H_D/t = 900 n/cm^2-sec x (1 mrem/hr per 272 n/cm^2-sec)
= 3.3 mrem/hr

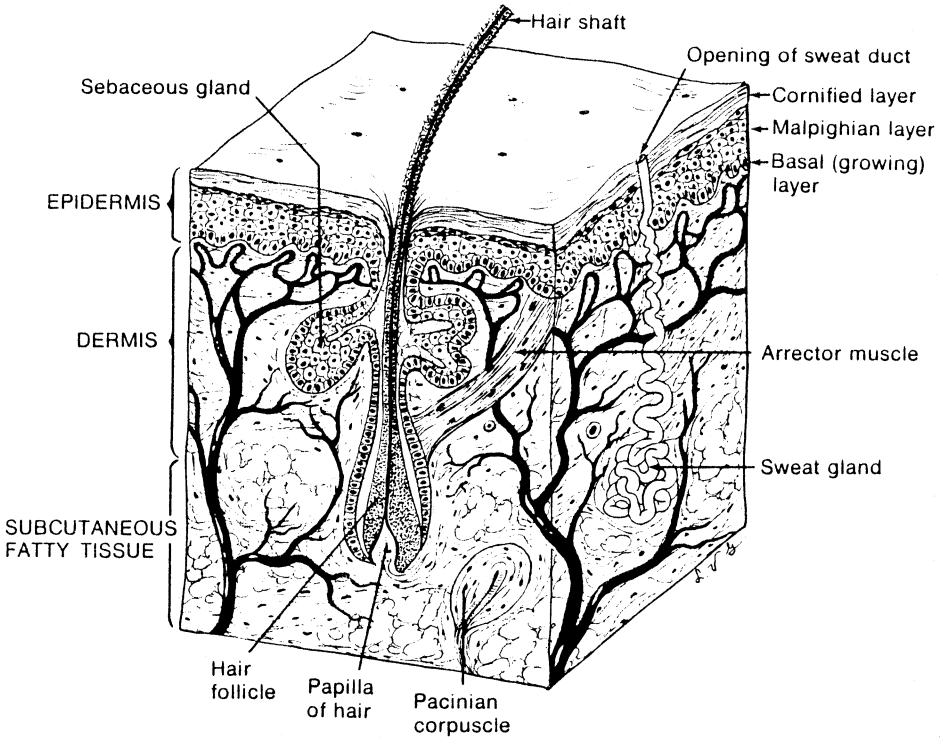
So, the TOTAL dose equivalent rate = 6.6 mrem/hr or 0.066 mSv/hr.

Skin Dosimetry

Another vexing problem for radiation protection technologists is the estimation of dose received by skin as a result of the presence of radioactive surface contamination. Practically, this problem in beta dosimetry is very evident in the nuclear power industry, or on decontamination project sites, where such contaminating events are fairly common. Note that the dose due to beta rays totally dominates the picture. In Chapter 3 it was pointed out that alphas are stopped by the dead layer of skin so they do not produce a "skin" dose, i.e., deposit energy in the growth layer. Photon emissions present in mixed beta-gamma contamination only contribute a tiny fraction of the total dose received by the skin.

Many attempts have been made in the past to come up with reasonable estimates of the dose. This is complicated by two factors. People have not yet been able to agree on a definition of "skin" and the physics problem of determining what fraction of energy is deposited at different depths of tissue, particularly for beta particles entering from all angles (not just simple perpendicular incidence), is formidable.

For years, the ICRP has defined "skin," for radiation protection purposes, to be the basal cell layer of the epidermis. This is a region of active growth at the bottom of the outer layer of skin, as shown in Figure 25. There are two additional recognized layers to the skin. The dermis lies beneath the epidermis and the subcutaneous fatty tissue is the deepest skin layer. The problem with the definition is in figuring out how



From Textbook of Physiology, 18th Ed., Schottelius and Schottelius, C.V. Mosby Co., 1978.

Fig. 25 - Anatomical details of the skin

deep the basal cell layer is. Skin thickness varies considerably over the body. For example, the palms of the hands and soles of the feet are much thicker than facial skin. In 1991 ICRP published a comprehensive report devoted entirely to this problem. They recommend that, for radiation protection purposes, the basal cell layer should be considered to be covered by a dead epidermis 20 to 100 microns thick when stochastic effects (skin cancer) are the concern. The corresponding density thickness of this layer is 2 to 10 mg/cm^2. In preventing non-stochastic, late cosmetic effects in skin, the appropriate depth is increased to 300 to 500 microns (30 to 50 mg/cm^2). They then recommend different dose limits for these two categories of possible effects. In the U.S., the NRC has settled on the single value of 70 microns (7 mg/cm^2) depth, averaged over an area of one square centimeter, for measuring the shallow dose equivalent to worker's skin.

According to many investigators, a more accurate dose is found by taking into account the location of the contamination and then using the appropriate skin thickness for that location. Unofficially, the following thicknesses are often used in these calculations:

Head, trunk, upper arm, upper leg: 4 mg/cm^2
Lower arm, wrist, back of hand, lower leg, top of foot: 8 mg/cm^2
Palm of hand, sole of foot: 40 mg/cm^2

As mentioned above, the physics of the beta interaction is difficult to describe with a simple model that would allow hand calculation of reasonable answers. An elegant solution to this dilemma was published by D. Kocher and K. Eckerman from the Oak Ridge National Lab in 1987. Making use of a complex computer model developed by Berger at the National Bureau of Standards, they were able to calculate the actual energy deposited (dose) to the basal cell layer, shielded by 4, 7, 8, and 40 mg/cm^2 of epidermis, from some 200 different beta emitting radionuclides deposited on the skin surface. Their results, shown in Appendix A-2, are expressed in sieverts/year per becquerel/cm^2, but the numerical results can easily be converted to the "old units" with the following conversion:

Listed Dose Rate Factor X 422 = rad/hour per μCi/cm^2

The radionuclides were selected to be representative of potential contaminants from nuclear reactor accidents, radioactive wastes, natural radioactivity and medical uses. The listed factors do not include any gamma ray or alpha particle dose contribution to the skin. Also, the dose DUE TO RADIOACTIVE DAUGHTERS is not included. Thus, to find the total dose in that situation, the Dose Rate Factors for parent and daughter must be added. Sample Problem 8 illustrates the use of this method of skin dosimetry.

As noted at the beginning of this section, skin dose calculations usually are required as a result of accidental surface contamination over some substantial area of skin. There is, of course, another situation which has had much attention focused on it in the past. This is the "hot particle" problem seen in the nuclear power industry. The problem and its solution are dealt with at length in National Council on Radiation Protection and Measurements Report 130 (See "Other Resources," this Chapter). A concern which arose shortly after the discovery of "hot particles" was the possible need to change current dose limits for skin in this instance. The concern was that, due to the extremely small size of "hot particles," only a very small region of the basal skin layer would be receiving the dose, but that dose could be very high if the problem

Sample Problem 8

GIVEN:

A university biochemist working with a C-14 labeled compound splashes some of the liquid on his face. Six hours later, a contamination monitor shows a single tiny spot of 29 μCi.

FIND:

a) What is the best estimate of the skin dose to this worker?

b) What dose would be reported on the worker's personal dose record?

c) Why are these two dose results different?

SOLUTION:

Conversion to SI units gives 29 μCi x 3.7 x 10^4 Bq/μCi = 1.1 x 10^6 Bq. As noted above, skin doses are calculated as an average over 1 cm^2. Thus, the source in this problem is 1.1 x 10^6 Bq/cm^2.

a) The best estimate would be made using a skin density thickness of 4 mg/cm^2. From Appendix A-2, the dose rate factor is 7.9 x 10^{-3}. Thus, the dose, for the 6 hours elapsed exposure time, is H = 7.9 x 10^{-3} Sv/yr per Bq/cm^2 x 1.1 x 10^6 Bq/cm^2 x 1 yr/8766 hr x 6 hr = 5.8 Sv.

b) The legal dose requires calculation at the "average" skin depth of 7 mg/cm^2. The dose rate factor at this depth is 2.9 x 10^{-3}. Thus, the legally reportable dose would be 2.9/7.9 times as large or 0.37 x 5.8 Sv = 2.1 Sv.

c) The legal dose uses an average depth for regulatory simplicity but a better estimate can be made if actual skin thickness is taken into account.

was not recognized soon after contact with the "hot particle." This situation might increase the risk of future skin cancer. The 1991 ICRP report on skin also addressed this issue. They concluded that "experimental studies give unequivocal evidence that the cancer risk is not increased by very non-uniform irradiation." Thus, "hot particles" only cause a risk of skin cancer that is proportional to the dose they deliver, just like any other source of skin dose.

A final topic closely related to skin dose is the problem of estimating dose when radioactive material is injected through intact human skin, i.e., wound dosimetry. Published records indicate that well over 2,000 cases have been handled historically, most of them involving fingers punctured with radioactive material involved. Health physicists have had calculational models and computer software available for decades to deal with ingested or inhaled radioisotopes. The complexity of the wound problem has now, finally, been dealt with. A special cooperative venture formed between the National Council on Radiation Protection and Measurements and the International

Commission on Radiological Protection spent several years reviewing human and animal data on radionuclide behavior in wounds, examining the biology of the wound healing process and investigating techniques for practical radiation monitoring of wounds. All of this information was then published, in 2006, as NCRP Report Number 156, **Development of a Biokinetic Model for Radionuclide-Contaminated Wounds and Procedures for Their Assessment, Dosimetry and Treatment.** If in your practice of Radiation Protection Technology you encounter a radioactive wound case, be sure to consult the NCRP report for guidance in this exceptional situation.

Problem Set

1. What is the name, and abbreviation, for the latest internationally accepted set of quantities and units?

2. Name one advantage and one disadvantage of the "new" system of radiation quantities and units.

3. How many disintegrations per minute are equal to one curie? Why was this number chosen to be so large?

4. Calculate the number of curies in a terabecquerel (TBq).

5. What exactly was being measured by the quantity exposure? What unit was used?

6. Name three limitations on the use of the unit roentgen. Why were these limitations not considered worrisome when the unit was officially adopted?

7. Why is it improper to state the exposure (measured in R) received by a person standing in a photon radiation field?

8. What term is used to describe "energy deposited per mass?"

9. Calculate the absorbed dose in Gy to the skin of a person which receives 1500 ergs/gm of beta radiation from a cloud of radioactive krypton-85 gas. Recalculate the dose if the same energy were deposited by the betas from a xenon-133 gas cloud.

10. Why is it improper to record only the absorbed doses received by a person on a personnel radiation history?

11. Define the term "quality factor." What is its value for Co-60 gamma rays? for reactor produced thermal neutrons? for tritium beta particles?

12. What is the total dose equivalent of 1 rem of fast neutrons, 0.5 rad of low energy betas, 25 ergs per gram of thermal neutrons and 150 mR of diagnostic x-rays?

13. A survey meter has been accurately calibrated in mR/hr. The measured value is then used directly by a technologist in Texas to determine the length of time a radiation worker can remain near a large sealed Am-241 gamma source. What problem, if any, will this cause? How would you answer this question if the source were changed to Co-60?

14. Under what conditions could a gamma ray "check source" be considered a point source? What is the advantage of being able to approximate a source by a point source?

15. What accuracy should be expected when the exposure rate is calculated for a point source using the formula of Figure 15?

16. Approximately how many Bq would be needed to produce an exposure rate of 10 mR/hr at 10 cm from a Cobalt-57 source? (Co-57 decays 86% of the time by emitting a 122 keV gamma ray).

17. Calculate the specific exposure rate constant for Cs-137 and Co-60. How do the calculated values compare with the accepted values of 0.32 and 1.3 R/hr-Ci @ 1m?

18. Name at least three reasons why it is difficult to accurately measure the dose equivalent rate due to a neutron field.

19. Why are the first and last entries in the table of conversion factors shown in Fig. 24 the most useful in practical radiation protection technology?

20. Approximately how much more tissue damage would be inflicted by a 2.5 MeV fast neutron compared to a 100 keV intermediate energy neutron?

21. Calculate the fast neutron flux needed to produce the same injury as 0.50 mSv/hr of thermal neutrons, and then, as 6,000 thermal neutrons per sq. cm per sec.

22. Why was the quantity Dose Equivalent introduced into radiation protection? What is the recommended symbol for dose equivalent?

S-1. What are the two design conditions necessary before an ion chamber can be considered a Bragg-Gray chamber? What could such a chamber be used for?

S-2. Why was the earliest definition of the roentgen incorrect when it named the roentgen as a unit of dose?

S-3. A research technician working with Sr-90 solution manages to contaminate the back of his hand by leakage under his rubber glove. It is estimated that he had 25 μCi per cm^2 present for 2 hours before discovery and immediate cleansing. Calculate the

dose to his skin.

S-4. The text points out that the dose equivalent, in rem or Sv, applies to all radiations, all absorbers and all energies. Why is it then of use to define the Effective Dose Equivalent and the Committed Dose Equivalent?

S-5. Based on multiple badges attached to a worker who enters a high radiation area, the worker has received only a partial body exposure. The dosimetry shows 0.037 Sv to the lung and thyroid glands. Calculate the effective dose equivalent for this situation.

Other Resources

1. "Fundamental Quantities and Units for Ionizing Radiation," Report Number 60, International Commission on Radiation Units and Measurements, Bethesda, MD, 1998.

2. "Recommendations of the ICRP," Publication #103, International Commission on Radiological Protection, Pergamon Press, New York, 2007.

3. "SI Units in Radiation Protection and Measurements," Report #82, National Council on Radiation Protection and Measurements, Bethesda, MD, 1985.

5. "Biological Effects and Exposure Limits for 'Hot Particles'," NCRP Report 130, National Council on Radiation Protection and Measurements, Bethesda, MD, 1999.

6. "The Radiological Health Handbook," U.S. Department of Health, Education and Welfare, Document #PB 230846, 1970. (Still available from the National Technical Information Service, Springfield, VA).

7. "Development of a Biokinetic Model for Radionuclide-Contaminated Wounds and Procedures for Their Assessment, Dosimetry and Treatment," NCRP Report 156, National Council on Radiation Protection and Measurements, Bethesda, MD, 2006.

6

Radiation Sources

Outline of this Chapter

Chapter Summary

Radiation Sources is the final chapter in the Radiation Protection Theory Unit of the book. The properties of a multitude of natural and artificial radiation sources are described, usually along with some estimate of the radiation dose levels being delivered to either workers or the general population.

Natural sources are frequently considered to consist of three components. The rocks making up the crust of the earth contain low concentrations of radioactive minerals which produce an external radiation field at the surface. Some long-lived natural radionuclides also find their way into our bodies via drinking water, food intake or through inhalation of radioactive gases and particulates. Finally, the human population continues to be constantly bombarded by radiations originating in deep space, i.e., the cosmic ray component. All three of these natural background sources are explored and quantified. This section then concludes with information on radiation in space that is relevant to space flight operations, and with background information on the complicated problem of Naturally Occurring Radioactive Material or NORM.

Artificial sources constitute the bulk of the chapter. Fallout from atmospheric nuclear testing is the first topic. Consumer products that emit radiation when energized or contain radioactive materials are the next two topics. The emphasis here is on the genetically significant doses delivered to the population from these sources. The physics and engineering of x-ray production leads to a discussion of x-ray machines used in medical applications and in industrial applications. The extreme hazards associated with misuse of this equipment are shown. Another medical topic explores the use of radioactive materials in both the diagnosis and treatment of human disease. The basic principles involved in nuclear medicine and in radiation oncology are discussed and illustrated. Attention then shifts to nuclear particle accelerators. Following an analysis of the physics principles involved, the engineering aspects are exemplified by descriptions of various medical, industrial and research facilities using accelerators. Two topics on nuclear reactors are the next order of discussion. First, civilian uses of research/training reactors and nuclear power reactors are described. Basic reactor physics principles are followed by material on reactor designs in use around the world. Then, the nuclear reactors used in the production of nuclear weapons are examined.

The chapter concludes with a brief discussion of some miscellaneous industrial radiation sources. Topics include neutron sources based on radionuclides, neutron and gamma ray sources used in oil and gas exploration and intense gamma sources used in industrial sterilization.

Natural Radiation Sources

Introduction

In previous chapters the nature of radiation has been examined, along with the mechanisms that deposit energy in absorbers in general and living tissues in particular. The concepts and quantities of dosimetry have been covered along with their units. Attention is now turned to the multitude of radiation sources that exist naturally within our environment or have been introduced as a result of our technology. Each source will be described as to its operating principles and then in terms of the expected accompanying radiation fields.

In 1987, the NCRP issued Report 93, which was the first comprehensive summary of data on all sources of radiation exposure to individuals in the United States. The overall total average Effective Dose Equivalent to the population was found to be 3.6 mSv (360 mrem) per person. At the turn of the century, it was realized that the Report 93 numbers needed to be updated, primarily due to the large increase in medical procedures involving radiation exposure of patients. In 2009, NCRP Report 160, **Ionizing Radiation Exposure of the Population of the United States**, was released to supersede the 1987 work. The exposure categories used were natural background, occupational, consumer products, medical and miscellaneous. The reporting year was chosen to be 2006. The table in Figure 1 compares the 2006 effective dose results with the corresponding values reported for the 1980s.

The dose increased 170% in the two decades between the studies (from 3.6 up to 6.2 mSv). Virtually all of the increase was due to the medical category - it rose by 565%! It has been suggested that this may be a result of US physicians practicing

Radiation Source	Ave Annual H_E (mSv) to Entire U.S. Population in the 1980s	Ave Annual H_E (mSv) to Entire U.S. Population in the year 2006
Natural Background		
Radon	**2**	**2.1**
Other	**1**	**1.0**
Medical	**0.53**	**3.0**
Occupational	**0.009**	**0.005**
Consumer Products	**0.13**	**0.13**
Miscellaneous	**0.001**	**0.003**
ROUNDED TOTAL	**3.6**	**6.2**

Fig. 1 - Sources of background radiation in the USA, 1980s and 2006

"defensive medicine," i.e., ordering tests to reduce the risk of lawsuits rather than in the best interest of patient health.

Exactly half of the U.S. average effective dose to the entire population is now due to natural sources. The breakdown shown in the pie chart of Figure 2 illustrates this. If the trend documented by NCRP Report 160 continues, medical exposure of the US population will soon exceed all other radiation sources combined!

Three natural environmental "background radiation" sources will now be detailed - the rocks in the crust of the earth, radionuclides in air, food and water, and cosmic radiation. It is well to note that this natural component, which accounts for 50% of each person's total annual radiation dose, has been irradiating people since the beginning of time. Also, note that there is no habitable place on the earth, above it, or below the surface which is totally radiation free.

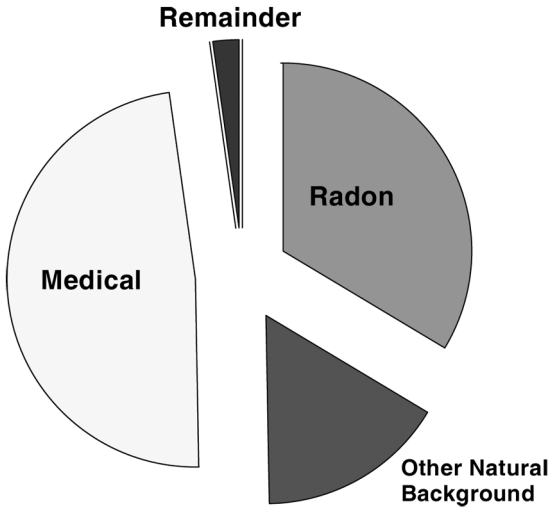


Fig. 2 - Contributions to U.S. population background radiation

External Terrestrial

The term terrestrial radiation sources refers to radionuclides which are deposited in the crust of the earth. Some of these radioactive materials decay through photon emissions which penetrate the overlying rocks and cause a gamma ray field at the earth's surface. There are a large number of naturally occurring radionuclides which are found in trace amounts in rocks. The composition changes with location and type of rock. The largest contributions to the external gamma ray field are usually from the radioisotopes Ra-226, U-238, Th-232 and K-40. The last one listed is a long-lived isotope of potassium which has a natural abundance of 0.01%. The others are all heavy metal nuclides with very long half-lives. Although the concentrations in the rocks are in the parts per million range, a substantial quantity of the nuclides are present due to the vast extent of the earth's crust. In the United States, on the average, a square mile of surface dug out to a depth of one foot would contain one ton of K-40, three

tons of U-238 and 6 tons of Th-232.

The type of rock which makes up the crust at some location is a major factor in determining the exposure rate of the external field. Generally, igneous rocks (rocks which are of volcanic origin) have the highest concentrations of radionuclides. In the USA, surface exposure rates average 125 mR/yr over igneous rocks. Sedimentary rocks usually show much lower exposure rates. These rocks result from the settling of organic material to the bottoms of lakes and oceans which compresses over geologic time spans to produce layered rocks. U.S. sandstone averages 50 mR/yr while limestone deposits give an exposure rate of only 25 mR/yr. Based on actual measurements, the average person in the USA is exposed to a level of 40 mR/yr from external gamma rays originating in the earth's crust.

It should be clear from the radiobiology chapter that a dose rate of 40 mrad/yr (or 5 microrad per hour) will not be expected to ever produce acute somatic radiation injury. Even the probability of late effects is almost negligible at these low rates. On the other hand, genetic mutations are transmitted on to our offspring who will then be exposed during their lifetimes. Thus, over many generations the cumulative effect on genetic mutations might show a very slight increase due to background radiation. For this reason, a concept which is more meaningful for use when discussing background levels in populations is the average genetically significant dose (GSD), rather than the average dose. The GSD includes only the fraction of the radiation which actually deposits energy in the gonads (ovaries or testes) of persons of childbearing age. The shielding effects of surrounding body tissues and the structures in which we usually live and work must be taken into account. That means only part of the external field will reach the gonads. Then, the gonad doses received must be further reduced by a "weighting factor" which is the probability of the exposed individual producing more offspring. This number can be calculated from birth records. The values for males and females shown in Figure 3 are from U.S. population figures. The result of performing the GSD calculation for external terrestrial radiation is that the U.S. average GSD from that source, in 1982, was 28 mrem/yr. Unfortunately, the NCRP

Age Group	Expected # (Males)	Expected # (Females)
0 - 14	2.6	2.4
15 - 29	2.7	2.5
30 - 44	1.1	0.6
45 - 64	0.06	0.02

Fig. 3 - Expected number of future children, USA, 1970

decided to stop reporting GSD results, "because the GSD is not a dose quantity in current use by . . . NCRP." It is sad to see it phased out.

Before leaving the subject of rocks, it is interesting to note that there have been several areas identified around the globe where this external terrestrial component of natural radiation is much higher than in the USA. Two of the more well known areas are the Kerala region along the

southwest coast of India and the Minas Gerais state in northeast Brasil. Both of these regions contain relatively high concentrations of monazite, a thorium mineral. The population of the Kerala region is about 70,000. About 16,000 persons receive an annual dose in excess of 5 mSv/yr. Approximately 500 people receive a terrestrial radiation dose of over 20 mSv/yr. This value is about 50 TIMES HIGHER THAN THE U.S. AVERAGE. The highest reading recorded was 58.65 mSv/yr, a dose rate higher than the maximum permitted a U.S. radiation worker and more than 250% the limit allowed under the ICRP 2007 recommendations of Publication 103. The population in Minas Gerais near the monazite deposits is only a few hundred persons total. The average terrestrial background radiation level there is 16 mSv/yr with a maximum measured rate of 120 mSv/yr (this is 18 times higher than the average for exposed workers in all U.S. nuclear power stations).

Internal Terrestrial

Sometimes natural radionuclides end up in food or water used for human consumption or become airborne as radioactive gases or particulates. In turn, some fraction of these becomes deposited internally in the body. This so-called body burden produces a dose as the radionuclides decay in place inside the body where they have bound themselves to various tissues.

There are about 70 different naturally occurring radioisotopes which contribute to the internal genetically significant dose. Most of the dose is due to K-40 (about 80%) with the remainder due chiefly to nuclides from the heavy radioactive series. The U.S. average GSD from all of the internal emitters, estimated in the 1987 NCRP report, is 36 mrem/yr. (Radon delivers 210 mrem/yr U.S. average effective dose equivalent but radon produces 0 GSD as the exposed organ is the lung.)

As was the case with the radiation level from rocks, there are situations in which much higher levels of natural internal radioactivity are encountered. Occasionally drinking water supplies have high concentrations of radium. In the U.S., measured levels of radium and other alpha emitters are usually below 1 pCi/l of water. In the Colorado plateau, these concentrations can be as high as 40 - 50 pCi/l. In Brasil, commercially bottled mineral waters have radium concentrations as high as 240 pCi/l. Finally, a small private school near Slinky, Finland has such a high level of radium in the water supply that the permanent staff receive an annual lung radiation dose of 240,000 mrem due to inhalation of the radon gas produced by the decay of the radium.

Cosmic Radiation

The source for the third natural component to the background radiation field, cosmic rays, is “out of this world.” Some of them are of extra-galactic origin (shades of Star Wars!!). The highest energy measured as of 2010 was a proton at 3.2×10^{20} eV. A much lower energy component originates from our closest star, the sun, during solar flares. Measurements by satellite mounted cosmic ray telescopes during the late

1990s demonstrated that, except for those with energies above 10^{15} eV, most cosmic rays originate in our Milky Way galaxy, in supernova remnants. Also, it was determined that the heavier elements found in cosmic rays are indeed ejected from supernova explosions.

Astrophysicists have been speculating for years about the source of the highest energy cosmic rays. As of 2010, both theory and measurement appear to have zeroed in on the answer - the nuclei of active galaxies, an AGN (Active Galactic Nucleus). These special galaxies contain a supermassive black hole that is gobbling up huge amounts of nearby matter. This incredible energy release produces two jets of protons shooting out in opposite directions - the ultra high energy cosmic rays.

At the top of the earth's atmosphere, the cosmic rays are composed of 87% protons (hydrogen nuclei) and 12% alpha particles (helium nuclei). The remaining 1% consists of other light nuclei. This composition again supports the theory of their origin in the stars. Most of the mass of stars is hydrogen and helium which undergo fusion reactions to power the star. By the time the rays have filtered down through the shielding provided by several hundred miles of air, the composition has totally changed. At sea level, cosmic rays consist of 63% mesons, 15% electrons and 21% neutrons. The mesons are released when the high energy primaries collide with the nitrogen and oxygen nuclei making up the atmosphere. The energy that is transferred to the target nuclide greatly exceeds the total binding energy of the nuclide so it literally explodes in a shower of protons, neutrons and mesons.

The annual external dose rate from cosmic rays depends slightly on latitude and strongly on altitude. The latitude effect is due to the charged particle nature of the primary cosmic rays. As the particles approach the earth, its magnetic field tends to deflect the rays away from the equator and toward the poles. The annual dose rate increases about 8% in going from the equator to 50 degrees north latitude (approximate northern border of USA). The strong altitude dependence is shown in Figure 4. It is a direct result of the reduced shielding with increasing altitude. The values given in Figure 4 are external radiation field dose equivalent rates and do include the neutron component. These numbers were taken from the 1987 NCRP report on U.S. background radiation levels. The NCRP estimates a 10% structural shielding factor is appropriate for the energies involved. The result gives a population averaged U.S.

Altitude (ft)	Dose Rate (mSv/yr)	Example
Sea Level	0.31	Los Angeles
5,000	0.55	Denver
10,000	1.37	Leadville, CO
30,000	19.0	Normal airplane
50,000	87.5	SST airplane
80,000	122	Spy plane
120,000	105	Space Shuttle

Fig. 4 - Altitude dependence of cosmic ray dose rate

Sample Problem 1

GIVEN:

The 100,000 persons in Berkeley, CA are relocated to Leadville, CO.

FIND:

Estimate the upper limit of the number of lifetime cancer deaths "caused" by the move to higher elevation.

SOLUTION:

Assume an average remaining life expectancy of 35 years per person. From Fig. 4, the annual change in radiation dose is 1.37 – 0.31 = 1.06 mSv/yr. So the lifetime added dose is 35 yr x 1.06 mSv/yr = 37 mSv. Since cancer risk figures are not available in this text for continuous exposure, the upper limit (worst case) will be found by assuming the 37 mSv is instantaneous and then using the risk estimates of Chapter 4, Fig. 14. If we assume equal numbers of males and females, the lifetime risk is 696 excess deaths / 0.1 Sv. Thus, this population could show up to 696/0.1 Sv x 0.037 Sv = 257 excess deaths from the higher cosmic radiation level. Compared to the normally expected 20,360 cancer deaths (from Chap. 4), the cosmic rays might cause an increase of up to 257/20,360 = 1.3% in the cancer mortality rate of this city.

Genetically Significant Dose of 0.28 mSv/yr from all cosmic radiation. See Sample Problem 1.

During solar flares, which occur with a periodic 11 year cycle time, the dose rates at high altitude can become colossal. The largest solar flare measured to date reached 0.1 Sv/hr (10 rem/hr) at an altitude of 80,000 feet and was 1 mSv/hr (legally, a high radiation area for radiation workers) at 30,000 feet. A smaller 1989 solar storm produced a level of 80 μSv/hr at 30,000 feet, over 450 times higher than "normal." Because of the possibility of solar flares, supersonic transport aircraft are required to have an operating radiation detector connected to an alarm in the cockpit to warn of radiation levels above 0.5 mSv/hr. If the alarm sounds, the aircraft dives to a lower altitude to complete the flight. As of 1998, the Concorde had a 20 year history of 100,000 flights and had never been forced to descend to lower altitudes because of a radiation alarm.

Airline flight crew members appear to be among the higher dose ranges among various occupations with radiation exposures. A 1990 report by the U.S. Federal Aviation Agency gave dosimetry data for flight crews. Regular runs between New York and Chicago deliver about 5 mSv/year, while cross-country trips average about 6.5 mSv/yr. If a flight crew regularly flies a transatlantic route, they can receive about 9 mSv/yr. This is higher than the average for exposed workers at U.S. nuclear power plants.

The 1998 Annual NCRP Meeting focused on airline crew and astronaut cosmic ray exposure. The primary contributor to dose is galactic cosmic rays, while solar cosmic rays play only a minor role. The dose rate for this flying population depends on altitude, geomagnetic latitude (measured from the magnetic pole, not the geographic

pole) and, to a limited extent, on the phase of the 11 year solar flare cycle. The makeup of the radiation field is quite unique - up to 2/3 of the dose is from high LET radiation, mostly neutrons. The dose rate at the magnetic poles is double the rate at the magnetic equator. The 2009 NCRP Report 160 provided some updated estimates for average doses to flight crews. By assuming crews worked 90% on domestic flights and 10% on international runs, the overall average effective dose was 3.07 mSv for the year 2006. (Cockpit crews averaged 2.19 mSv while flight attendants received 3.76 mSv due to longer working hours.)

Studies are underway to look for an excess cancer incidence in airline flight crews. There is a fairly large number of such persons, but the relatively low doses will make definitive studies difficult. Out of a concern for air crew radiation doses, all European air crew members were classified as occupational radiation workers as of 2000. The individual doses will be measured, special radiation safety training is mandated and female crew can receive special consideration during pregnancy.

Radiation in Space

The earliest radiation safety guidelines for U.S. space missions were adopted by the National Aeronautics and Space Administration (NASA) from a 1970 report of the National Academy of Sciences. These allowed astronauts to receive a career limit of 4 Sv with special organ limits of 6 Sv to the eye, 12 Sv to the skin and 2 Sv to the testes. A 1989 report of the NCRP reexamined the latest data on exposures from previous missions, updated data on the space radiation environment and the trend toward many more women astronauts. Newer lifetime career guidelines limit females in space to 1 - 3 Sv and males to 1.5 - 4 Sv depending on age at first exposure.

Generally, space radiation is divided into three categories. Trapped particle radiation refers to the Van Allen belts of charged particles (electrons and protons) caught by the magnetic field of the earth. The electron belts extend out to 48,000 miles while the proton belt is of concern only for low earth-orbiting missions. Galactic cosmic radiation refers to the fairly uniform radiation field caused by protons (87%) and alpha particles (12%). The remaining 1% is, as mentioned earlier, composed of ions called HZE particles for “high Z and high energy.” Iron is of most concern as an HZE particle. It tends to leave a core of dead and incapacitated cells along the track taken through the astronaut’s tissues. This is of particular concern relative to central nervous system tissues and the retina of the eye.

The last category of space radiation is solar particle events. These are bursts of mainly hydrogen and helium ions from our sun during solar storms. The frequency of the storms follows the eleven year sunspot cycle. Dose rates are high enough to produce acute radiation effects. Therefore, on long duration missions, special heavily shielded “cellars” are planned for the spacecraft.

When working in space, the radiation levels encountered vary dramatically depending on length of the mission, altitude, inclination from the earth (due to the Van Allen belts) and amount of shielding in the spacecraft. The NCRP estimates that personnel on a space station at 450 km altitude will receive doses of about 1 mSv (100 mrem) per day. A spacecraft in geosynchronous orbit at 35,800 km altitude would receive about 1,600 mSv/year. To provide a bit more insight into doses received

Mission Series	Mission Dose(mGy)	Daily Ave(mGy)
Gemini	0.36	0.17
Apollo, lunar	4.6	0.50
Skylab	44	0.68
Space shuttle 1981-86	1.24	0.20

Fig. 5 - Average doses received by U.S. astronauts on space missions

by astronauts, the table in Figure 5 shows average values of doses for some U.S. space mission series. The "mission" doses are for the entire duration of a flight. The Apollo series flights lasted from 6 to 12 days. The space shuttle missions listed were typically 5 to 8 days in length. Currently, plans are being made for a possible 2.5 year mission to Mars. Doses for such a trip are estimated to be 63 mSv to 130 mSv depending on the solar cycle.

Much research remains before a valid quality factor can be assigned to the dosimetry data of space missions. In low earth orbit, high LET radiation accounts for about 1/3 of the dose equivalent rate. In deep space, this fraction rises to 86%. As of 2011, most of the data is from missions in low earth orbit. NCRP Report 153 from 2006 puts out a call for more research in radiation biology, radiation physics and in characterizing the radiation fields that will be encountered in deep space. They point out that their current recommendations for exposure limits in space missions apply only to low earth orbit. Recommendations for a return to the moon or to Mars have yet to be issued.

Naturally Occurring Radioactive Material (NORM)

Since the beginning of the atomic age, certain radioactive substances have remained outside of the regulatory framework designed to protect workers and the general public. Naturally occurring radioactive material or NORM falls in this category. Until the late 1980s, it was assumed that NORM was of such low concentrations that the associated radiation fields and radioactivity did not pose a threat to public health. In 1986, significant radiation field levels were discovered in tubing used in oil and gas production in the state of Mississippi. Subsequent to that time, surveys have been conducted of oil fields and associated production facilities across the U.S. Results indicate that NORM poses a potential problem primarily along the Gulf Coast, in northeast Texas, southeast Illinois and southern Kansas. To date, only the states of Louisiana and Texas have enacted regulations relating to NORM. The Louisiana regulations were adopted in 1989 while the Texas standards became effective in 1993. There are no specific federal laws covering the generation, storage, transport, or disposal of NORM. The Conference of Radiation Control Program Directors has published its own suggested set of NORM regulations for adoption by other states. However, little action has resulted.

At this point, a clarification is in order. The radioactive materials just mentioned above are actually designated "Technologically Enhanced NORM" or TENORM.

The U.S. EPA defines this as material which "contains radionuclides that are present naturally in rocks, soils, water, and minerals and that have become concentrated and/or exposed to the accessible environment as a result of human activities." Thirteen states have regulations to control TENORM as of 2010. An additional 9 are considering TENORM regulation.

Although NORM in the oil and gas production industry was the first to stimulate serious interest in health effects of natural radioactivity, NORM is found in other activities as well. The chief radionuclides of concern are radium-226, radium-228, their daughter products (lead-210 in particular) and potassium-40. In addition to oil and gas operations, some of these nuclides are found in phosphate and potash fertilizers, phosphogypsum used in agriculture, building materials, highway construction materials and, last but not least, Brazil nuts (high K-40 levels).

Returning to oil production, the radium isotopes are the chief concern. They are carried to the surface by well production fluids and usually concentrate in the water phase. The radium then precipitates into sludges and pipe and valve scale. The levels of NORM in pipe scale can reach thousands of pCi/gm while in the sludges associated with well operation the levels have been measured to several hundred pCi/gm. Contaminated soil is also a potential problem, particularly in areas where pipe cleaning operations are carried out to remove scale from well tubulars.

Measurements of the exposure rate at one foot above the ground at a pipe-reaming yard in Louisiana gave an average of 250 μR/hr and a maximum of 1800 μR/hr. Ra-226 concentrations in the soil were elevated down to a depth of about 12 inches.

In gas production and distribution facilities, the chief problem is associated with radon, particularly Rn-222. Among the daughter products of its decay is found lead-210, an alpha, beta and weak gamma emitter with a 22 year half-life. The lead-210 will "plate out" on the inside surfaces of pipes and valves in an invisible thin coating. Since the pipe wall removes 100% of the alphas and betas, it cannot be located by external measurements under realistic conditions.

Additional information on handling, surveying, transporting and disposing of NORM contaminated equipment and wastes from oil and gas production will be covered in later chapters. As more states join in regulating this radiation source, the true scope of the problem should become more evident.

Artificial Radiation Sources

Fallout

Atmospheric testing of nuclear devices has released large quantities of radioactive material (fallout) into the biosphere. Devices detonated near the earth's surface will take up large amounts of soil which becomes radioactive through neutron exposure. A 1 megaton device can activate up to 50,000 tons of debris. The total activity content of the remains of the device plus the activation products is shown in Figure 6.

During the 1950s, the U.S. had an active atmospheric testing program for nuclear devices which was carried out at the Nevada Test Site. Between 1951 and 1958, 119 tests were carried out at the NTS, most of them detonated at or above the surface. The U.S. declared a moratorium on testing between 1958 and 1961.

Following the signing of the Limited Test Ban Treaty, on August 5, 1963 in

35 kCi of Carbon-14	**70 MCi of Iodine-131**
3 x 10^{11} Ci of Sodium-24	**140 kCi of Cesium-137**
17 MCi of Strontium-89	**100 kCi of Strontium-90**
3,600 Ci of Plutonium-239	

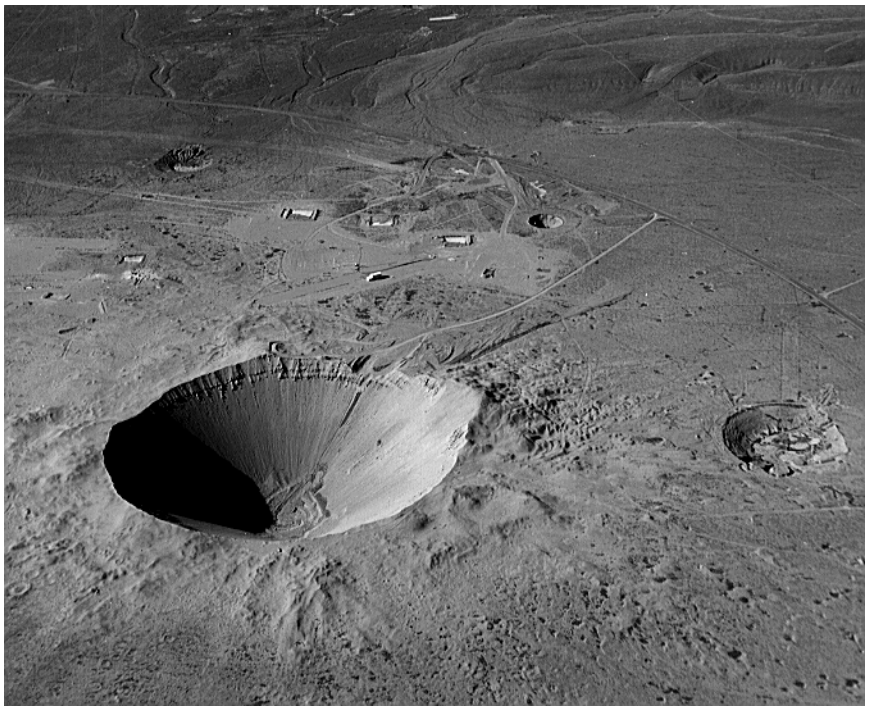
Fig. 6 - Fallout from a 1 megaton nuclear device

Moscow, the U.S. conducted only underground tests of nuclear devices. These have all been carried out at the Nevada site. Most of the tests were in the Yucca Flat area, shown in Figure 7. More than 200 craters dot the valley. The nuclear explosion creates an underground cavity when the rock is vaporized. When the cavity collapses, it leaves the surface formation known as a "subsidence crater." The diameter and depth of the crater are dependent on explosive yield, burial depth and the local geology. The Sedan crater is pictured in Figure 8. It was formed by a 100 kiloton device buried 635 feet deep in 1962. The resulting subsidence is 320 feet deep and a quarter of a mile in diameter. In 1992, the US halted all atmospheric and underground testing of nuclear weapons. This moratorium remains in place today.

By detonating the devices underground, radioactivity is, in principle, prevented from reaching the atmosphere. Since the 1970 Baneberry test, no significant venting has occurred in the U.S. underground program. The 10 kiloton Baneberry test accidentally vented to the surface, as shown dramatically in the photo of Figure 9. This

Dept. of Energy photo

Fig. 7 - Nevada Test Site, Yucca Flat area



Dept. of Energy photo

Fig. 8 - The Sedan crater

resulted in the creation of an "international incident." The plume was tracked as it was driven by atmospheric air currents until it crossed the border between the United States and Canada. Procedures were changed following Baneberry to minimize the possibility of ever venting to the atmosphere again.

For the history buffs in the reading audience, some photos are included of the early atomic bombs, Figures 10 and 11. The Hiroshima device, "Little Boy," was a gun barrel which fired a chunk of ^{235}U into another uranium mass. It was surrounded with a steel shell. The Nagasaki device, "Fat Man," had spherical sections of ^{239}Pu which were imploded by a thick shell of high explosive. The first atomic bomb was detonated at Alamogordo, New Mexico on July 16, 1945. The test, code named Project Trinity, was a 20 kiloton imploding plutonium device similar to Fat Man.

While active atmospheric testing was in progress, the U.S. average GSD from external fallout was about 3 to 4 mrem/yr. In addition, adults received about 13 mrem/yr internal dose due primarily to the isotopes Sr-89, Sr-90 and Cs-137. In growing children, thyroid doses up to 80 mrem/yr were measured as a result of intake of milk containing I-131. In 1987, the NCRP estimated that the annual dose equivalent rate to the general population had declined to less than 1 mrem/year from fallout. They also estimated the U.S. population average total dose commitment from the onset of atmospheric testing through the year 2000. The results gave 75 mrem/person from external whole body exposure. Internally deposited radionuclides gave 25 mrem/person whole body, 112 mrem/person to bone and 40 mrem/person to lung.

Dept. of Energy photo

Fig. 9 - The Baneberry Test venting

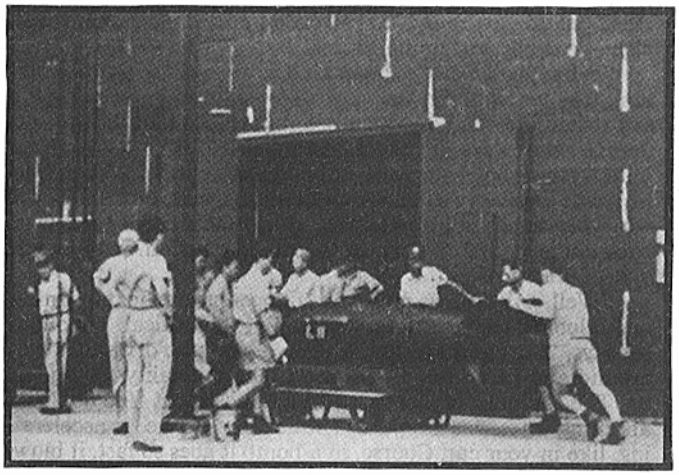


Fig. 10 - "Little Boy" on the island of Tinian

Many persons have expressed concern for individuals who were in the path of the fallout from the Nevada Test Site. A 10 year study, by the National Cancer Institute, was released in 1992 which examined a possible link between fallout and thyroid cancer in residents of southwest Utah. It concluded that there was a statistically valid increase in

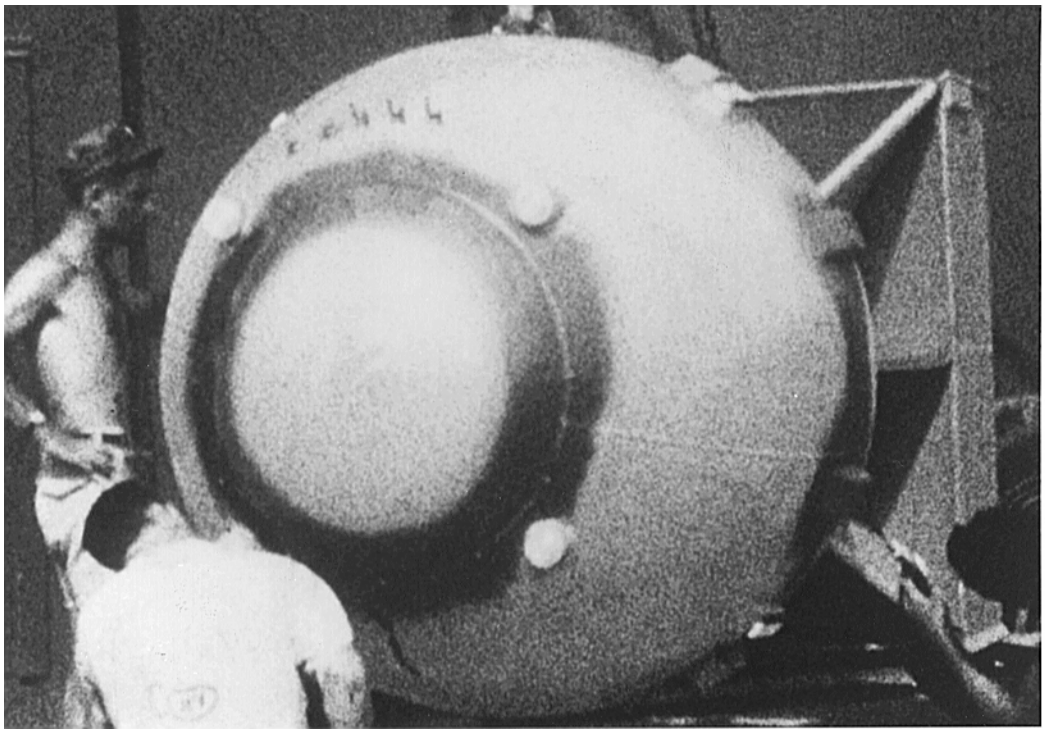


Fig. 11 - "Fat Man" being readied

Los Alamos National Laboratory

thyroid cancer among the exposed children. In the study group of 3,545 children whose dose could be assigned, 19 thyroid cancers were identified. Only 10 subjects had thyroid doses over 100 rads. (The top 5 drank milk from a backyard goat. Goats excrete relatively more radioiodine than cows.) The median dose for the subjects that drank milk was 3.0 rads in contrast to 0.05 rad median dose for milk abstainers. The same research group also looked at leukemia incidence in the fallout victims. In a 1990 report, they found that fallout from the test site "may have been responsible for 3 to 6 percent of Utah leukemia deaths over three decades from 1952 to 1981." In 1992, the U.S. Department of Justice implemented the Radiation Exposure Compensation Act of 1990 which gave $50,000 to $100,000 awards to fallout victims and uranium miners who met specified criteria.

Electronic Product Radiation

A large number of consumer products and industrial devices are a source of radiation due usually to the high potential differences needed to operate the circuits. Typical examples include color television receivers, video display terminals, cold cathode discharge tubes, electron microscopes, airport baggage inspection systems and shoe fitting fluoroscopes. X-rays are released at low exposure rates from television sets as a result of components being bombarded by electrons. Although the numbers of persons so exposed in the United States is a large fraction of the total population, the doses received are small in comparison with natural background. A summary of a 1987 study by the NCRP on genetically significant doses from consumer electronic products is given in Figure 12. Note that most of the numbers in the last column, U.S. Population Average GSD are smaller than the previous column since only a portion of the population is exposed to those sources. Also, in the case of airport baggage inspection, the tabulated doses are to personnel, not the suitcases. The actual average x-ray dose to inspected items is about 0.8 rem.

Airport passenger screening systems have come into wide use (and wide criticism) after the 9/11 attacks. One of the more popular types is the backscatter x-ray system. In addition to privacy concerns, air travelers are frequently concerned about the radiation dose received from the scanning procedure. Current equipment typically produces an x-ray energy of 50 kVp, considered to be a low energy x-ray. The low energy is essential because the equipment is measuring reflected (backscattered) x-rays only. Virtually none have enough energy to penetrate through the body.

Based on measurements, current backscatter x-ray scanners delivers less than 10 μrem per scan. Since a cross-country six hour flight delivers a cosmic radiation dose of 2000 μrem per trip, this air traveler would receive 200 times more dose flying than being scanned. As of 2011, standards are beginning to appear for the airport machines. A draft Food and Drug Administration report calls for a limit of 10 μrem per scan. A comprehensive standard was updated in 2009 by the American National Standards Institute, ANSI. Their report N43.17-2009 recommends a maximum dose of 25 μrem per screening from backscatter x-rays. This allows for repeat scans as needed without exceeding the limit.

Figure 13 shows the system. These units can detect concealed plastic and metal weapons, plastic and liquid explosives, drugs, ceramics, and other contraband

Source	People Exposed (Thousands)	Ave Annual GSD Dose to Exposed (μSv)	Ave Annual Population GSD (μSv)
Domestic water supply[a]	230,000	10-60	10-60
Building materials	120,000	70	36
Mining & agriculture products	200,000	5-50	<10
Coal fuel	230,000	0.3-3	0.3-3
Natural gas heaters	16,000	18	1
Natural gas cooking ranges	125,000	4	2
Dental prostheses	45,000	0.7	0.1
Ophthalmic glass	50,000	4	1
Television receivers	230,000	<<10	<<10
Luminous watches & clocks	15-20,000	0.4-1	0.01-0.05
Airport inspection systems	30,000	0.02	<0.01
Smoke detectors	100,000	0.08	<0.01
Road construction materials	5,000	40	1
Electron tubes	230,000	0.04	0.04
Fluorescent lamp starters - Th	50,000	0.0002	0.00001
Gas mantles - Th	50,000	2	<1
Tungsten Th welding rods	300	160	0.2
Check sources[b]	800	<10	<0.04

Footnotes:

a) Of this number, the predominant exposure is to that portion of the population using groundwater sources.

b) Sources used to check the performance of radiation monitoring instruments.

Fig. 12 - U.S. consumer electronic product radiation doses

Fig. 13 - An airport backscatter x-ray security scanner

concealed anywhere on a person's body. The scan image presents only an outline of the person to protect passenger privacy.

Product Radioactivity

A different collection of consumer and industrial products contributes to the population exposure from artificial sources as a result of containing radioactive materials. The NCRP estimated in 1987 that 28 million Americans wear radioactive wrist watches and that 30 million radioactive alarm clocks were still in use in this country. Luminous dials are also popular on aircraft and ships. The radioisotopes commonly used for illumination devices are Ra-226, H-3 and promethium-147. Since radium-226 is a gamma ray emitter, it produces a much higher genetically significant dose rate than the tritium or promethium, both beta emitters which can be effectively shielded by the case of the luminous device.

Tritium exit signs became popular safety devices in the 1980s since they glow in the dark without any electrical input. The annual number registered with the US Nuclear Regulatory Commission peaked around 1990. With a useful lifetime of 15 to 20 years, many are still being disposed of. A best guess is that only about 1 in 4 signs are making it back to the manufacturer for proper disposal. Many of the rest are showing up in local community landfills. A 2006 study in Pennsylvania noted that 93% of the state's landfills tested positive for tritium. Although the signs contain tritium in the form of gas in glass tubes (Figure 14), when buried in the city dump, the glass tubes are fractured and the tritium is thought to oxidize into tritiated water. The 2007 NCRP Report 160 dismisses the average population dose from all luminous devices as negligible.

Tobacco products contain significant levels of Pb-210 and Po-210, both alpha emitters. In 1977 about 50 million persons in the U.S. smoked an average of 1½ packs of cigarettes per day. By 2007, the number had dropped to about 45 million. The NCRP estimates that each of these individuals receives a lung dose of 160 mSv per year (160 times higher than the maximum permitted whole body dose to members

Fig. 14 - Tritium exit sign

of the public and 50 times the natural background level).

About half the population receives an excess radiation dose from commonly used building materials. Masonry contains 1 to 5 parts per million of uranium and thorium. These same nuclides are also used in highway construction materials. Persons driving on such roads receive a gonadal dose rate about 3 times the natural background rate.

Radionuclides are also released into the environment from combustion of fuels. Coal burned to generate electrical power generates fly ash which carries measurable levels of Ra-226, Th-232 and isotopes of uranium, lead and polonium. The average concentrations for U.S. coal are 1.8 ppm of uranium and 4.7 ppm of thorium. Some lignite coals in the U.S. have 25 ppm of U and 40 ppm of Th. Radiation doses from the average U.S. coal are about 4 mrem/yr (GSD) with dose rates to the bone of 36 mrem/yr for a person living 500 meters downwind from a 1,000 MW plant. If the worst case, lignite coal, is burned, these doses are increased 1,000%. A fuel source to which an even larger population is exposed is natural gas for home cooking use. This gas contains about 10 to 20 pCi/liter of radon. The radon produces a lung dose of about 6 to 9 mrem/yr to about 125 million persons in the U.S. population.

The radiation dose produced by ionization type smoke detectors is sometimes of concern. A long half-life alpha emitter produces an electrical current which is changed by the presence of charged ions in smoke, thus triggering the alarm. Residential units use about 1 microcurie of Am-241. As of 1975, about 10 million persons were being exposed to this radiation field. Present designs on the detectors have reduced dose rates somewhat over earlier versions. Approximately 0.01 mSv/yr was the 1987 NCRP estimate of GSD to the exposed individuals.

Two final examples will illustrate how widespread radioactivity is in modern uses. The porcelain used for making dentures contains K-40 and is usually doped

1. Radium wrist watch - 3 mrem

2. Tritium wrist watch - 0.6 mrem

3. Radium dial alarm clock - 7 to 9 mrem

4. Cigarettes, 1 1/2 pack per day, to lungs - 16,000 mrem

5. Building masonry - 7 mrem

6. Road construction materials - 4 mrem

7. Coal fired power plant, to lungs - 1 to 4 mrem

8. Cooking with natural gas stove - 6 to 9 mrem

9. Home ionization type smoke detector - 1 mrem

10. Dental porcelain in false teeth, to gums - 60,000 mrem

11. Thorium rose tinted eye glasses, to eyes - 4,000 mrem

12. Phonograph record static eliminator - 0.001 mrem

13. Reading a book, 3 hours per day - 0.5 mrem

14. Aircraft luminous instrument dial - 1,000 to 5,000 mrem

15. Radium pocket watch - 0.5 mrem

16. Radioactive lightning rod - 0.05 mrem

17. Uranium glaze in dinnerware, to skin - 2,400 mrem

18. Gas camping lantern mantle - 0.1 to 0.4 mrem

Fig. 15 - U.S. average annual doses from radioactive products

with uranium to give a more natural color to the false teeth. An average dose to the tissues of the mouth of 600 mSv/yr is estimated for U.S. made teeth. Dose rates 10 times higher have been reported for dental porcelain from Great Britain. Some rose-tinted eyeglasses contain up to 0.25% thorium by weight. Average annual dose rates have been measured to be between 10 and 40 mSv to the eye. A summary of a variety of reported values for radiation exposures to consumer products which contain radioactive materials is given in Figure 15.

X-ray Tubes

Before discussing particular types of x-ray machines, the theory of operation of x-ray tubes will be covered. Electronically speaking, an x-ray tube is a vacuum diode; it consists of two elements inside an evacuated glass tube. Figure 16 is a cross-sectional sketch of an early tube design.

The introduction of the heated filament was the idea of W.D. Coolidge in 1913.

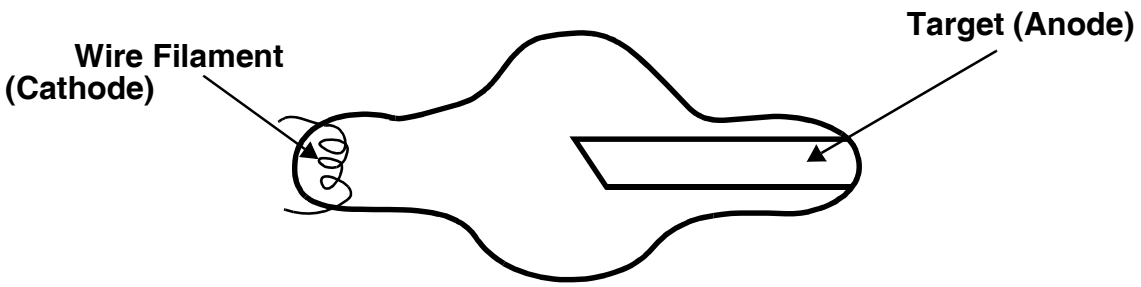


Fig. 16 - Basic components of an x-ray tube

The cathode consisted of a tungsten wire which was heated by passing an electric current through it to cause the release of electrons through thermionic emission. An alternating potential difference from a high voltage step-up transformer was applied between the cathode and target. During that half of each cycle when the target was positive with respect to the cathode, electrons under the influence of the Coulomb force would accelerate across the gap and strike the target. This would lead to the emission of bremsstrahlung radiation which constitutes most of the energy in the spectrum of x-rays emitted by the tube. It turns out that the intensity of the x-radiation is directly proportional to the square of the potential difference across the tube. In almost every modern type of x-ray tube, the applied voltage is a sine wave. Thus, the x-ray intensity will be proportional to a $\sin^2$ function. Since the square of a sine wave is sharply peaked around the maximum of the sine wave, the x-ray output will occur in "bursts," one for each cycle of the input voltage. This is illustrated by Fig. 17.

Figure 18 shows the typical x-ray spectrum of energies emitted by a tube. The spectrum has photons from 0 energy all the way up to the energy corresponding to the maximum energy of an electron striking the target, i.e., the energy of one electronic charge accelerating through the peak tube potential difference (called the peak

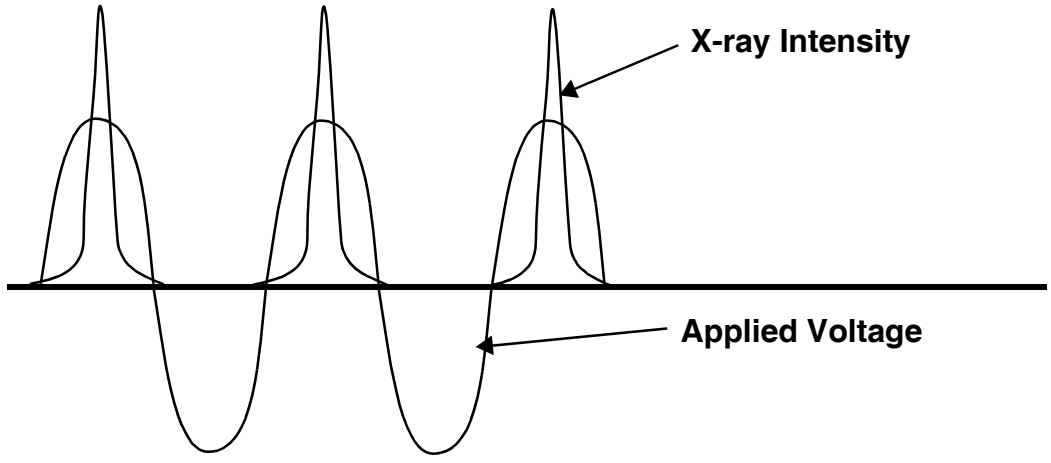


Fig. 17 - X-ray tube input and output waveforms

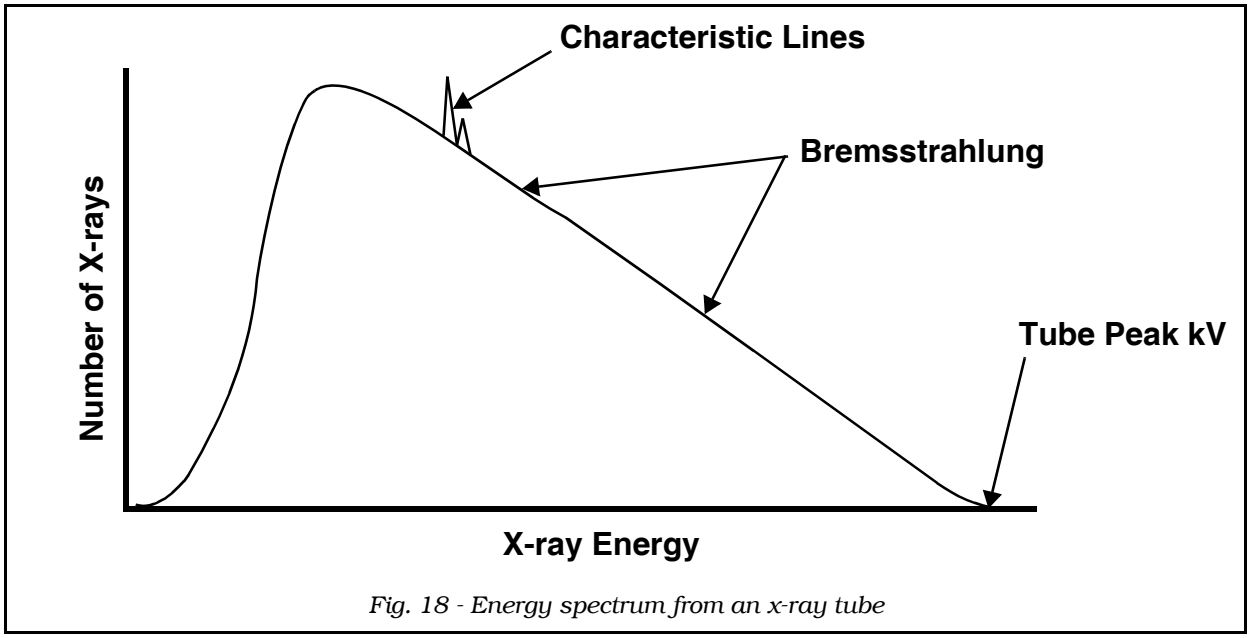


Fig. 18 - Energy spectrum from an x-ray tube

kilovoltage or kVp). The smooth part of the curve is due to the bremsstrahlung which is radiated as electrons are deflected from their paths by target atoms. The superimposed "spikes" are "characteristic x-ray lines" and their location depends on the atomic number, Z, of the target. Physically, they are caused by photons which are released by the target atom when a higher shell electron drops down to fill a vacancy created when one of the K shell electrons is ejected by an incoming electron from the tube cathode.

If a pair of external rectifiers are added to the circuit, "full wave rectification" results in the replacement of the negative half-cycle by another positive half-sine wave. This will double the output of the X-ray tube, producing 120 bursts per second for 60 Hz line frequency. Many modern x-ray machines carry this one step further by use of "three phase" power lines. The voltage waveform for such lines consists of three sine waves, each one a third of a cycle behind the next. This makes it possible to get six x-ray bursts per cycle.

The major engineering problem associated with almost every x-ray tube is assuring the adequate removal of the heat generated when electrons enter the target. The reason for this is clearer upon the study of a typical range energy curve for electrons in a metal target. A curve for tungsten is shown in Figure 19.

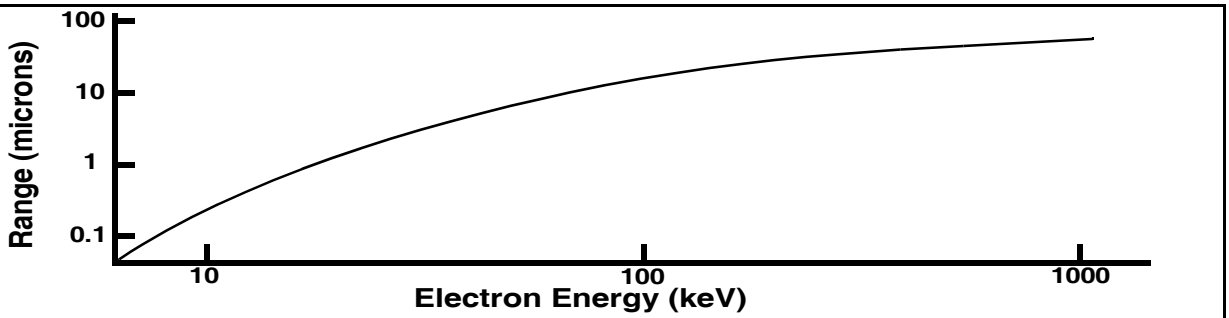


Fig. 19 - A range-energy curve for electrons in tungsten

Sample Problem 2

GIVEN:

An x-ray tube rated at 100 kVp has an instantaneous current rating of 300 mA.

FIND:

The thermal power (heat energy) generated at maximum ratings.

SOLUTION:

The electrical power generated in the target is found from the product of the current (in amps) and the potential difference (in volts). Thus, P(watts) = 100 kV x 1000 V/kV x 300 mA x 1 A/1000 mA = 30,000 watts! About 98% of this energy is converted to heat so the heat generated = 98% x 30 kW = 29 kW.

At common x-ray machine energies, electrons will penetrate less than 0.1 mm (100 microns) into the target. This is equivalent to the thickness of one sheet of paper. Based on actual measurements, about 98% of the incoming electron energy appears in the form of heat in the target. Thus, a very thin surface slice of metal absorbs this heat and gets very hot. See Sample Problem 2 for an example. Even considering the pulsed nature of the beam, the target must be designed to stand up to extremely high temperatures and thus requires an efficient heat dissipation mechanism. Practical methods for heat removal will be discussed shortly.

The second design criterion, after heat dissipation, is maximizing the bremsstrahlung radiation. As mentioned in Chapter 3, the amount of bremsstrahlung produced in an absorber is directly proportional to the atomic number, Z, of the absorber. Therefore, high Z metals which have high melting points are the best choice for x-ray tube targets. Tungsten, with a Z of 74 and a melting point of 3370 degrees Celsius (6100 degrees F) is frequently used. In spite of careful tube design and operator training, tubes still occasionally overheat to the point of local melting of the target surface. This generally causes a decrease in fine detail in the x-ray image.

Returning to practical tube design, several different heat removal systems are employed, depending primarily on the final use of the x-ray machine. Figure 20 is a photograph of a dental x-ray tube. It illustrates

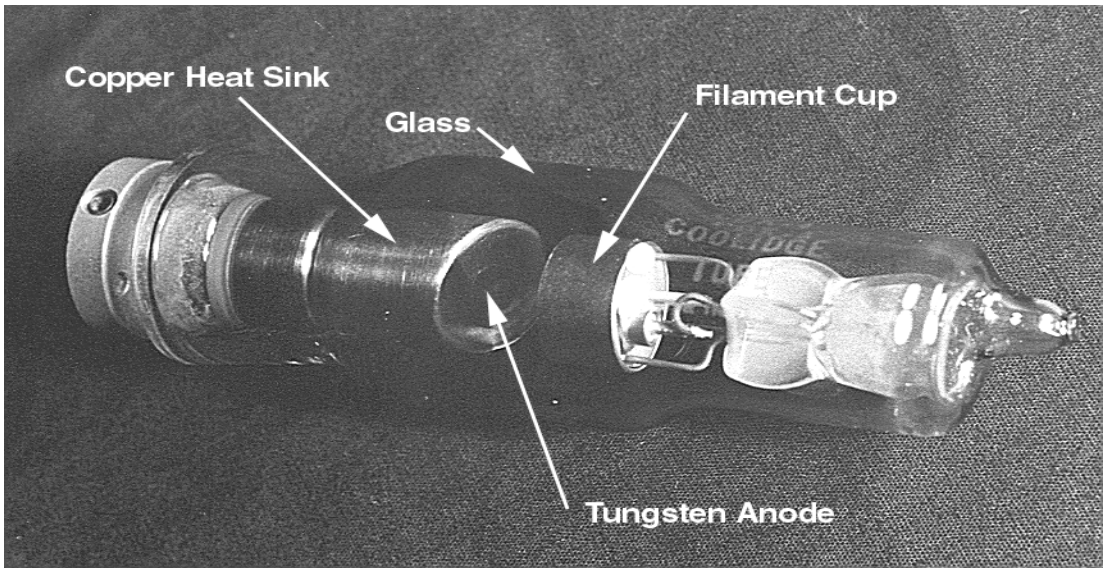


Fig. 20 - A dental x-ray tube

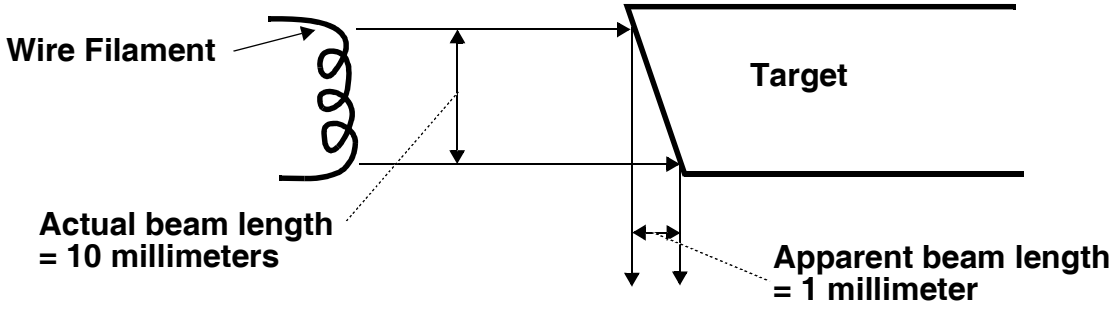


Fig. 21 - The geometry of the "Line Focus Principle"

two heat removal techniques. The thin tungsten chip of metal that receives the full electron beam is embedded in a copper heat sink. The copper rod conducts the heat out the left end of the tube where a radiator assembly (not shown in the photograph) transfers the heat to a volume of cooling oil inside the tube housing.

The second technique in this tube for heat dissipation involves a fact of geometry called the "line focus principle," which is illustrated by Figure 21. By using only those x-rays which emerge at 90 degrees to the tube axis and by INCLINING THE FRONT FACE of the target at 17 degrees, the APPARENT length of the x-ray source is 10 times smaller than the actual length of the electron beam on target. (This same geometrical effect can be even better illustrated by drawing a line on the palm of your hand and then watching it "shrink" in length by viewing it at a glancing angle.) The practical result of the line focus principle is reducing the heat load PER UNIT SURFACE AREA OF THE TARGET by a factor of 10 times.

The analytical x-ray tube shown in the photo in Figure 22 illustrates a different approach to heat removal. The shower head, visible when the top of the tube is unbolted, directs a stream of water droplets onto the back of the hollowed-out tube target. The tube filament is barely visible through the cutaway section of the tube wall. When a water droplet hits the hot metal surface, it vaporizes into steam. This change of state from liquid to vapor removes an extra amount of heat, as a bonus, due to the needed "heat of vaporization"(539 kcal/kg for water).

Figure 23 is a photo of an industrial radiography x-ray tube with the glass envelope removed and the tube elements mounted in a metal support bracket. This tube uses forced oil cooling. A pump supplies oil under pressure from a tank which acts as a heat sink. The oil is directed against the backside of the hollowed out anode and then recirculates back to the storage tank. The actual target is a thin slice of tungsten metal embedded in a copper anode assembly to conduct heat to the oil. Remember that the high Z tungsten will produce a much higher x-ray intensity than would a low Z copper target. This tube also uses the line focus principle by slanting the target face.

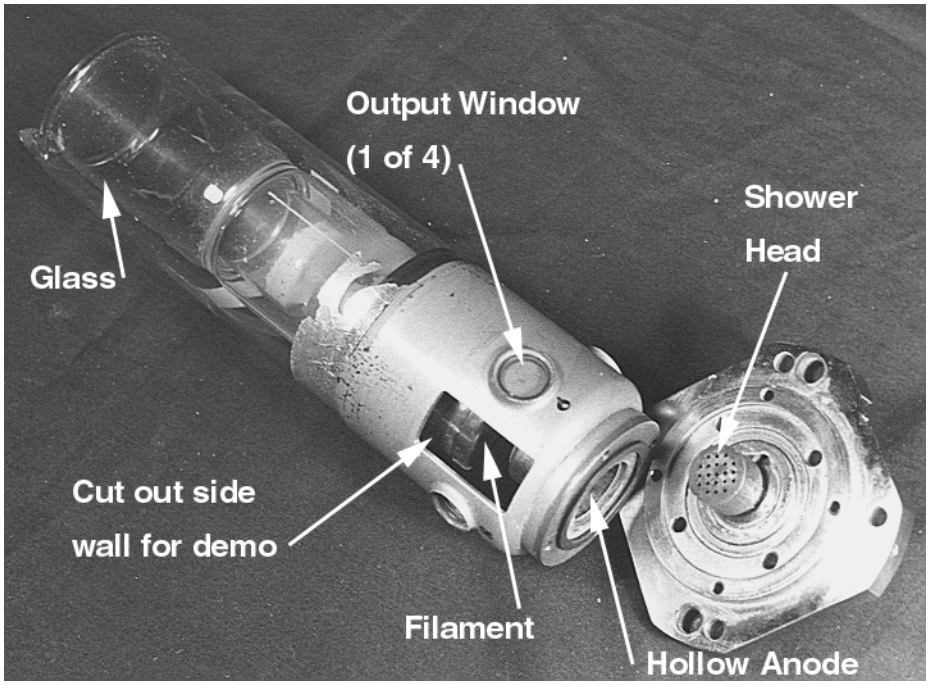


Fig. 22 - An analytical x-ray tube

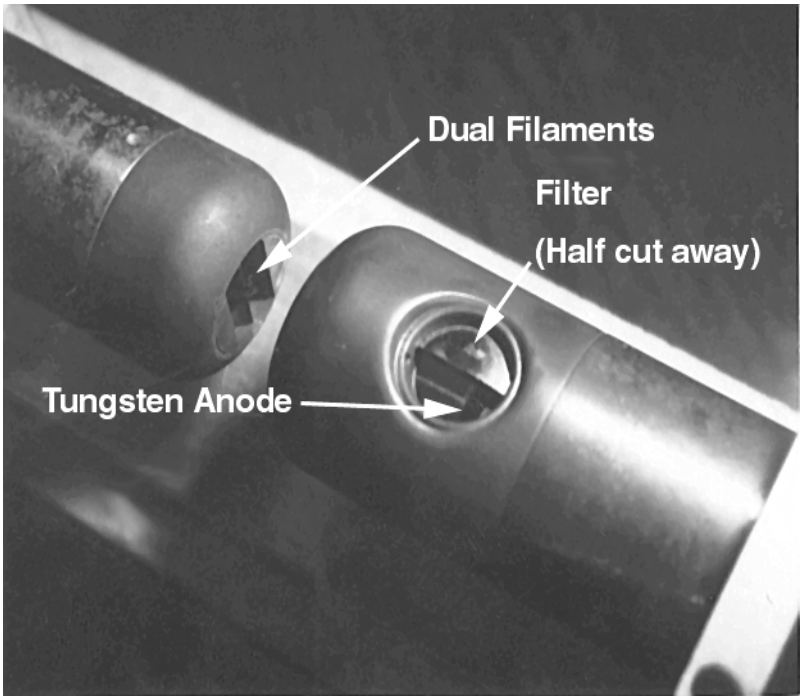


Fig. 23 - An industrial radiography x-ray tube

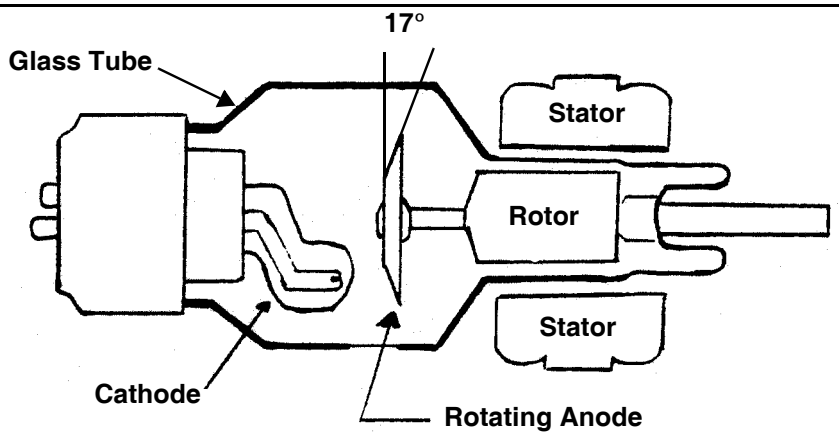


Fig. 24 - The rotating anode medical radiography x-ray tube

The drawing in Figure 24 shows a medical radiographic x-ray tube used almost universally for routine diagnostic medical x-ray procedures, the rotating anode tube. In the example shown in the photograph in Figure 25, most of the glass envelope has been removed and the tube components mounted in a support stand.

This tube design uses two basic heat dissipation techniques. A wire filament, not visible in the photograph, directs the electron beam onto the outer edge of the circular tungsten anode disk. This outer region of the disk is beveled at 17 degrees to use the line focus principle. In addition,

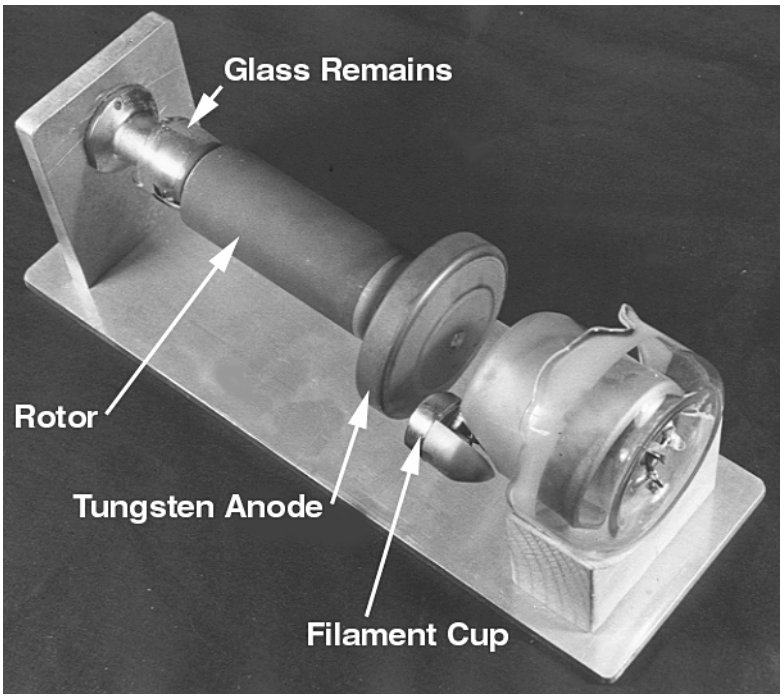


Fig. 25 - Rotating anode tube with glass removed

the entire target assembly is the rotor of a high speed induction motor. It rotates, inside the glass tube envelope, on ball bearings with an angular frequency of typically 4,000 to 8,000 rev/min. The driving coils of the induction motor are located outside the glass tube. During the fraction of a second in which the electron beam is directed on the target, the target disk turns to continuously bring new, cold tungsten metal under the beam of electrons. This, in effect, spreads the area over which the heat energy is actually deposited. The sector of metal struck by the beam is hundreds of times larger than the comparable area in a fixed anode tube. This is important in diagnostic medical applications because exposure times of the patient must be kept very short to reduce smearing of the image caused by patient motion.

The last tube to be discussed is the "flash x-ray tube," an example of which is shown in Figure 26. This device is designed to deliver single pulses of x-rays a few nanoseconds long (1 nanosec = 10^{-9} sec). To get a reasonable intensity, a huge instantaneous tube current is required. Sufficient current cannot be generated by thermionic emission of electrons from a heated filament. In these tubes, the filament consists of a "pin cushion" of tiny needle points which release a huge pulse of electrons through field emission when a very high voltage is applied across the tube. The electron pulse exits the tube through a thin metal window as indicated in the photo. The window in the photograph has been punctured by repeated electron bombardment. Directing the electrons on an external target assembly produces the short burst of x-rays at extreme intensity.

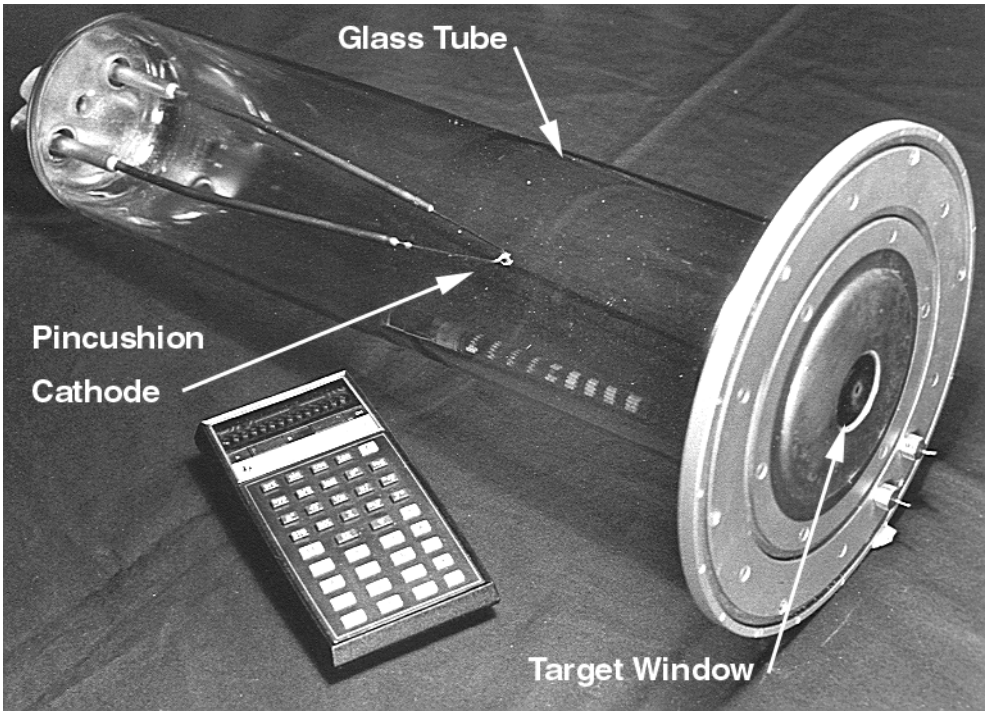


Fig. 26 - A "flash" x-ray tube

Tube Type	kV	mA	Duration	Spot Size
Med. Diagnostic	30-130	to 600	2 sec	1-5 mm^2
Dental	60-90	50	1/5 sec	4 mm^2
Therapy	10-6000	to 30	minutes	30-50 mm^2
Analytical	10-100	to 25	hours	10 mm^2
"Flash"	2000	4 x 10^8	20 nsec	5 mm^2

Fig. 27 - Summary of typical x-ray tube parameters

A summary of some of the operating characteristics of typical x-ray tubes is shown in Figure 27. Typical x-ray output intensities are of the order of 1 R/sec at one meter for medical tubes. Note that, while this RATE is very high, the unit is typically operated for only a few hundredths of a second at a time. The parameters listed in the table for the flash x-ray tube result in an instantaneous dose rate of 4 X 10^{12} R/hr.

X-ray Machine Applications

There are a number of different categories of use for x-ray machines. An x-ray machine usually consists of the appropriate x-ray tube, an electrical source for high voltage, a source of tube filament current and radiation shielding to collimate the beam to some limited size and shape. Some of the more commonly used medical and industrial units will now be covered.

Medical x-ray machines are divided into two basic groups - diagnostic and therapeutic. A diagnostic medical x-ray procedure is used to obtain an image of some body part on photographic film. The 2009 NCRP Report 160 identified medical diagnostic x-rays as the second largest contributor to the radiation exposure of the US population. (Radon was first, but only by 1%.) Now, we will examine several popular medical x-ray procedures and equipment in some detail. Following that will be an examination of the doses delivered to both the patient and to the general population.

Three of the more common conventional diagnostic applications are illustrated in Figure 28. Radiography involves placing the patient between the x-ray tube and a

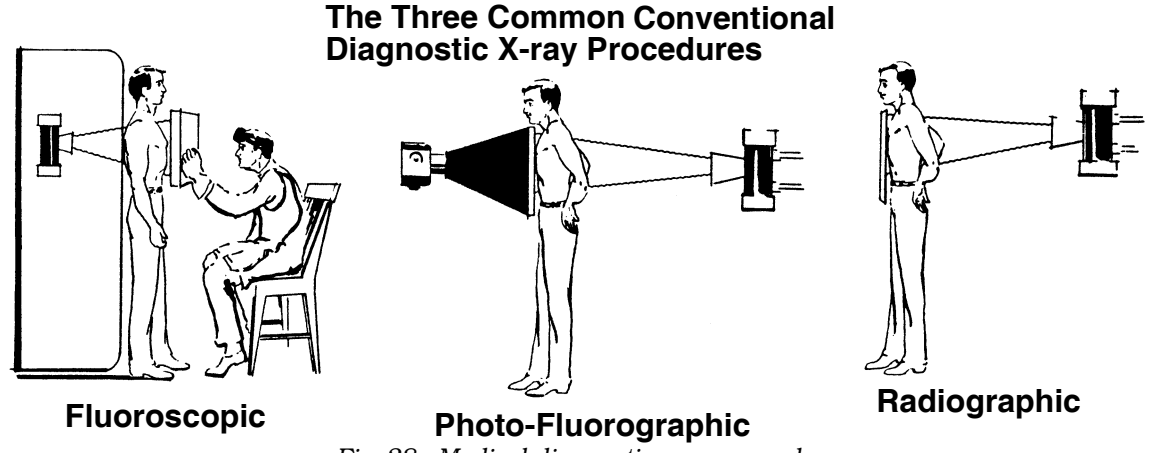


Fig. 28 - Medical diagnostic x-ray procedures

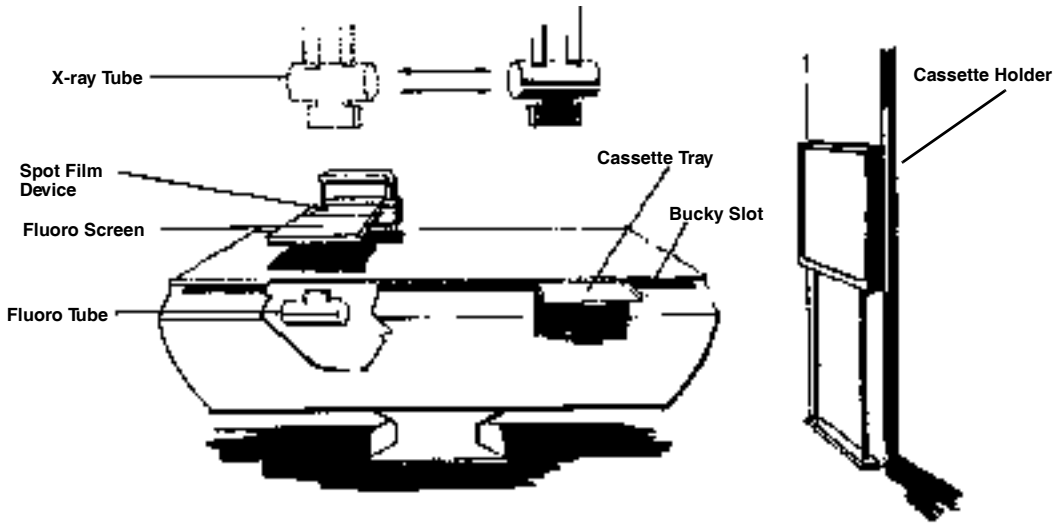
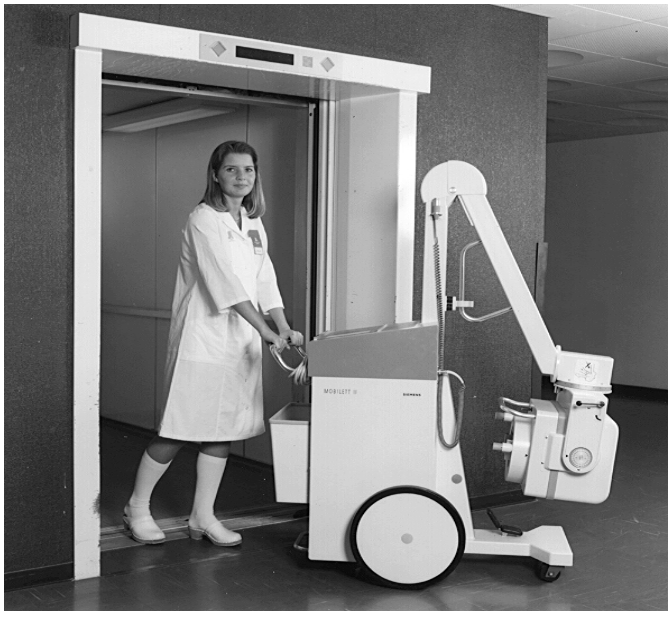


Fig. 29 - A medical general radiography machine

film cassette. The x-rays penetrate the patient and produce a shadow picture on the film. The routine "chest x-ray" is an example. The drawing in Figure 29 illustrates the components that are commonly found in a general radiographic x-ray machine. In some cases, a patient needing an x-ray procedure is too ill to come to the x-ray department in the hospital. Mobile radiographic units have been designed to solve this problem. The x-ray tube, power supplies and shielding are mounted on a portable cart which is brought to the patient bedside. A film cassette is slipped under the patient and the necessary exposures made. Figure 30 shows an example of this type of portable equipment.



Courtesy Siemens Medical Systems

Fig. 30 - A mobile medical x-ray machine

Fluoroscopy equipment substitutes an electronic imaging device (image intensifier) in place of the film cassette. The information from the image intensifier is sent directly to a video monitor on which the physician (usually a radiologist) can watch a live, play-by-play picture of the internal body action. An example of this use would be a “GI series” in which a patient drinks a liquid x-ray absorbing substance (barium) while the physician watches, on the monitor, its progress down the outlined tract. Ulcers in the lining of the GI tract show up when the barium contrast agent penetrates the site. Figure 31 shows a fluoroscopy procedure in progress. Often a video recorder is attached to the side of the intensifier. The radiation output of a fluoro unit is limited by law - 5 R/min (machines without an automatic exposure rate control) 10 R/min with a control and 20 R/min (machines with an optional high level control).

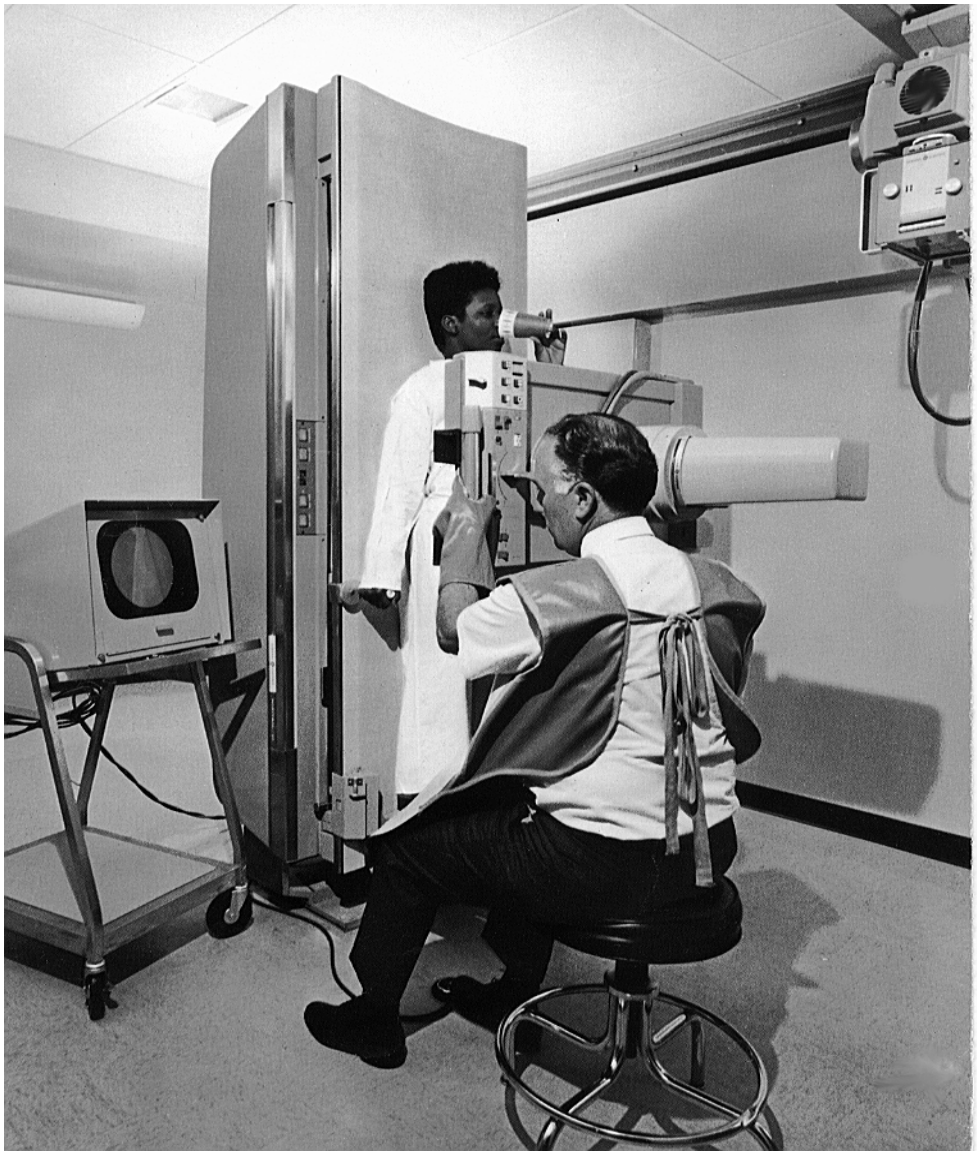


Fig. 31 - A fluoroscopy procedure in progress

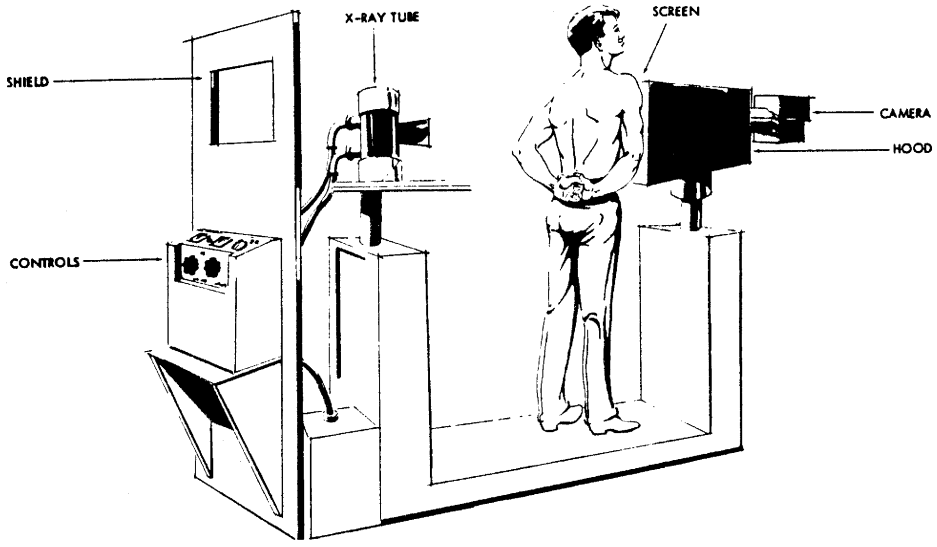


Fig. 32 - A photofluorographic x-ray system

Photofluorography is similar to the radiography described above except the film cassette is replaced by a fluorescent screen which forms an image of the x-rays which penetrate the patient. This screen image is, in turn, photographed by a 70 mm camera. A large film magazine in the camera will store many images in a small roll of film. After development, the film is viewed by a radiologist in an optical magnifier. This equipment is often used in screening programs with large numbers of persons, for example, in prison admission procedures. Figure 32 is a drawing of the basic components of such a system.

Mammography, a diagnostic x-ray examination of the breast, calls for specialized equipment. Medical experts feel that this is the best available method of breast cancer detection in the earliest stages before a lump can be felt or the cancer has metastasized (spread to other tissues). Although somewhat controversial a few years ago, criteria have now been established by such organizations as the National Cancer Institute, American Cancer Society and American Congress of Obstetricians and Gynecologists. They all agree that mammography should be performed on all women with breast cancer symptoms and that routine mammograms should be taken on all women over the age of 40. The NCRP has established technical criteria which include use of vigorous breast compression (see Figure 33), x-ray tubes with molybdenum target anodes operated at 22 - 26 kVp for screen-film mammography and tubes with tungsten targets operated at 40 - 45 kVp for xeromammography. The NCRP recommendations also limit average glandular dose to <0.1 rad for screen-film mammography and <0.4 rad for xeromammography. An example of a mammogram is given in Figure 34.

A final, relatively recent addition to the arsenal of medical diagnostic x-ray equipment is computerized axial tomography (CAT or CT Scanner). A tiny, highly focused x-ray beam is scanned over a portion of the patient. The fraction of the beam intensity which is transmitted through the body part is measured by a detector placed

Fig. 33 - Mammography x-ray machine

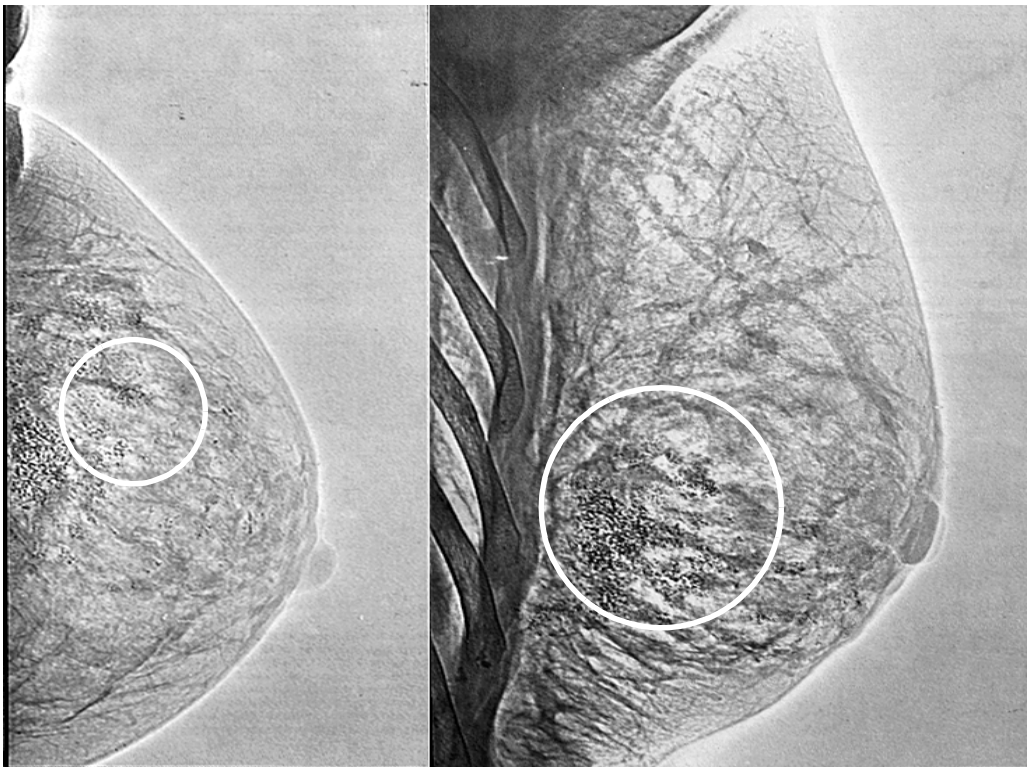


Fig. 34 - Examples of mammograms showing breast cancer

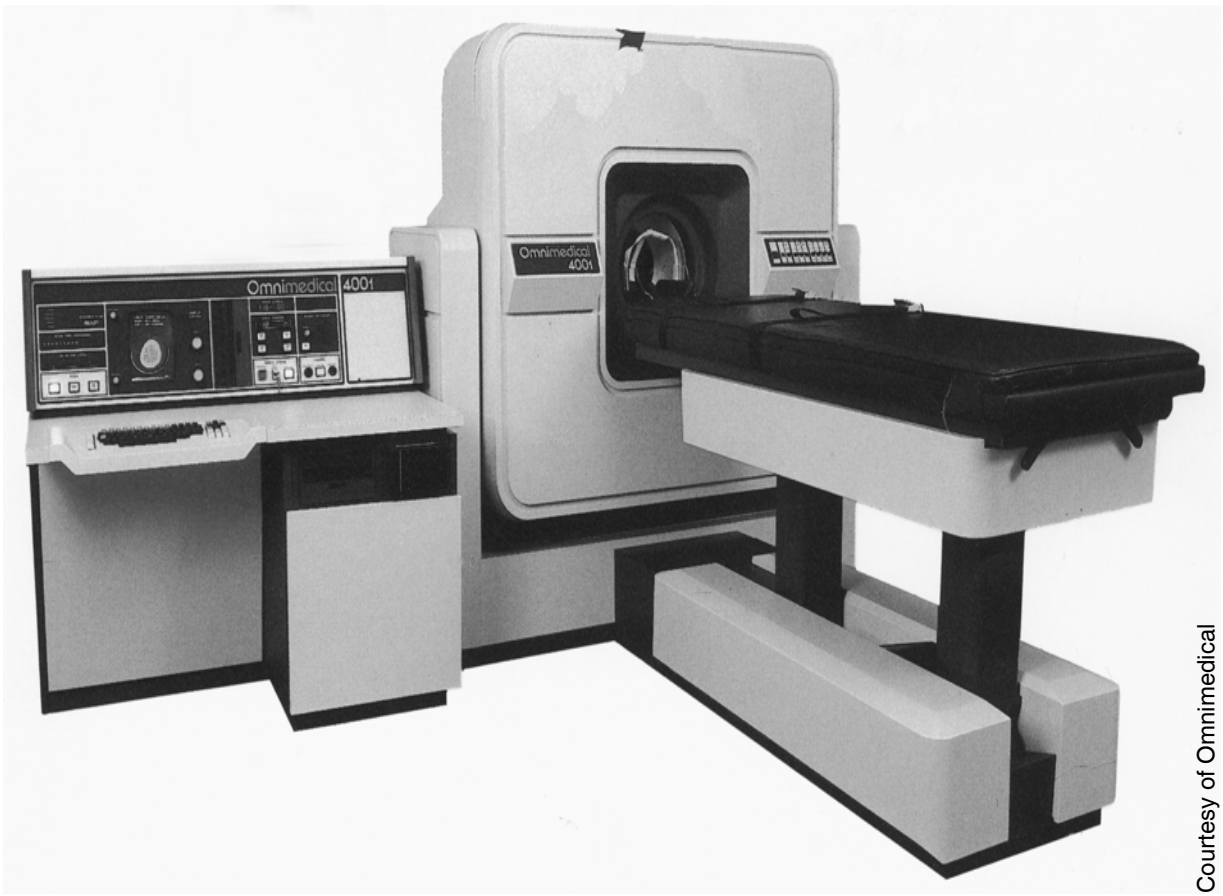


Fig. 35 - A CAT head scanner

on the far side of the body part. By moving a small distance and making repeated measurements of the transmitted fraction, a large number of data points are generated and stored in a computer. Then, the computer analyzes the pattern of data points and reconstructs a cross-sectional view of the body part which must have been in the beam path to produce the pattern of data points that was recorded. A three-dimensional picture is thus reconstructed which gives a lot of anatomic detail, particularly in soft tissues. A common example of the use of this equipment is in tumor localization. An example of a modern instrument used for head scans is shown in Figure 35. A similar instrument is available for "whole body scans." The opening is larger to allow the gantry and patient to move through the source/detector "doughnut." An example of an actual series of image "slices" at different depths of a tumor mass located adjacent to the mid-plane of the brain of a patient is shown in Figure 36.

We will now turn our attention to the dosimetry aspect of diagnostic medical x-rays. Dose information is usually reported in one of two different ways. The radiation dose delivered to the patient undergoing diagnosis is frequently of concern. In addition, the public health risk of the dose can be measured by calculating the average dose to a member of the general population.

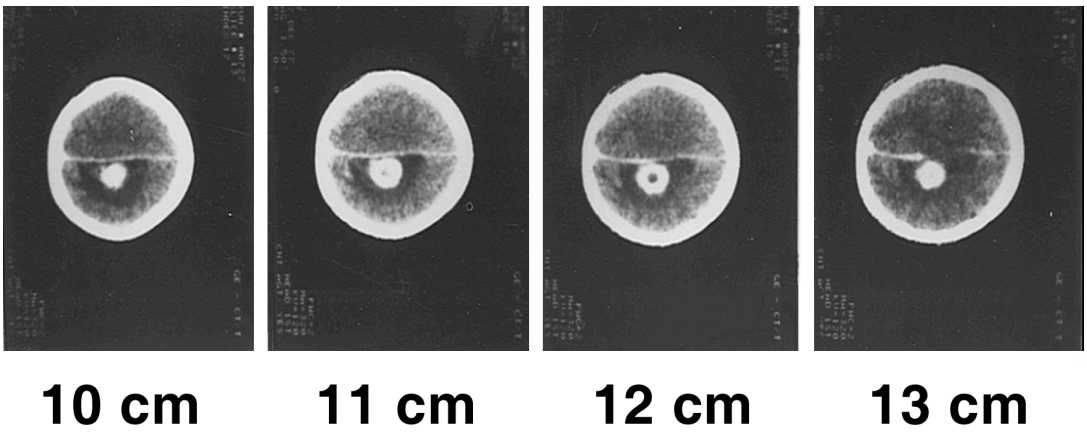


Fig. 36 - CT scan "slices" of a human head at the depths indicated

The 2009, NCRP Report 160 found that half of all medical radiation exposure to the US population was from CT scanning. In 1987, a nationwide survey was conducted of CT Scanners to assess the average dose from a typical head scan. This procedure accounted for over 60% of the CT studies in the U.S. at that time. The typical peak kilovoltage ranged from 120 to 140 kVp. The slice thickness almost always used was 10 mm. The average dose to the head from a 3rd generation scanner was measured to be 3.8±1.6 rad while the 4th generation units delivered 5.8±2.2 rad.

Since the 1980s, the number of CT scans had increased over 700% to 62 million procedures in 2006. Major advances in the equipment have also been made with the introduction of helical scanners and multidetector scanners. These developments speed up the data acquisition and improve the image resolution of the scan. The table in Figure 37, based on a 2006 Health Physics Society Fact Sheet, shows the effective dose for some common CT procedures. Recall from Chapter 5 that the effective dose is the whole body equivalent dose that has the same effect as the partial body exposure actually received by some tissue. By convention, current practice is to use effective dose when reporting patient and population doses from x-ray procedures. This gets us around the considerable problem of determining the biological radiosensitivity of

Examination	Eff. Dose (mSv)	Examination	Eff. Dose (mSv)
Head	2.0	Chest	8.0
Abdomen	10	Pelvis	10

Fig. 37 - Patient doses from some CT scans as of 2006

the tissue being x-rayed. In other words, it "levels the playing field" by referencing the tissue dose to the equivalent effect of a uniform whole body x-ray exposure. The average effective dose per person for CT scans in the US population increased almost 600% from the 1980s, to a value of 1.47 mSv per person in 2006. For completeness, the average population value for fluoroscopy and conventional radiography was 0.76 mSv.

Effective dose results for x-ray procedures other than CT scanning are presented in Figure 38. The effective dose equivalent to the organ of the patient for a number of common x-ray diagnostic procedures is listed. These values were current as of 2006.

Medical diagnostic x-ray exposures account for well over half of the U.S. average genetically significant dose to the population from all sources. Way back in 1960, the U.S. Public Health Service conducted an assessment of the average GSD doses delivered to the population from medical x-ray procedures at 10 year intervals. Surveys were made in 1961, 1970 and 1980. Due to the large amount of data to be analyzed and the complexity of the computer analyses, it took 3 to 4 years to obtain the results for U.S. average GSD. As a result of budget cutbacks, funds have not been available to obtain average GSD values or doses for some procedures for the 1980 and later surveys.

The overall U.S. average GSD was computed for 1964, 1970 and 1982 using measured doses along with estimates of the number of examinations of the various types per year and statistical information on childbearing probability. The final results for these calculations are shown in Figure 39. These average doses resulted from an estimated 130 million persons in the U.S. being examined in 1970 with 661 million x-ray films taken. In 1980, 180 million medical x-ray examinations were conducted.

The second medical use category is therapeutic x-ray procedures. X-radiation has been found to be useful in the management of malignancies. Certain forms of skin cancer respond well to very low energy x-rays, of about 10 to 40 kVp. This low

Procedure	Eff. Dose Eq. (mSv)	Equipment Type
Skull, AP	0.03	Radiographic
Thoracic Spine	0.4	Radiographic
Chest, AP	0.02	Radiographic
Chest, LAT	0.04	Radiographic
Pelvis	0.7	Radiographic
Hip	0.8	Radiographic
Extremities	0.005	Radiographic
Barium Enema	7	Fluoroscopy
Barium Swallow	1.5	Fluoroscopy
Teeth	0.02	Dental

Fig. 38 - U.S. effective dose equivalents from medical x-ray procedures, in millisieverts, 2006 data

1964 Survey Results:

U.S. GSD Average = 17 ± 6 millirem

1970 Survey Results:

U.S. GSD Average = 20 ± 4 millirem

1982 Estimates:

U.S. GSD Average = 7 ± 3 millirem (males)

U.S. GSD Average = 19 ± 1 millirem (females)

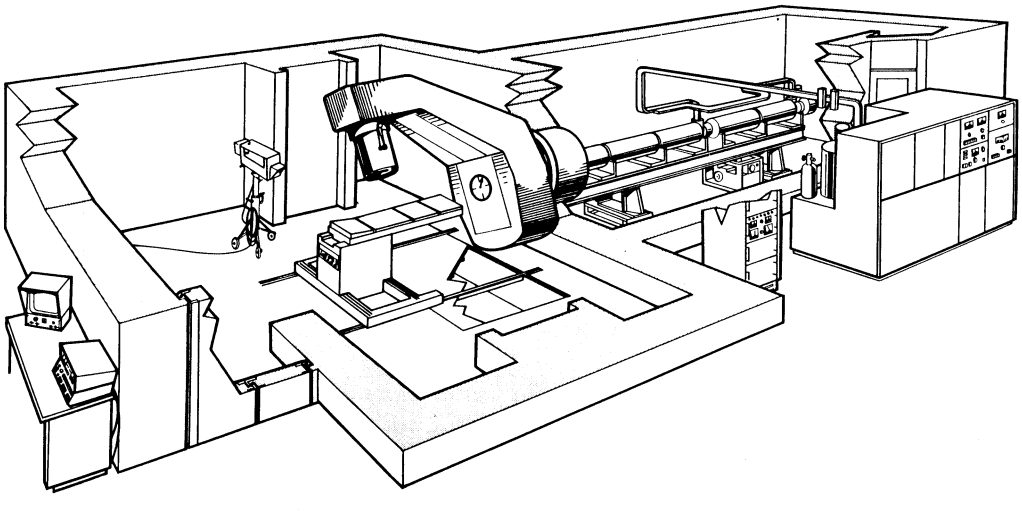
Fig. 39 - The average U.S. GSD from medical x-ray examinations

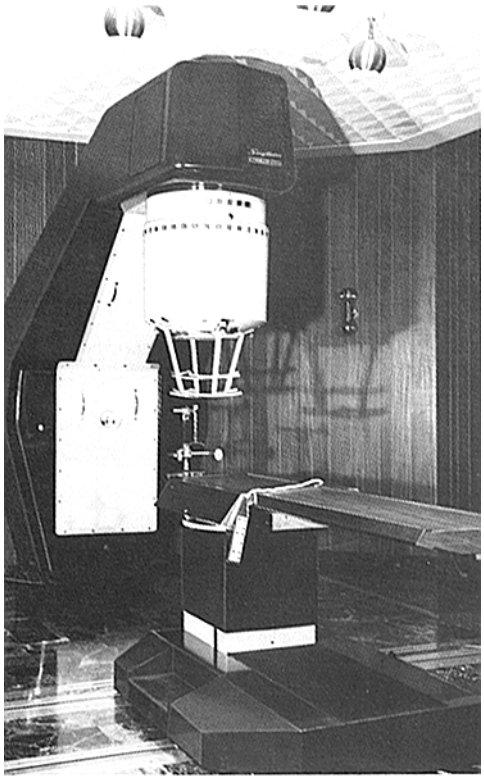
energy causes the dose to be deposited almost completely in the skin and so deeper lying healthy tissues are unaffected. Equipment used for this purpose is designated "superficial x-ray" equipment.

Historically, before the common availability of Cobalt-60 gamma ray sources or medical linear accelerators, higher energy x-rays were used to treat deeper lying tumors. Machine potential differences of 250 kV and 400 kV were common. Such units were given the name "orthovoltage." A few medical centers still use orthovoltage equipment. However, when these low energy machines were used for deep tumor treatment, a serious problem arose. Recall from Chapter 3 that low energy photons interact by the photoelectric effect where absorption is strongly Z dependent. For photon energies above 1 MeV, the absorption is independent of Z because Compton Scattering dominates. Thus, when orthovoltage beams included a patient's bone in the treatment area, the bone received an unacceptably large radiation dose as bone has an effective Z of 13 compared to 7.5 for a soft tissue tumor. Only by moving to higher photon energy will the bone and tissue doses be similar in magnitude.

The common photon generating equipment used by radiation oncology departments for deep tumors today is the medical linear accelerator. The principles of operation of such devices are discussed later in this chapter. Basically, for medical applications, the machines produce high energy electron beams in a microwave waveguide. The electrons are then directed onto a tungsten target and the resulting bremsstrahlung radiation used for treatment. A typical dose rate at 100 cm treatment distance is 300 or 400 rad/min.

Common maximum x-ray energies produced by different commercially available accelerators in modern medical centers are 4 MeV, 6 MeV, 15 MeV, 18 MeV, 25 MeV and 40 MeV. The accompanying drawing and photographs (Figure 40) show a 40 MeV treatment facility. Although radiation therapy treatments deliver large doses (typically 6,000 rad over a few weeks to the tumor site), only relatively few members of the population of childbearing age receive such procedures. The annual U.S. GSD from radiation therapy works out to be approximately 2 mrem per person averaged over the entire population (probably a useless number because, either you get the dose or you don't!).





Courtesy of CGR MeV and CGR Medical Corp.

Fig. 40 - A 40 MeV medical accelerator facility

The final machine category to be covered is industrial x-ray. Approximately 10% of the x-ray machines registered in this country are industrial machines. The primary use is in nondestructive examination (NDE) in which a variety of products are radiographed to show up flaws, metal fatigue, etc. Analytical x-ray equipment forms a smaller category within industrial applications. These machines are used to study the structure of materials by looking at the pattern of x-rays that are scattered off the atomic planes of a crystal or metallurgical sample.

X-ray machines used for NDE applications are usually much smaller than their medical cousins. Units used in aircraft structural inspections operate between 150 and 400 kVp. They usually employ a "panoramic" collimator which emits x-rays over a 360 degree circular arc. Pieces of photographic film encased in light-tight paper packages are placed around the outside of the desired structural component. The x-ray machine is then inserted inside the component to make the exposure. The films are subsequently removed, developed and examined. For examining pipeline welds in the field, a small x-ray machine on crawler treads is remotely advanced inside the pipe until it reaches a weld. A panoramic exposure is made after wrapping the outside of the weld with film. The machine then crawls to the next weld and the process is repeated. The films are usually processed on the spot in a portable darkroom in a van. Figure 41 shows an x-ray machine in use for NDE of tire casings.

There are several thousand analytical x-ray machines in current use in the USA. They typically operate between 50 and 75 kVp with a few mA of current. They are designed for continuous operation, in sharp contrast to the fractional second exposures typical of most medical procedures. A collimated beam of x-rays is

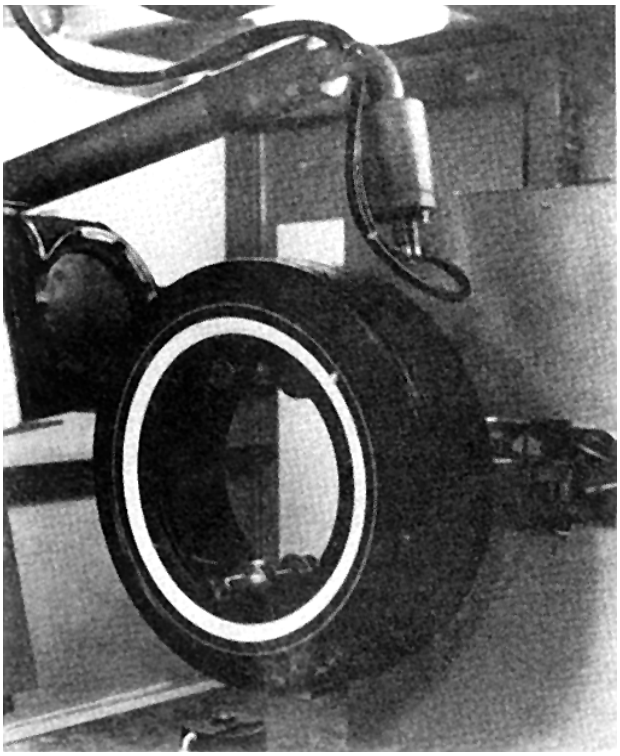
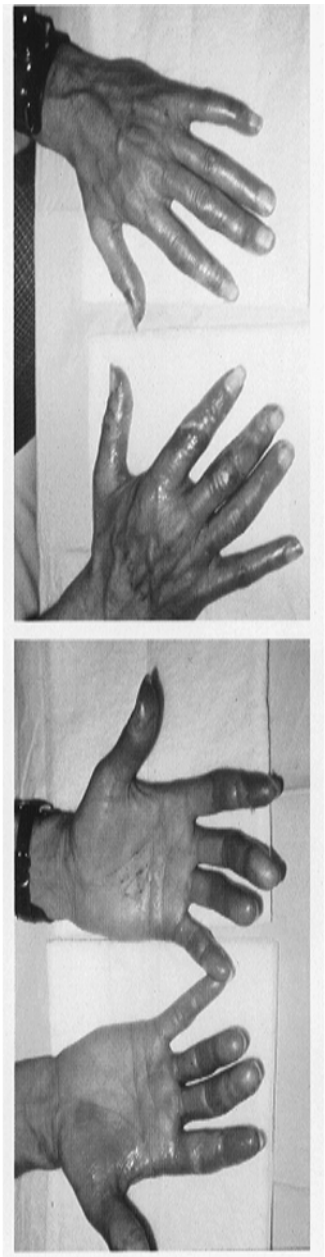


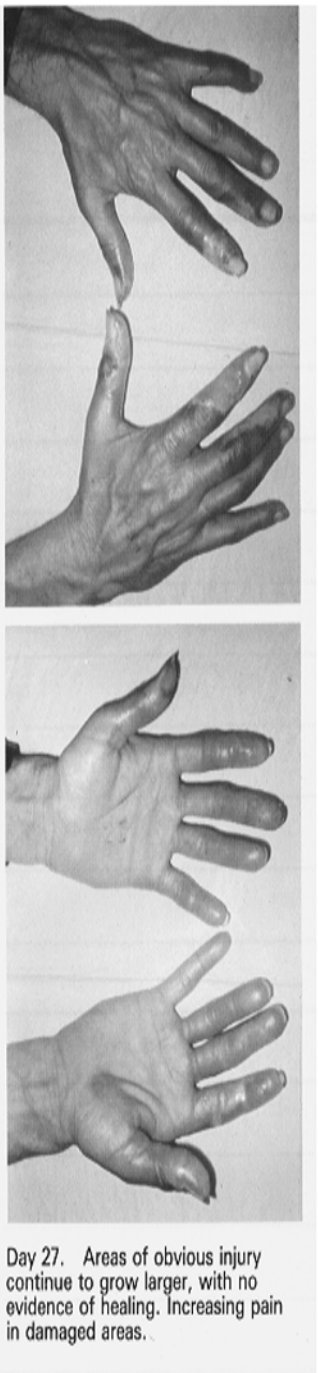
Fig. 41 - Non-destructive testing of tires by x-ray

"bounced off" the sample and the pattern of scattered rays is measured by a detector. Often the x-ray tube contains four beryllium windows (Z = 4) to allow the very low energy x-rays to emerge without substantial attenuation. This means that very high dose rates to the skin can occur. The tube that was shown in Fig. 22 earlier in this chapter had a measured output exposure rate at 5 cm (about 2 inches) of 58,800 R/minute - virtually A THOUSAND ROENTGENS PER SECOND. Figure 42 is a photo of a typical machine with the steel x-ray shield in place. In older machines, it is possible to remove the shield and operate the x-ray tube and adjust the sample by hand to maximize signal strength in the detector. Frequently persons have accidentally exposed fingers or portions of their hands to the direct beam. A significant x-ray burn can occur in a few seconds. Skin layers are often destroyed right down to the underlying bone. These injuries are unusually painful and slow healing due to the destruction of the skin's underlying regenerative tissues. Figure 43, taken from NUREG/BR-0001, shows the progressive deterioration in the hands of an industrial radiographer exposed to between 22,000 and 30,000 rems.

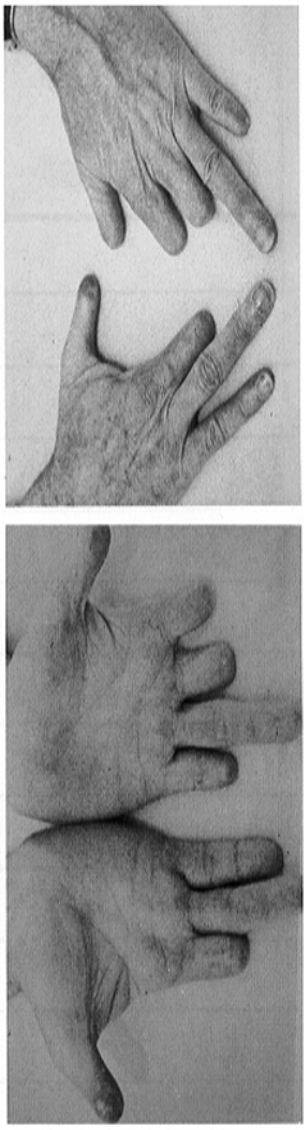
Fig. 42 - A typical industrial analytical x-ray machine



Day 24. Blisters are breaking and dead skin is sloughing off, exposing raw tissue underneath.



Day 27. Areas of obvious injury continue to grow larger, with no evidence of healing. Increasing pain in damaged areas.



About 5 years after the original exposure the individual has lost the complete index finger from his right hand and has experienced complete or partial loss of nine other finger joints. The hands are very sensitive to heat and cold and additional amputations may be required.

Fig. 43 - X-ray burns to the hands of an industrial radiographer

Medical Radionuclide Applications

Radioactive materials are used routinely in two different departments in medical centers - Nuclear Medicine and Radiation Oncology (Radiation Therapy). Most nuclear medicine procedures are diagnostic, i.e., they attempt to determine the nature or extent of a medical problem in a patient. Small amounts of a photon-emitting radioisotope are attached ("tagged") to a pharmaceutical which is administered by mouth or intravenously to the patient. The drug carries the radioisotope to the organ of interest where it is deposited. An external radiation detector is then scanned over the patient. The pattern of radiation detected from the radiopharmaceutical is used to form an image of the organ or to study the physiological functioning of the organ. Typical examples would be a thyroid scan or a lung function test.

Under ALARA, it is desirable to keep the radiation dose to the patient as small as possible while still obtaining the necessary medical information. This leads to two criteria for the radioisotope. The chosen isotope should decay by emitting photon radiation predominantly. Any particulate radiations (alpha or beta) will be locally absorbed in the organ and thus contribute to patient dose without contributing to the information signal. The metastable nuclei discussed in Chapter 2 are ideal. The decay from the nuclear excited state to the ground state produces only a gamma ray. Secondly, the physical half-life should be short. As will be shown in Chapter 9, the patient dose is directly proportional to the half-life. This means that radionuclides with half-lives of the order of a few hours are ideal. These half-lives are sufficiently long to last through a scan which may take a half hour and may have to be repeated. It is short enough to maintain patient doses ALARA.

Considering the state of the art of package transportation, it is not possible to routinely ship radionuclides with hour half-lives to points more than a few km from the isotope production site. However, these days, most urban areas do lie within the service area of a nuclear pharmacy. As of 1999, 85% of the nuclear medicine doses administered in the U.S. come from a central nuclear pharmacy. Sufficient numbers of pre-measured doses of various radioactive drugs are delivered daily to local hospital nuclear medicine departments. This saves time and also puts the quality assurance burden on the pharmacy rather than the hospital. The nuclear pharmacy makes the needed isotopes daily, on-site at its facility. This is accomplished by use of a radionuclide generator or radioactive cow. The cow consists of a porous "column" of material onto which is deposited a parent radionuclide which is chosen to decay to a radioactive daughter IN A META-STABLE STATE. The daughter activity is "milked" daily from the cow to provide the needed tag for the pharmaceuticals for that day's orders. The chief properties of the most common Tc-99m generator system are listed in Figure 44. Generator activities of 830 mCi up to 16.6 curies are commonly available today. The smaller sizes are used at large teaching hospitals for in-house procedures while the large activity generators are favored by the central nuclear

PARENT	DAUGHTER	HALF-LIFE	PHOTON ENERGY
Mo-99	Tc-99m	6 hours	140 keV

Fig. 44 - The most common radionuclide generator ("Cow")

Fig. 45 - The insides of a "radioactive cow"

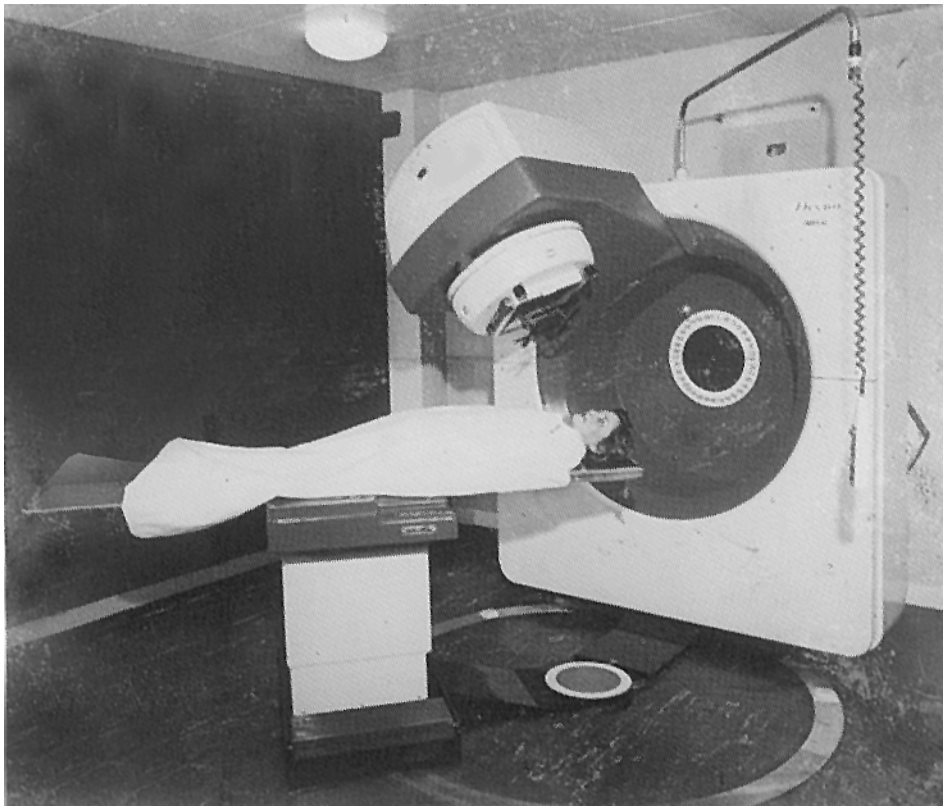
pharmacies as being more cost-effective. A photograph of the insides of a small radioactive cow is given in Figure 45.

In 1991, there were approximately 6000 licensed nuclear medicine departments in the USA using some 50 different radionuclides in over 150 different human diagnostic procedures. A 2006 estimate by the NCRP showed about 18 million procedures were performed that year involving administration of radionuclides to patients. Although over 150 different procedures are approved for use, only 10 account for over 90% of the *in vivo* procedures used in U.S. clinical practice. A brief list of some of the more common ones, along with the isotopes used and effective radiation doses to the patient, is given in Figure 46. Although nuclear medicine procedures are quite widespread throughout the U.S. population the doses to the organs are rather small, and the gonad doses almost negligible in many cases. The U.S. average annual effective dose was estimated by the NCRP to be 0.8 mSv per person in 1996.

Radiation Therapy (more commonly known in medical circles as Radiation Oncology) is the other hospital department that uses radionuclides. These are employed in two ways. The more common is the use of high activity sealed gamma ray sources for external beam tumor treatment. In this country, the radioisotope cobalt-60 is used almost exclusively. In Canada, use is also made of cesium-137 for such treatments. In the U.S., hospitals have almost totally replaced Co-60 units with linear accelerators (linacs) which were mentioned earlier for this application. The advantage for the cobalt-60 unit is the relative simplicity of the equipment. Thus, those units are still quite popular in the less developed countries around the globe. A mechanical device moves the Co-60 source to an opening in the collimator to project a beam of

Procedure	Nuclide	Activity (MBq)	Effective Dose (mSv)
Thyroid Uptake	I-123	7.4	0.06
Hyperthyroidism	I-131	185	11.5
Heart Perfusion	Tl-201/Tc-99m	150/1480	17.7
Kidney	Tc-99m	370	2.2
Lung Ventilation	Xe-133	740	0.5
Lung Perfusion	Tc-99m	185	2.0
Bone Scan	Tc-99m	1110	6.3
Brain PET	F-18	740	14.1

Fig. 46 - Typical patient effective doses from nuclear medicine procedures, 2006



Courtesy of CGR MeV and CGR Medical Corp.

Fig. 47 - Co-60 treatment room

photons. Reversing the source terminates the treatment. A disadvantage is that dose rates are not as high as desirable. A typical cobalt-60 unit with a fresh 6,000 curie source will deliver about 1 Gy/min to the tumor, while the usual linac is capable of producing 0.5 to 10 Gy/min. Figure 47 shows a typical Cobalt-60 treatment facility.

The other common use of radionuclides in a radiation oncology setting is for implant therapy or brachytherapy. A variety of radioisotopes, including Ra-226, Rn-222, Cs-137, Ir-192, I-125, Pd-103 and Au-198, are used to treat malignancies by placing the source near or inside the affected tissues. An intracavitary implant uses a body cavity such as the rectum or vagina. An interstitial implant uses needles containing the radioisotope which are inserted into the tissue. Figure 48 is a radiograph of an interstitial treatment of a tumor at the base of the tongue and also shows an intrauterine treatment procedure for cervical cancer. In 1991, about 50,000 brachytherapy treatments per year were performed.

In its 1987 Report 93, the NCRP attempted to estimate the contribution to the U.S. average GSD from radiation oncology procedures. In fact, since only a very small fraction of the U.S. population receives such treatment in a given year, and the treated patients receive thousands of rem, the concept of average GSD is probably meaningless. The value arrived at by the NCRP then was about 1 mrem per person. The 2009 NCRP Report 160 chose not to report GSD estimates or the average population effective dose. The number of external beam radiotherapy patients in the U.S. in 2006 was given as 980,000. The average effective dose received per patient was 0.4 Sv.

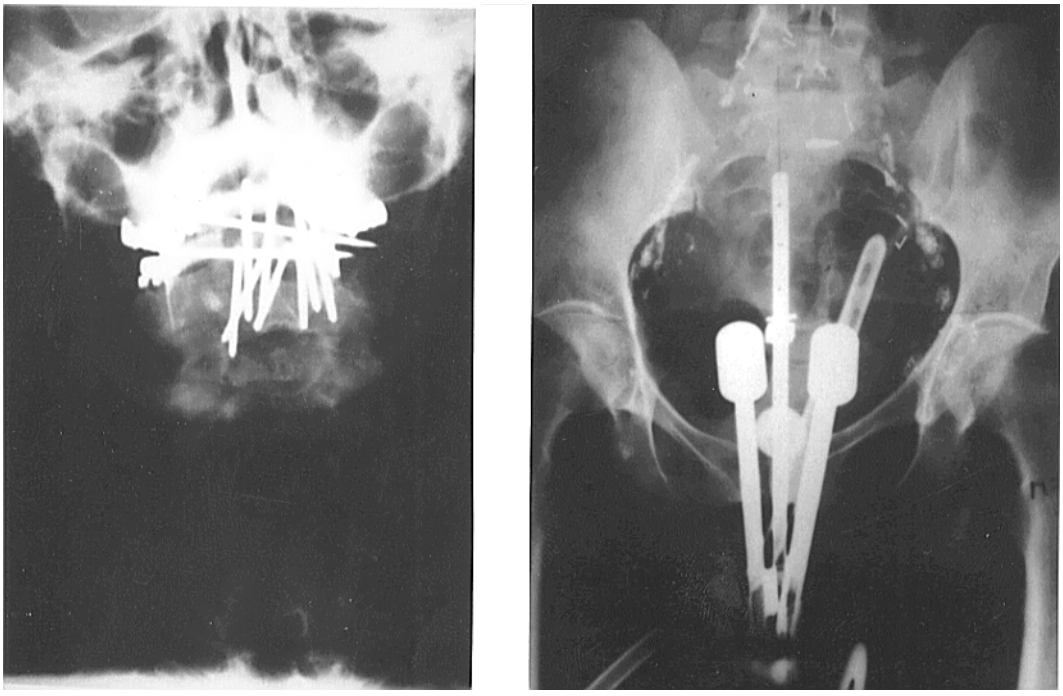


Fig. 48 - Radium treatment of tongue tumor (left) and cervical cancer (right)

Nuclear Particle Accelerators

All nuclear accelerators contain a number of common components. These are indicated by the block diagram in Figure 49. The basic objective is to produce a high energy stream of ions, directed along some path. This is accomplished by generating ions and then causing them to accelerate through a large potential difference. The fundamental relationship between the ion energy and the potential difference is shown in Figure 50. Take care to note the distinction between electron volt (eV is a unit of energy) and volt (V is a unit of potential difference). By definition, one electron volt is the energy acquired by a particle with one electronic charge, "e", falling through a potential difference of one volt.

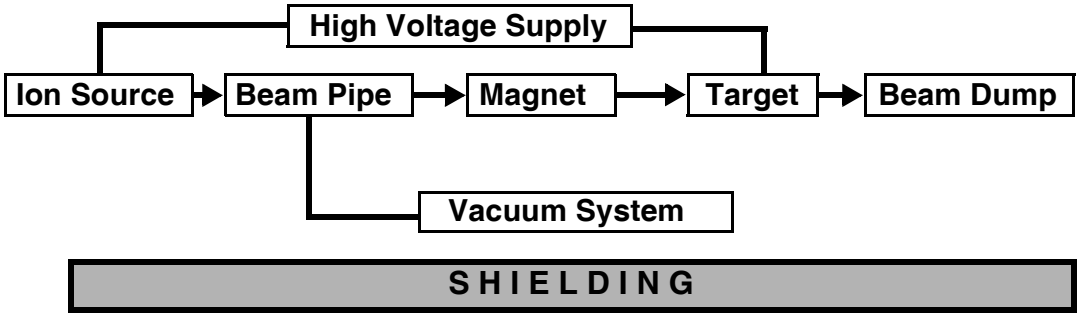


Fig. 49 - The common components of nuclear accelerators

Returning to the block diagram in Figure 49, the ion source is the device for producing a plasma of free ions. In positive ion accelerators, it often is of the radio frequency stripping type. A high frequency alternating potential difference causes sufficient forces on the electrons of a neutral gas so that they break their binding forces (are stripped) to produce positive ions. By connecting a high voltage supply in such a way that a potential difference appears between the ion source and the target section, the ions accelerate toward the target under action of the Coulomb force. The different configurations by which the potential difference is supplied lead to the variety of modern accelerator types. Sample Problem 3 shows how the beam energy is calculated.

The beam pipe is an evacuated section through which the accelerated ions pass. A vacuum system removes enough of the air molecules to prevent loss of beam ions due to collisions with these extraneous molecules. Magnets are used in two ways. First, as a lens to focus the beam into a tight "pencil" and second as a deflection force to steer the beam along a desired path of beam line. The target section is where the "useful" work is done. In a research application, the effect of the beam on some object is studied at the target. In industrial applications, the beam is directed on some product which receives the dose to enhance its properties. (One example would be the

Energy (eV) = Charge (e) x Potential Difference (V)

Fig. 50 - Relationship between accelerated particle energy and potential difference

Sample Problem 3

GIVEN:

A nuclear particle accelerator applies a 1.5 kV potential difference to the accelerated ions on each of 1200 cycles before the beam is ejected.

FIND:

What is the maximum beam energy if protons are accelerated? Li^{+3} ions?

SOLUTION:

From figure 50, E(eV) = Q(e) x V(V). This energy is given each time the ion passes thru V, which in this case is 1200 times. Since the charge, in units of the electronic charge e, on a proton is 1, E = 1 x 1.5 kV x 10^3 V/kV x 1200= 1.8 x 10^6 eV or 1.8 MeV. Similarly, for the +3 lithium ion, E = 3 x 1.5 x 10^3 x 1200 = 5.4 x 10^6 eV or 5.4 MeV.

familiar, almost impossible to open "shrink wrap" plastic film packaging material. The radiation produces molecular crosslinks between a "sandwich" of multiple thin films of the plastic.) The beam dump is a section at the end of the beam pipe to remove any remaining energy from the beam and to safely dissipate the resulting heat. It often involves some form of water-cooled apparatus. Finally, the accelerator is surrounded by a biological shield to protect operating personnel and the general public.

Linear accelerators, or linacs, provide the acceleration along a straight line beam path. In low energy accelerators (defined to be accelerators which produce energies not exceeding 50 MeV) this is frequently done by connecting the full high voltage between source and target. For example, if 10 million volts were the supply voltage, the accelerator would then be capable of producing 10 MeV protons, 20 MeV helium ions (Z = 2), etc. Relatively low "high voltages" can be produced by a series of step-up transformers (the Cockcroft-Walton principle, used incidentally in the first man-made accelerator, in 1931) or by static charges sprayed on a belt and transferred to a conductor (the Van de Graaff principle). Examples of each of these two types are given in Figures 51 and 52.

High energy accelerators (beam energy in excess of 1 GeV = 1,000 MeV) often are of the traveling microwave type. Pulses of radio frequency microwave power produced by klystrons are injected into a microwave waveguide and lead to a traveling wavefront that carries ions along under continuous acceleration similar to a surfer riding a water wave crest. The longer the ride, the higher the energy. The world's longest electron machine of this type is at the SLAC National Accelerator Laboratory, on the campus of Stanford University in California. The waveguide is 2 miles long. Originally built to accelerate electrons to 20 GeV, the accelerator was upgraded to produce electrons at 52 GeV of energy in pulses 0.1 microsecond long at an average current of about 1 microamp. An aerial photo of the site is shown in Figure 53. A view of the underground accelerator gallery is shown in Figure 54.

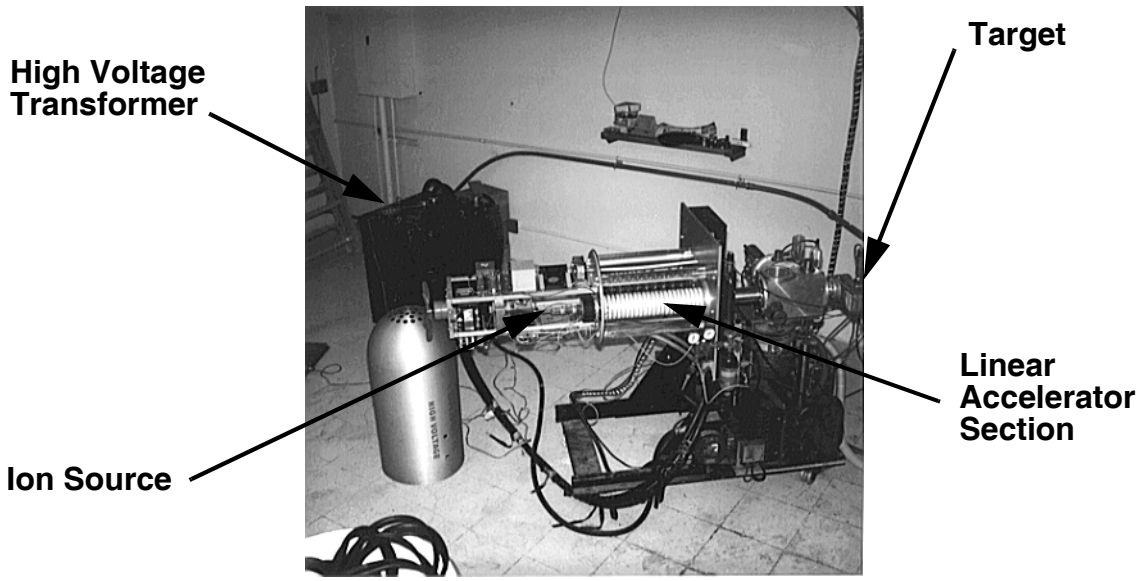


Fig. 51 - Cal State University Los Angeles Cockcroft-Walton accelerator

The Stanford Linear Collider was completed in 1987. This consisted of adding a positron beam to the accelerator and then allowing the positrons to collide head-on with the electron beam. Simultaneous electron and positron beams have been produced with a collision energy of 114 GeV being achieved.

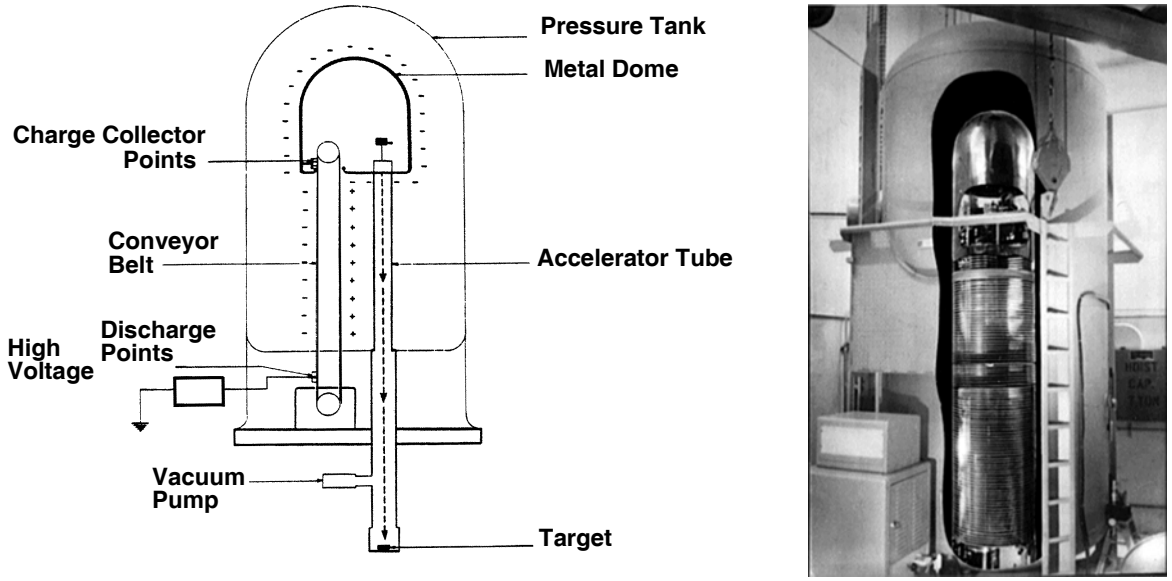


Fig. 52 - Cal State University Los Angeles 4 MeV Van de Graaff accelerator

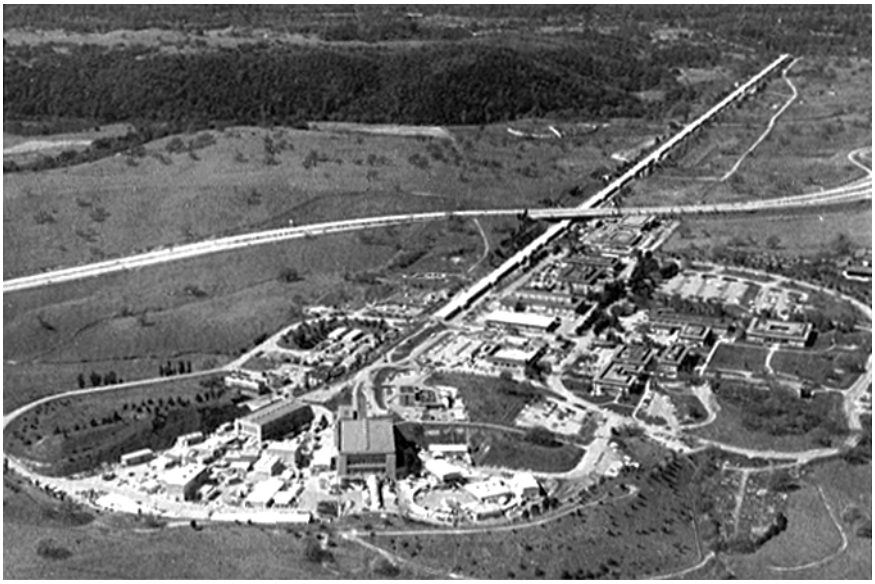


Fig. 53 - Aerial view of the Stanford Linear Accelerator Center

Cyclic accelerators cause the ion beam to travel in a roughly circular path. The earliest of this type was constructed in 1931 by Lawrence and Livingston and was called a cyclotron. The evacuated accelerator section is in the shape of a pair of letter "D"s (each section called a dee) placed between the poles of a large electromagnet. The magnetic field confines the beam to a circular path. Each time the beam passes

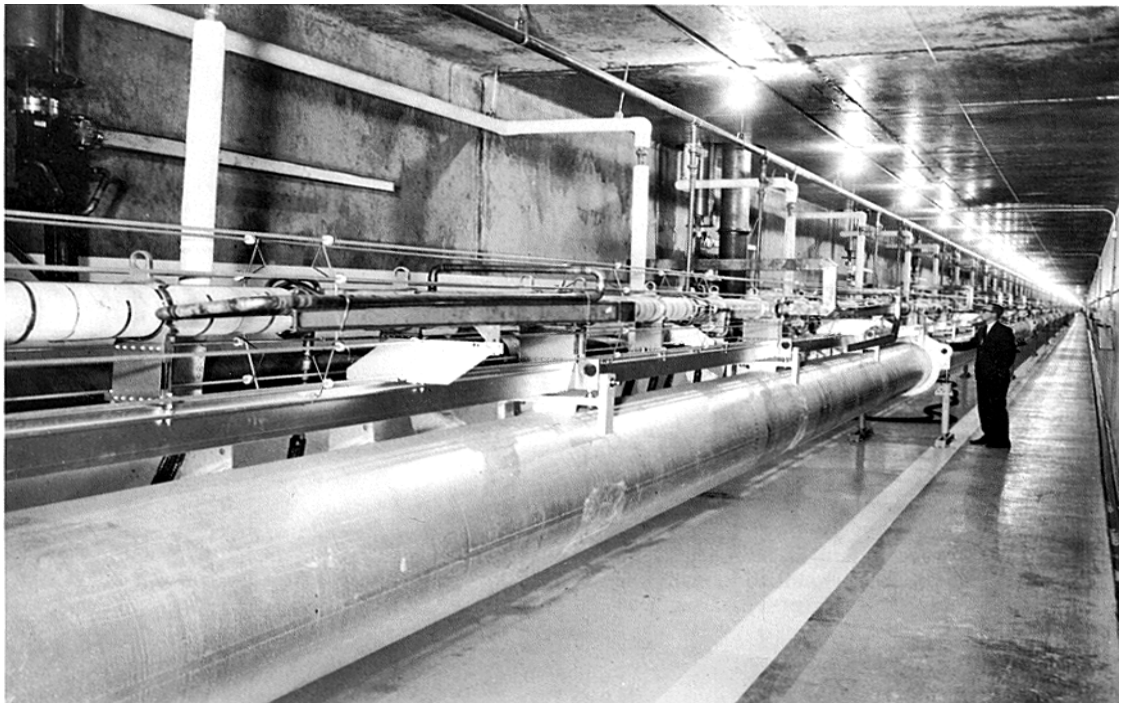
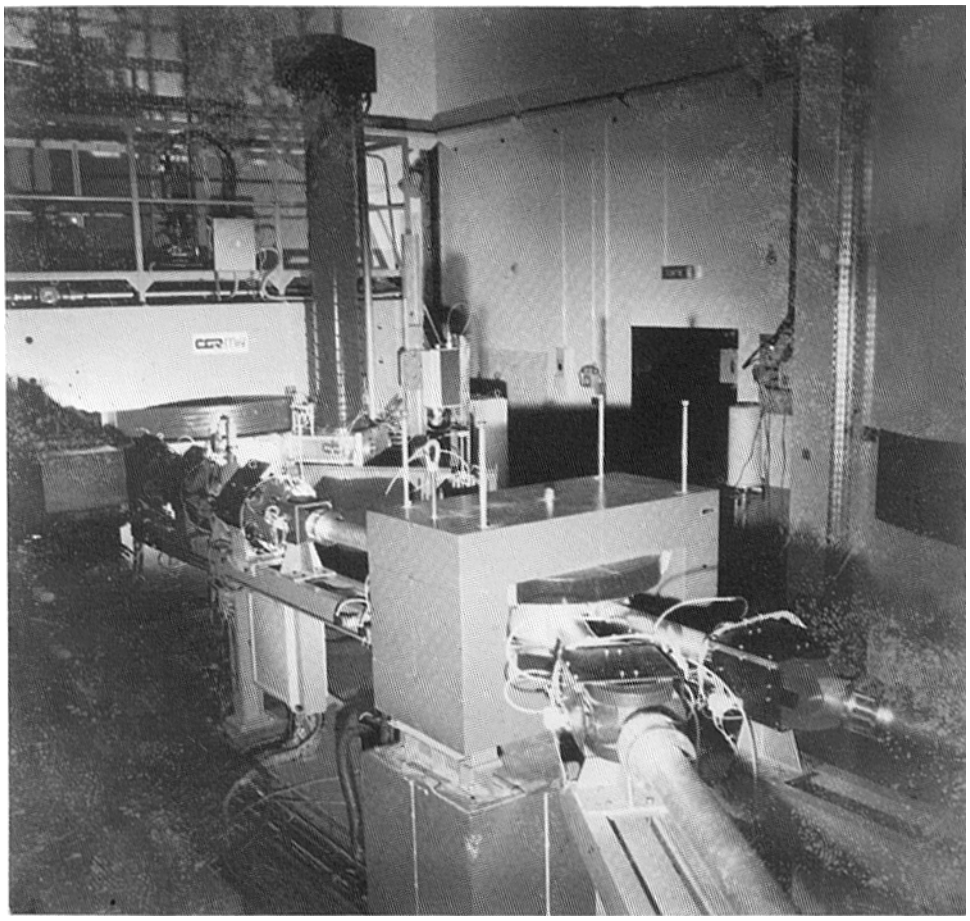


Fig. 54 - The SLAC underground accelerator gallery



Courtesy of CGR MeV and CGR Medical Corp.

Fig. 55 - A cyclotron laboratory

through the gap between the dees, acceleration takes place. The relatively small potential difference (e.g., 50 kV) is applied repeatedly so the particles emerge with energies of the order of 10's of MeV. Figure 55 shows a photo of a cyclotron that is used for medical radioisotope production.

Major design improvements ultimately led to the development of the synchrocyclotron. This machine uses a variable frequency oscillator connected across the dees to produce energies in the hundreds of MeV range for protons.

Both betatrons and synchrotrons are also cyclic accelerators, but the beam path is confined to a constant diameter doughnut shaped vacuum vessel. The betatron is used to accelerate electrons, and has a large magnet with pole faces the diameter of the doughnut. Figure 56 is a drawing of a betatron. A synchrotron has smaller magnets arranged only along the doughnut ring and is used for higher electron energies. As of 2011, there were 36 electron and proton synchrotron facilities operating worldwide. Many have been designed specifically for the production of intense ultraviolet light and low energy x-rays ("synchrotron radiation") for research in biology, chemistry, geology, medicine and physics. See Figure 57.

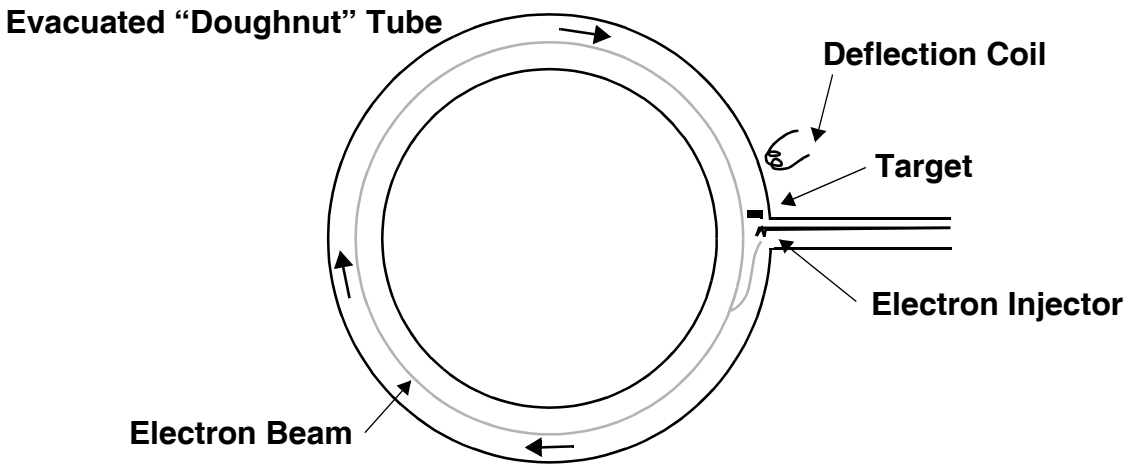


Fig. 56 - Components of a betatron (magnetic field perpendicular to the page)

As of 2011, the highest energy nuclear accelerator in the United States is the proton synchrotron (named the Tevatron) at Fermilab in Batavia, Illinois. It has reached a beam energy of 1.96 TeV. This energy is achieved by accelerating both protons and antiprotons to 0.98 TeV in the same ring (but they travel in opposite directions) and then bringing them together in a head-on collision. An aerial view of the

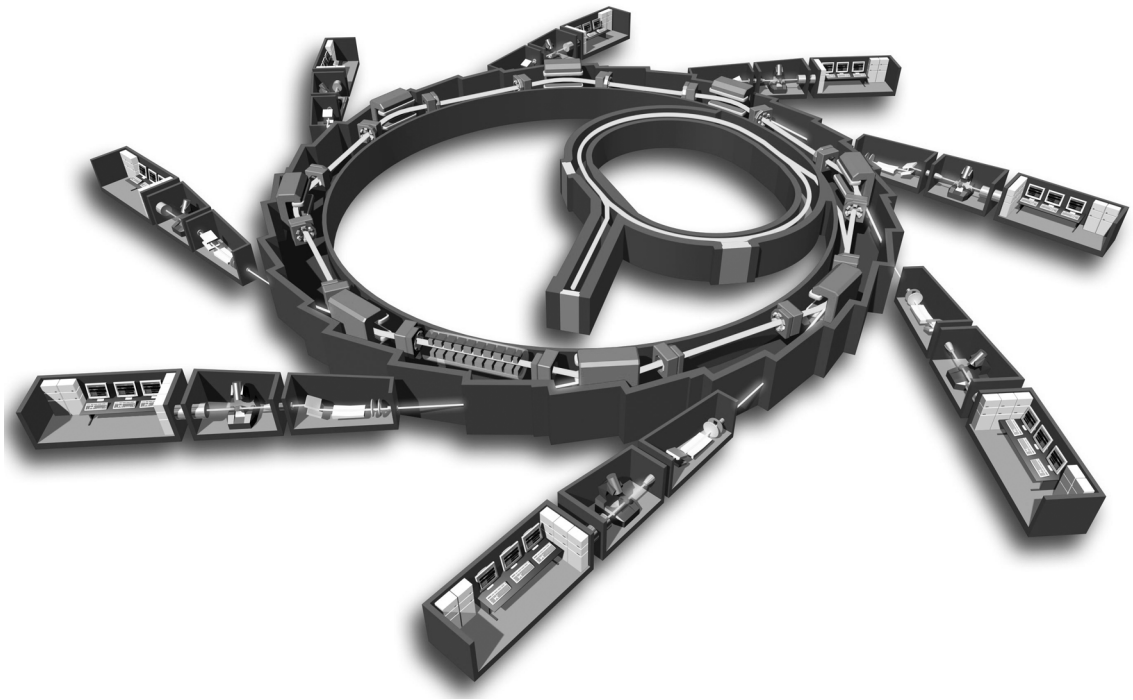


Fig. 57 - Schematic of a synchrotron laboratory

Fermilab Photo

Fig. 58 - Aerial view of Fermilab and vicinity

Fermilab complex is shown in Figure 58. The main ring of the Tevatron is four miles in circumference! It is buried below ground. Figure 59 is a view inside the underground circular tunnel and Figure 60 is the proton anti-proton Collider Detector. It

Fermilab Photo

Fig. 59 - Fermilab underground accelerator tunnel

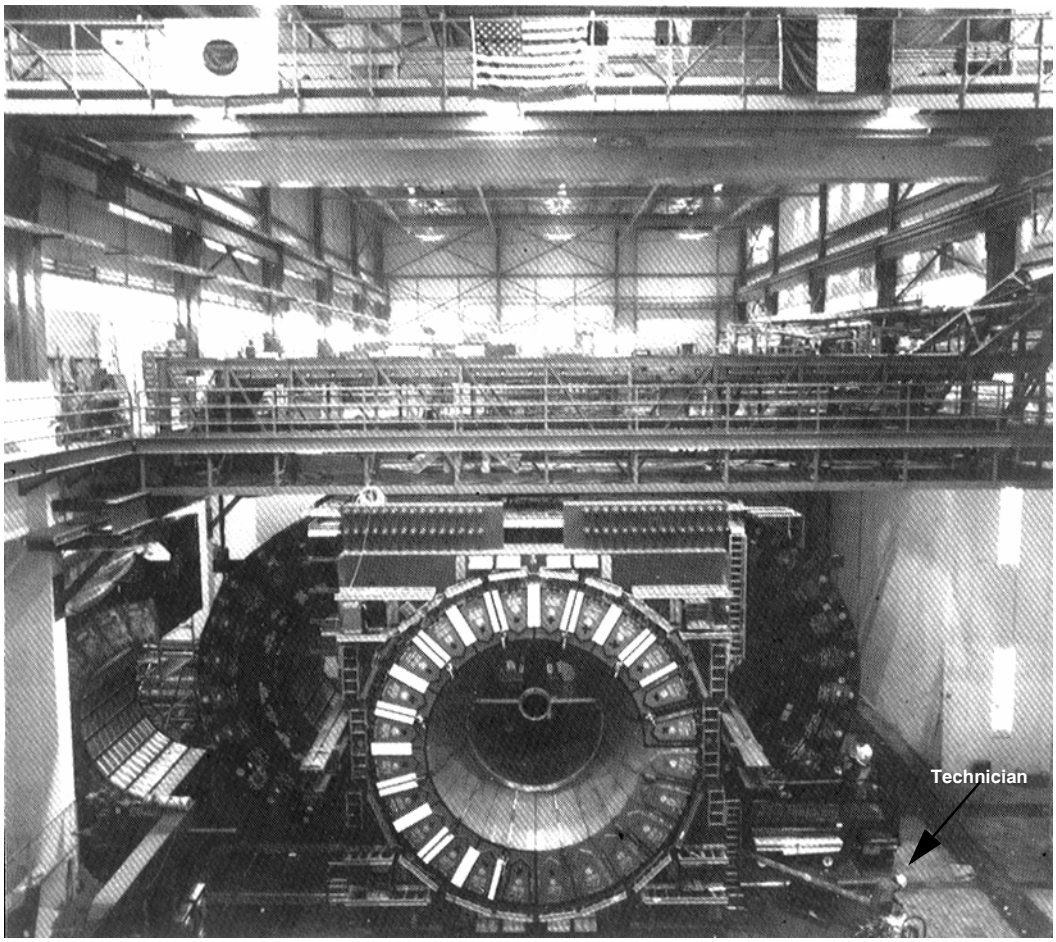


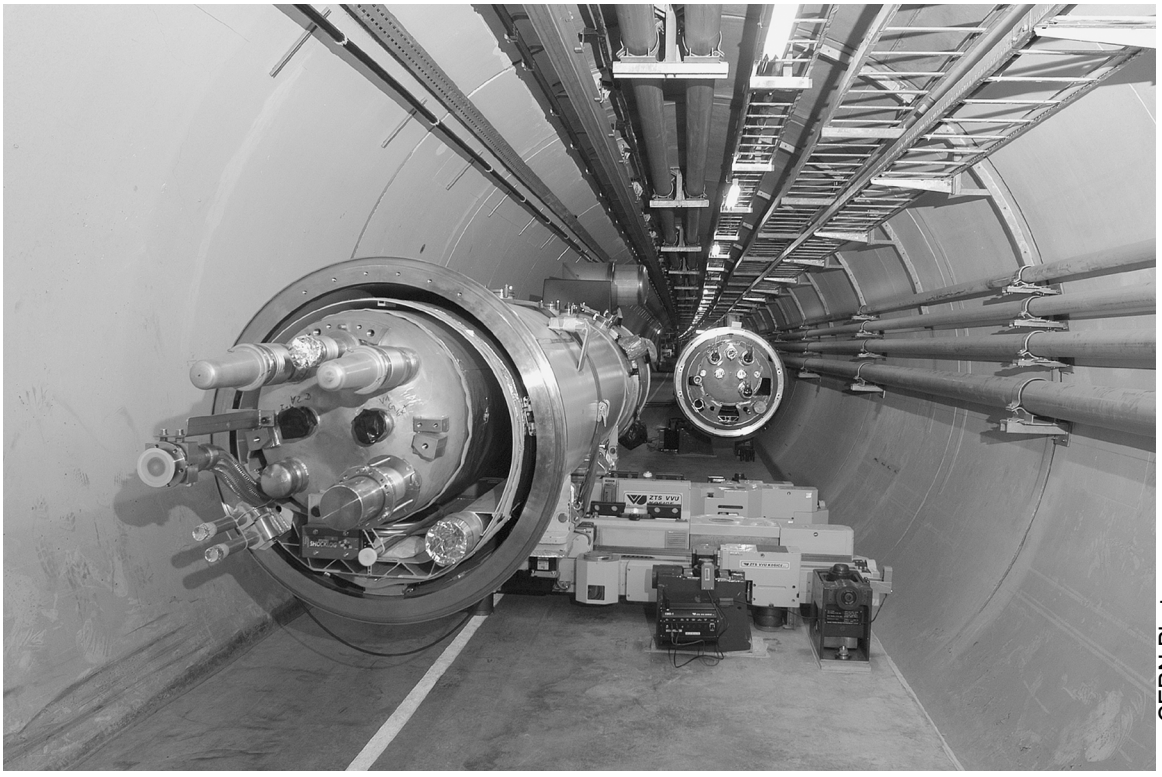
Fig. 60 - Collider Detector at Fermilab

weighs 4500 tons and stands two stories high. Figure 61 is a photo of Wilson Hall, the 16 story high administration building for the Fermilab complex.

The 2010 world record holder for high energy accelerators is the Large Hadron Collider at CERN, the European Organization for Nuclear Research in Geneva Switzerland. The machine accelerates two beams of protons and then allows them to smack into each other to eventually produce a collision energy of 14 TeV! The LHC reached 7 TeV in March, 2010. All of this action takes place in a circular underground tunnel, about 17 miles in circumference and 100 meters below ground level. Most of the tunnel actually underlies France which shares a common border with Switzerland. Initial machine operations produced a proton beam in 2008. More than 1,600 superconducting magnets (27 tons each) are employed to confine the two proton beams. Figure 62 shows a view down the underground tunnel with a superconducting magnet under test and Figure 63 shows the layout geographically.

The LHC was built to shed light on some very fundamental physics and astrophysics questions. The Standard Model of Matter (Chapter 1) requires the existence of an exotic particle named the Higgs Boson. If theory is correct, it will explain why photons have no mass while quarks do. It is also hoped the LHC will finally explain Dark Matter, the unknown 80% of all mass in our known universe.

Fig. 61 - The Fermilab main administration building, Wilson Hall



CERN Photo

Fig. 62 - The LHC tunnel with a superconducting magnet

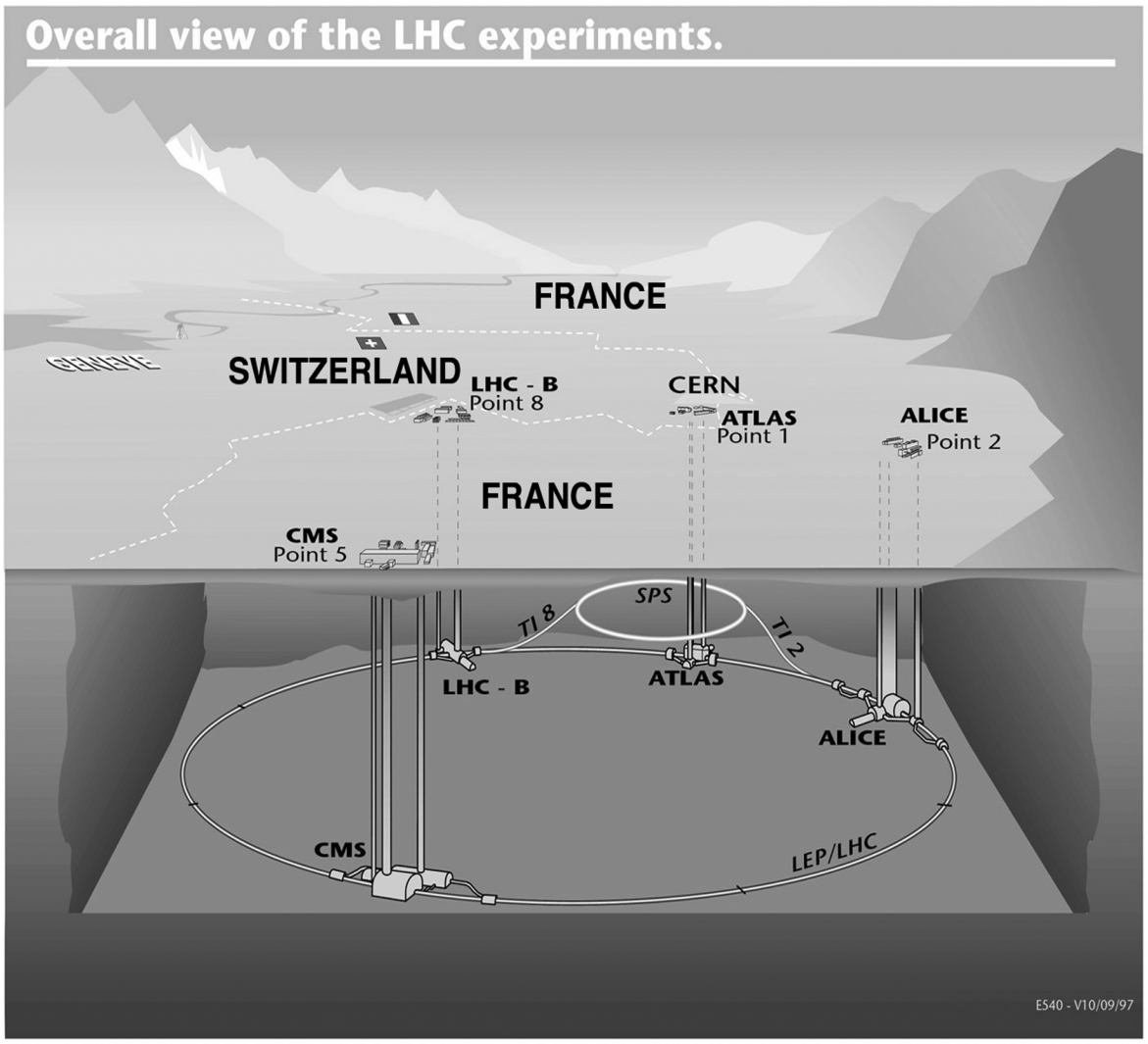


Fig. 63 - The CERN LHC accelerator site

From the point of view of radiation protection technology, there are two types of accelerator radiation fields of interest. The PROMPT FIELD is present only when the nuclear accelerator beam is on. In electron machines, the prompt field is dominated by bremsstrahlung photons at all electron energies. In the case of electron machines operating above about 10 MeV, there is an additional problem from neutrons produced in photonuclear reactions caused by high energy bremsstrahlung. As the energy of the electron machine is increased, the physics of the interactions shows that the prompt field becomes much stronger along the axis of the beam. Prompt fields perpendicular to the beam line are typically less than 5% of the exposure rate measured parallel to the beam. In positive ion accelerators, neutrons dominate the prompt field in low and medium energy machines. Neutrons result from nuclear reactions in which accelerating ions induce neutron emission from internal accelerator

components. In high energy machines (>1,000 MeV), muons are an added complication. Muons are produced from the decay of short-lived pi and K mesons which are produced when the high energy ion collides with the target nuclei.

The INDUCED FIELD is the second nuclear accelerator radiation field of concern. This field consists of the emissions of radioactive nuclei which have been activated by prompt neutrons striking machine components, the surrounding shielding and the air inside the accelerator vault. There are many different half-lives involved. In a typical facility, the dose rate often falls by a factor of 2 in 50 hours. In concrete shields, Na-24 is the dominant gamma ray emitter. Air activation products include O-15, N-13, C-11 and Ar-41.

Experience at many positive ion accelerators indicates that the induced field, after shutdown, is the largest source of personnel exposures at these facilities. As just mentioned, most of this field comes from the concrete shielding in the accelerator vault. Maintenance technicians are exposed inside the vault during repairs or changes in the experimental configuration. Unfortunately, it takes time for this field to decay and, when shutdown, the facility is not earning its keep. The radiation protection supervisor must use judgement in weighing the costs of higher doses versus downtime at a large accelerator complex.

Technologists should also be aware of the fact that many accelerator prompt radiation fields are, by nature, pulsed. This means that the actual radiation level produced during operation consists of a continuous series of bursts of radiation. During acceleration, the physics of the design causes ions to bunch together in groups that are accelerated. When this succession of ion groups hits the target section of the machine, the short bursts of radiation are produced. Each burst might typically be of the order of one microsecond long. Between bursts, essentially no prompt radiation is released. This pulsed nature of accelerators has serious implications for radiation monitoring. As will be seen in Chapter 12, many radiation survey meters are incapable of correctly measuring the field when it consists of short pulses.

Nuclear Reactors

Nuclear reactors are such an important sector of the radiation protection technology field that an entire chapter is devoted to the details of this subject - see Chapter S-1 near the end of the book. The highlights of reactor design and use will be covered in this Chapter. There are several different types of nuclear reactors in which controlled nuclear fission is used as a source of either neutrons or heat. The two broad categories which cover the various designs are research, or non-power reactors (which utilize the neutrons) and power reactors (which utilize the heat). Before discussing the different designs, a review of the physics of fission is in order.

As shown from the average binding energy curve of Chapter 2, fission releases about 200 MeV of energy per atom in the form of kinetic energy of the fission fragments. Since these are large charged particles (average atomic mass numbers = about 140 and 95) they have a huge stopping power and so their range will be short. This means that the energy will show up in the form of heat generated close to the fission site. In the nuclear reactor, this heat energy appears in the primary coolant. To produce heat energy at the rate of 1 joule per second (i.e., 1 watt), a huge number of separate atoms must undergo fission.

K_{eff} = Effective Multiplication Constant

K_{eff} <1 = Subcritical Condition

K_{eff} =1 = Critical Condition

K_{eff} >1 = Supercritical Condition

Fig. 64 - The effective multiplication constant

The actual relationship between the power and fission rate is:

1 watt = 3.3 x 10^{10} fissions/sec. **[Eqn. 1]**

The effective multiplication constant, k_{eff}, is needed to understand the principle of reactor start-up and control. It is defined as the ratio of the number of neutrons in the reactor in one generation (at some point in time) to the number of neutrons in the PREVIOUS generation (an instant earlier in time). On the average, 2.5 neutrons are emitted per uranium fission. If k_{eff} has a value >1, the neutron flux is increasing and conversely, if it has a value <1, the flux is decreasing with time. Figure 64 illustrates the reactor condition for various values of the multiplication constant.

In a subcritical reactor (k_{eff} is <1), the neutron flux and power output will die off in time. When critical, the reactor operates at a steady neutron and power output. A reactor must be supercritical to produce any increase in the neutron flux and power level. See Sample Problem 4 for a calculational example.

Sample Problem 4

GIVEN:

A 2970 MW_{th} nuclear power reactor experiences a sudden increase in K_{eff} from 1.000 to 1.005.

FIND:

The neutron production rate in the first generation after the change.

SOLUTION:

The neutron rate before the change was 2970 MW x 10^6 W/MW

x 3.3 x 10^{10} fission/W-sec x 2.5 neutron/fission = 2.45 x 10^{20} neutrons/sec.

After the increase, the number in the first generation will be 1.005 times larger (from the definition of k_{eff}) = 1.005 x 2.45 x 10^{20} neutrons/sec = 2.46 x 10^{20} neutrons/second.

All reactors have a number of components in common. These are listed in Figure 65 and shown in the sketch in Figure 66. The most common fuel is uranium, either as it naturally occurs or enriched in the isotope U-235. Remember, only the U-235 captures thermal neutrons and fissions. The usual moderators are ordinary water, heavy water (deuterium substituted for hydrogen in the molecule), beryllium or

FUEL: Source of Uranium-235

MODERATOR: Slows down the fast neutrons

COOLANT: Removes the heat of fission

REFLECTOR: Reduces neutron leakage

CONTROL RODS: Adjusts the value of K_{eff}

SHIELD: Protects operating personnel

Fig. 65 - The common components of a nuclear reactor

carbon (in the form of graphite). Popular coolants include water, liquid sodium metal (in the fast breeder reactor), and carbon dioxide or helium (in the gas-cooled reactor). Reflectors are designed to scatter neutrons heading out of the core region back into the core where they have a chance to induce fission. Beryllium and carbon are often used. The function of the control rods is to absorb neutrons and thus lower the effective multiplication constant. Substances with high neutron capture probability are used. These include boron, cadmium, hafnium, indium and silver. Finally, common shield materials include water, concrete, steel and earth.

To start-up the reactor, a control rod is slowly pulled out until k_{eff} is just slightly above 1.000. The neutron flux from the internal "start-up source" is then multiplied. The neutron flux or power is closely monitored until the desired operating level is reached. Then, the control rod is inserted far enough back into the core to make the multiplication exactly 1.000. In theory, at this point the reactor will operate at a constant power level. In practice, temperature changes in the components, buildup of neutron absorbing "poisons" from fission products, and other factors will require many small changes in control rod positions to maintain k_{eff} exactly = 1.000.

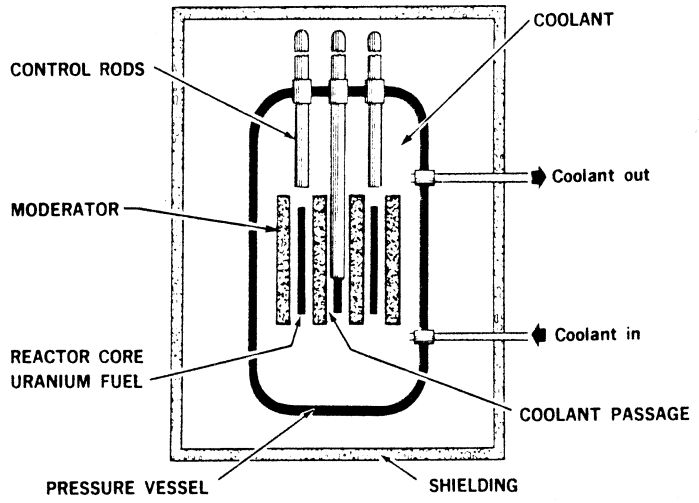


Fig. 66 - Reactor components illustrated

Non-power (Research) reactors are designed primarily as an intense neutron source and for training or isotope production purposes. As of 2009, there were 240 operating reactors in this class in 56 countries worldwide. The U.S. NRC had active licenses issued to 31 research reactors during 2010. The majority are located on college and university campuses. The power levels range from 1 watt in the smallest (a training reactor) to a few megawatts in the larger research facilities. The U.S. Department of Energy had 15 active licensed research reactors in 1999. The largest DOE licensed research reactor has a thermal power of 250 megawatts. In addition to the operational reactors, in the U.S. there are 148 research reactors currently in shutdown mode and 82 have been decommissioned through 2010.

Many of these non-power reactors operate with highly enriched uranium (HEU) fuel. Enriched uranium means the concentration of the ^{235}U isotope has been artificially increased above the natural abundance level of 0.72%. "High enrichment" means the U-235 is above 20%. A typical 100 kW research reactor might contain 3 to 4 kg of uranium enriched to 93%. Material with enrichment this high is termed "weapons grade" uranium as this level of enrichment is needed to produce a nuclear detonation. In an attempt to reduce the threat of nuclear proliferation, the U.S. DOE has been encouraging the operators of research reactors to switch to fuel enriched to less than 20%. As of 2005, some 60 research reactors worldwide still used HEU. A goal is to have all converted by 2013. In a few cases, this will require development of a new fuel type.

The "swimming pool reactor," shown in Figures 67 and 68 was a popular design in the 60s and 70s for research reactors. The pool was a water tank about 25 feet deep. The water played multiple roles - shield, reflector, coolant and moderator. It also allowed easy access to the core for maintenance or for the irradiation of bulky,

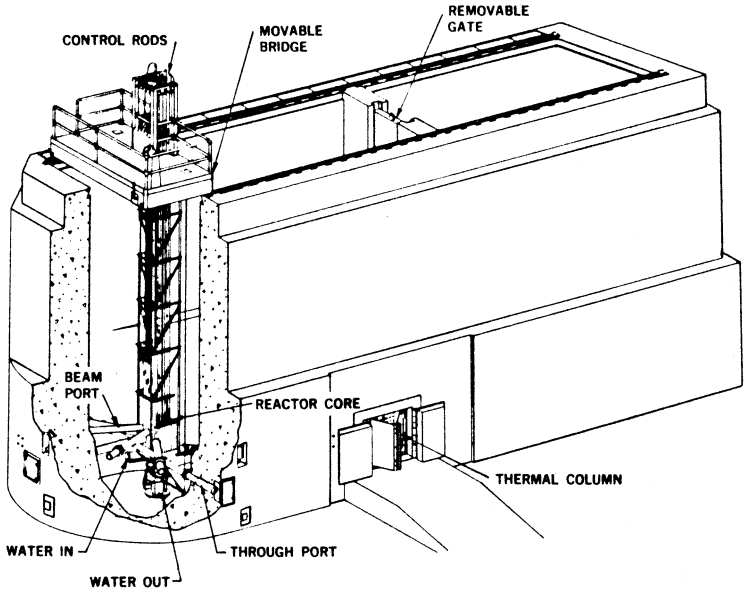


Fig. 67 - A swimming pool research reactor

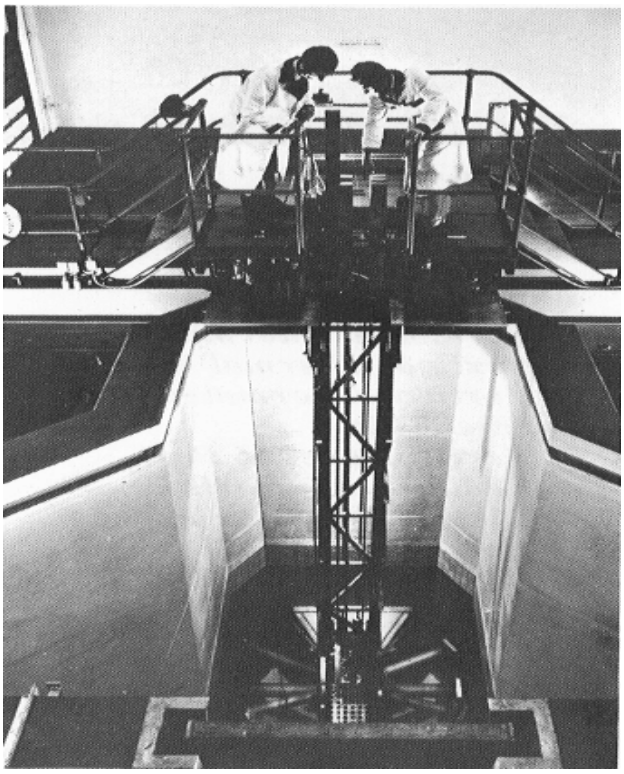


Fig. 68 - The view inside a swimming pool reactor

irregular-shaped objects – these could be lowered on a rope over the side of the pool. As of 2010, there were 4 operating pool reactors in the United States. The thermal power ratings for these pool reactors ranged from 1 kilowatt to 10 megawatts. At levels much above this, sufficient radioactive gases are released off the surface of the water to pose a significant hazard to personnel. One further example of a research reactor for medical uses is shown in Figure 69.

Power reactors are designed specifically as a heat source to produce steam to drive an electric turbine generator (see Figure 70). Most plants in the U.S. use uranium enriched to between 1% and 5% with the U-235 isotope. The uranium is in oxide form and is made into pellets which are about 1 cm in diameter and 1.5 cm long. The pellets are then encapsulated into 4 meter long tubes of Zircaloy cladding. A bunch of tubes are then placed into a fixture which becomes a fuel element. A power reactor may have a core inventory of from 150 to 600 fuel elements (or assemblies or bundles) containing a total of about 100 tons of uranium.

The electrical generating capacities of nuclear power plants vary considerably. As of 2010, the smallest U.S. plant online had an electrical output of 470 MW_e (Fort Calhoun in Nebraska), and the largest was 1410 MW of electrical power (South Texas Project). The “typical” size plant is about 1100 to 1200 MW_e (= megawatts, electric).

The fraction of U.S. electricity generated by nuclear power has gradually increased since the first plant came online in 1957. In 1985, nuclear accounted for 15% of all U.S. generated electricity. In 2010, that fraction rose to 20.2%. Nuclear exceeded oil by 20 times, renewable energy sources by 5.8 times, and hydroelectric by

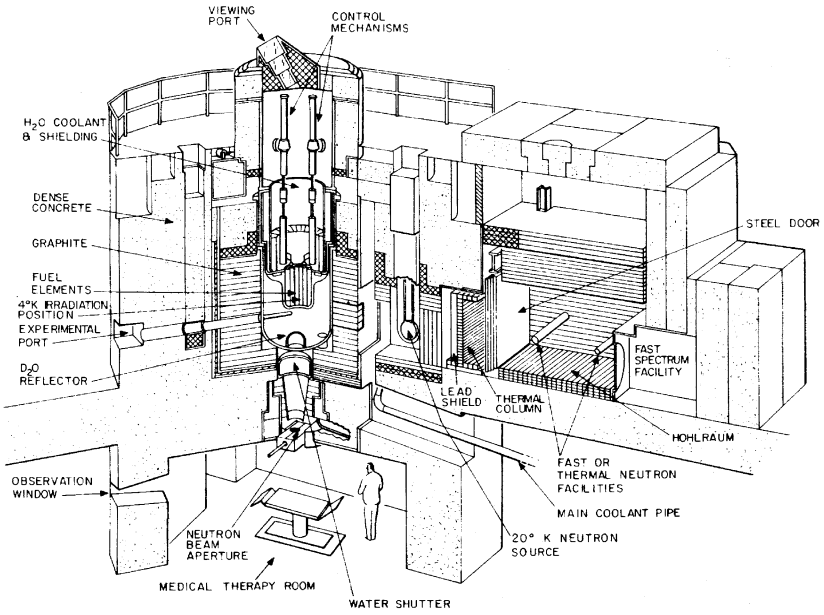


Fig. 69 - The MIT medical research and treatment reactor

Courtesy of MIT Nuclear Research Laboratory

3 times. Coal and gas sources exceeded nuclear, accounting for 70.5%. See Figure 71. The average cost to the U.S. consumer of nuclear electricity was 5.74 cents per kilowatt-hour in 1990. Production expenses for nuclear electricity continue to remain lower than for fossil fuel plants. In 2008, nuclear costs were $21.16 per MW-hr vs. fossil plant costs of $35.67.

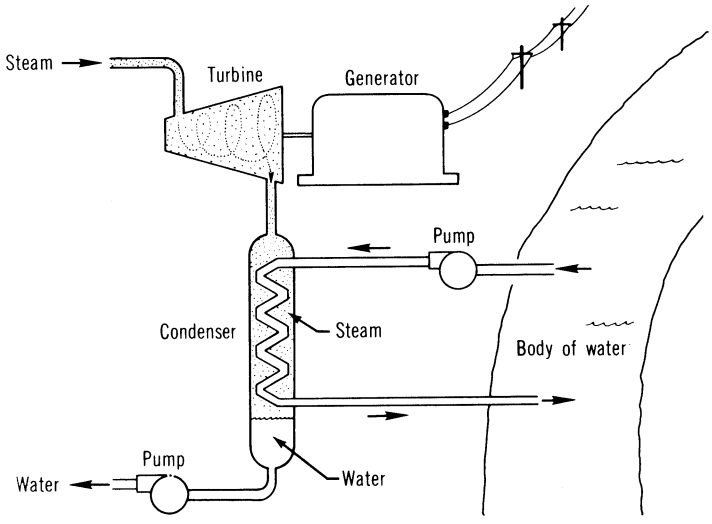


Fig. 70 - The power generating side of a power reactor

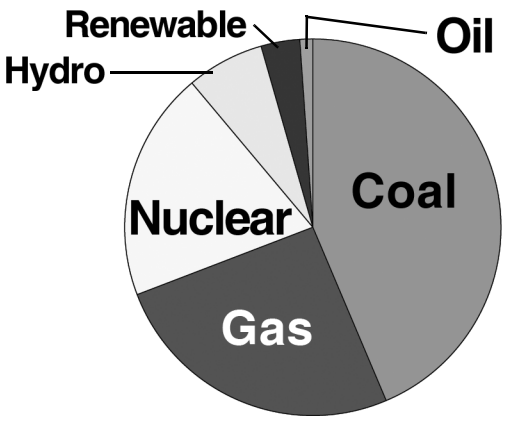


Fig. 71 - U.S. electrical generation by energy source - 2009

Worldwide, nuclear plants accounted for 24% of the total electricity generated in 2004. In several countries the nuclear fraction of domestic electricity exceeds that of the U.S. Figure 72 shows countries with a significant commitment to nuclear power. As of 2010, there were 438 power reactors producing electricity in 30 countries. There were 54 power reactors under construction. Countries that had more than 10 operating nuclear plants include Canada (21), France (58), Federal Republic

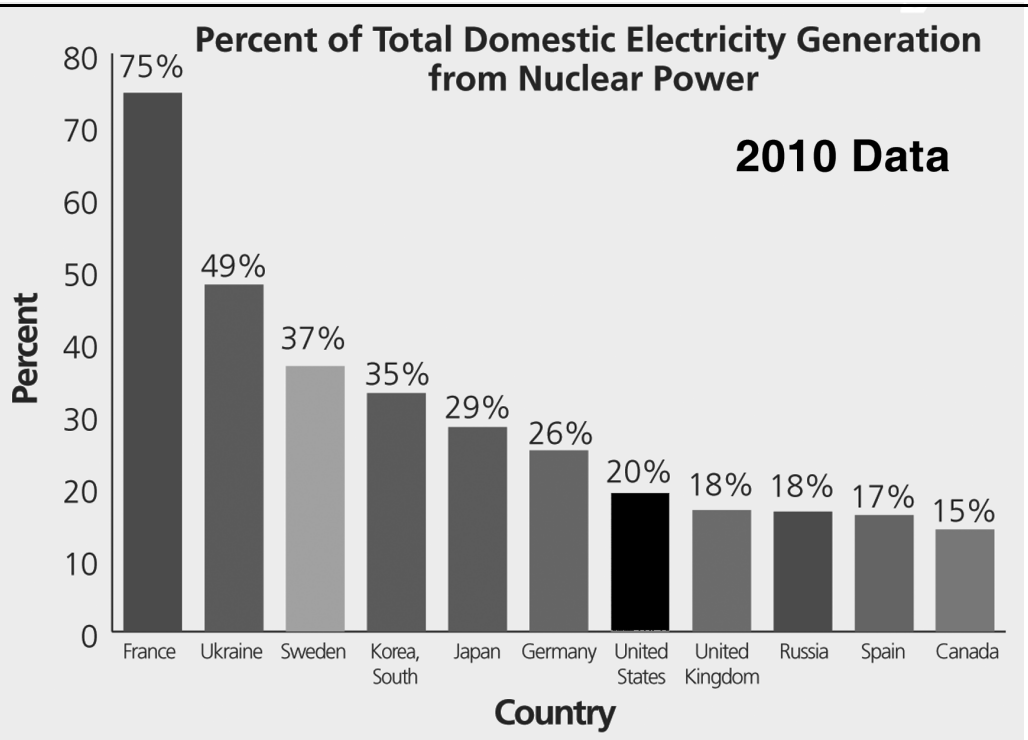


Fig. 72 - Major nuclear power countries, domestic nuclear electric generation - 2010

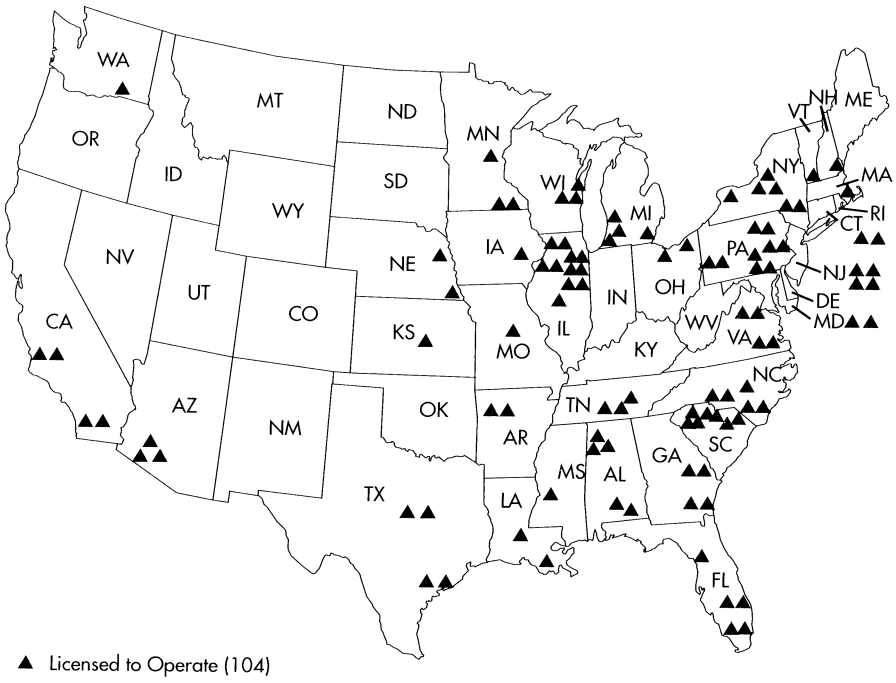


Fig. 73 - Locations of U.S. nuclear power plants - 2010

of Germany (17), India (17), Japan (56), Russia (31), South Korea (20), Sweden (10), Ukraine (15), United Kingdom (19) and United States (104). China has 9.

There are two different design types for the commercial power plants now operating in the U.S. In 1982 at the time of the writing of the First Edition of this text, there were 74 licensed plants. In 2010 there were 104 plants with 80 different designs licensed to 26 operating companies located at 65 sites in 31 states (see the map in Figure 73). About two-thirds are pressurized water reactors, PWRs, and about one-third are boiling water reactors, BWRs. Since 1989 there are no longer any high temperature gas cooled reactors, HTGRs, operating in the U.S. The two commercial units built have been retired. Including these two retirees, there are a total of 28 power reactors in the U.S. which have been removed from service over the years. The differences in design of the various types are shown in Figure 74.

As of January 2011, there was one power reactor under construction in the U.S. (Watts Bar Unit 2 in TN). During the period 2007 - 2010 the NRC received applications for 28 new power reactors. Hopefully, many will be financed and built successfully.

In Vermont, 72% of electrical production was nuclear in 2010. In Connecticut, New Jersey and South Carolina, nuclear accounted for over half of the generated electricity (Figure 75).

The BWR has a single loop design (Figure 74). The coolant water, in contact with the core fuel elements, boils to form steam that passes through the turbine. This simplifies the plumbing, but spreads radioactive contamination over a much larger section of the total plant. The PWR uses a two loop design. See Figure 76 for a detailed block diagram. The pressure is kept high enough in the primary loop so the water

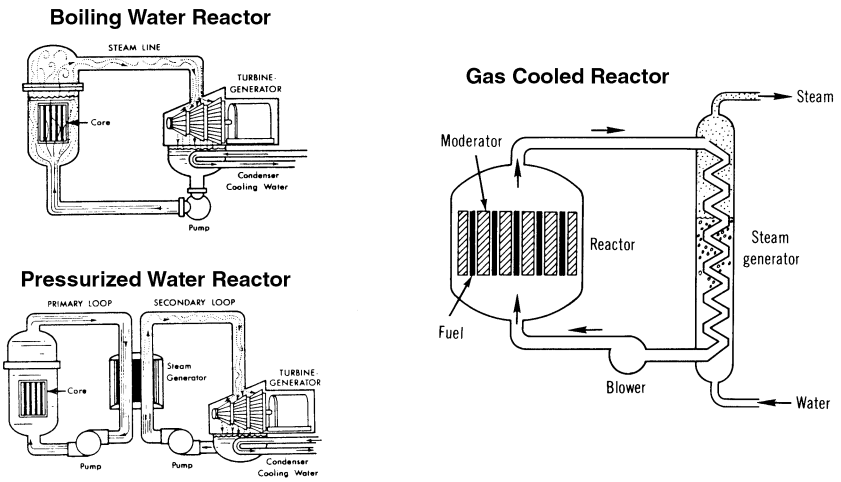


Fig. 74 - U.S. light water reactor designs

becomes superheated without boiling. Inside the steam generator, primary loop water passes near the secondary loop water and heat is transferred to the lower pressure secondary loop. It boils to power the turbine. There is more plumbing involved in this design, and the steam generator tubing tends to plug up causing long downtimes for

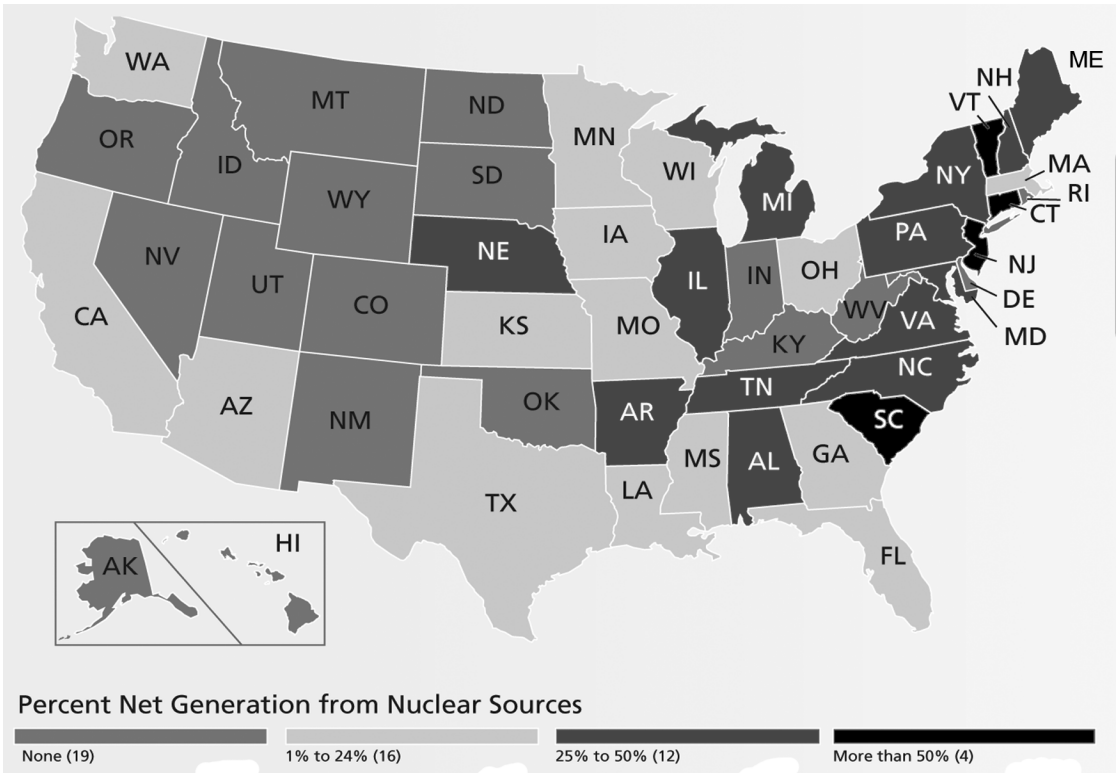


Fig. 75 - Nuclear electricity generation by state in 2010

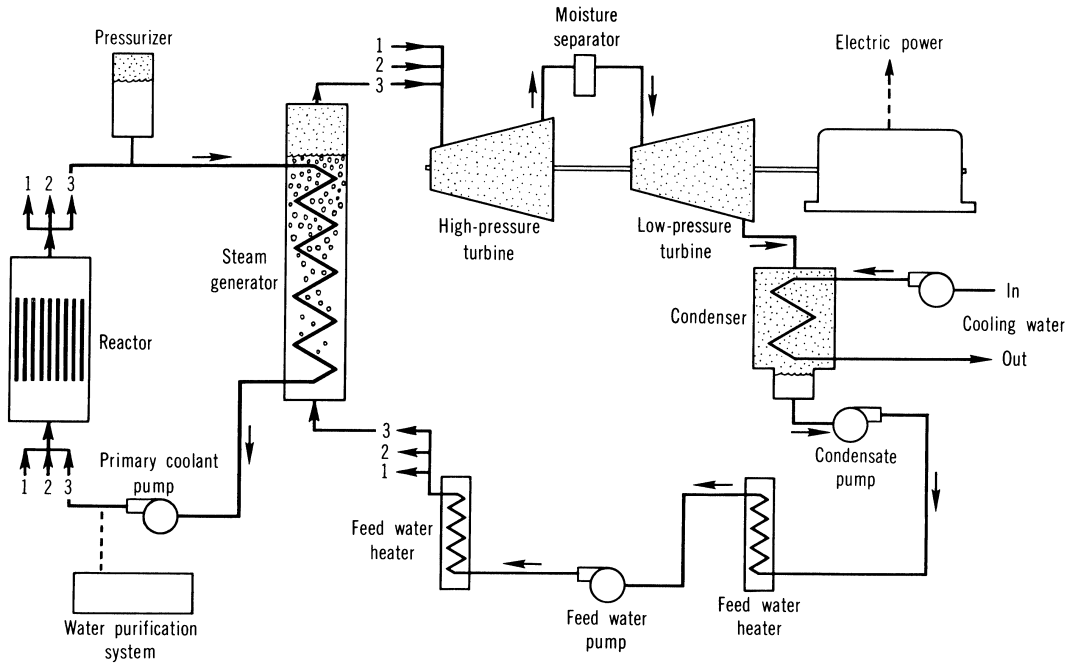


Fig. 76 - Block diagram of a pressurized water reactor

replacement, but primary coolant radioactivity is kept out of the turbine generator areas of the plant. The HTGR is a popular design in Europe. Figure 77 shows the prototype HTGR (built at Windscale, England) for a series of plants in Great Britain. A

Fig. 77 - The British prototype HTGR

Fig. 78 - A PWR nuclear power station

gas is used for cooling rather than water. This allows a somewhat more efficient heat transfer and thus higher electrical efficiency, the ratio of the electric energy output to the thermal energy output. The light water reactor's usual efficiency is about 31% to 32% while the HTGR gives about 40%. HTGRs require a higher degree of U-235 enrichment. Figure 78 is a photo of a typical PWR station.

The drawing in Figure 79 shows the schematic layout of the CANDU (CANadian Deuterium Uranium) design, developed in Canada. It features horizontal fuel elements of natural (non-enriched) uranium oxide and is moderated with heavy water.

Figure 80 is the popular Soviet Union RBMK-1000 power reactor. This design was used for Chernobyl Unit 4, the reactor which exploded in April, 1986. It is characterized by a graphite moderator with light water cooling. Both the Canadian and the Soviet designs allow routine refueling during full power operation. This is a big advantage over U.S. plants which require a lengthy shutdown for refueling.

The radiation fields associated with a nuclear reactor consist of prompt neutrons and gamma rays plus possible exposure resulting from contamination and/or environmental releases of radioactivity. The average annual dose to workers who actually receive a non-zero exposure in U.S. power plants has been tabulated annually by the Nuclear Regulatory Commission since 1973. It has dropped steadily from a high of 0.94 rem/worker-yr in the 1970s to an all time low of 0.1 rem/worker-yr (1.0 mSv/worker-yr) in 2009, the last year for which data was available at this writing.

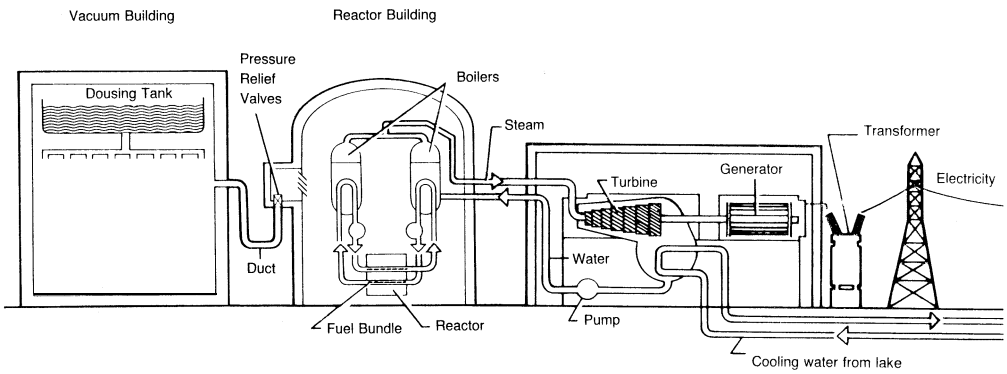


Fig. 79 - The Canadian heavy water reactor

Another way to calculate worker dose is the annual collective dose per reactor, measured in person-rem/reactor-yr. For both BWRs and PWRs in the U.S., this measure of dose has declined since 1983. Figure 81 shows these trends for the two reactor types. The year 2008 average collective dose for both BWRs and PWRs combined corresponds to only about one-eighth of the 1980 value. A job well done!

Federal law used to require the NRC to calculate the dose annually to populations living in the vicinity of all licensed nuclear power stations. The doses were calculated from the actual measured quantities of radioisotopes released into the air and water in the plant vicinity as measured by the plant environmental monitoring network. After 1990, the rules were changed and this complex calculation is no longer

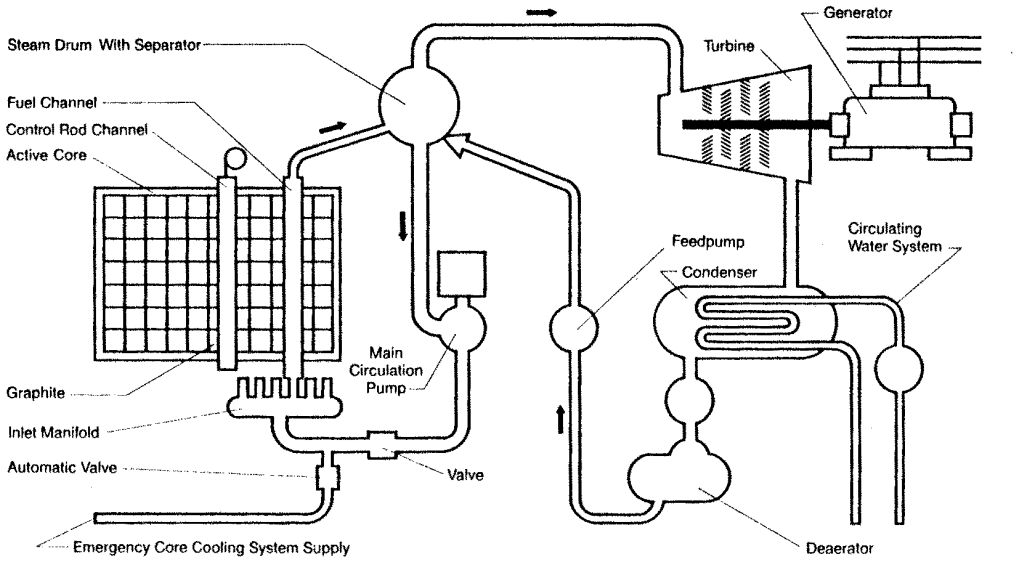


Fig. 80 - The Soviet RBMK-1000 power reactor design

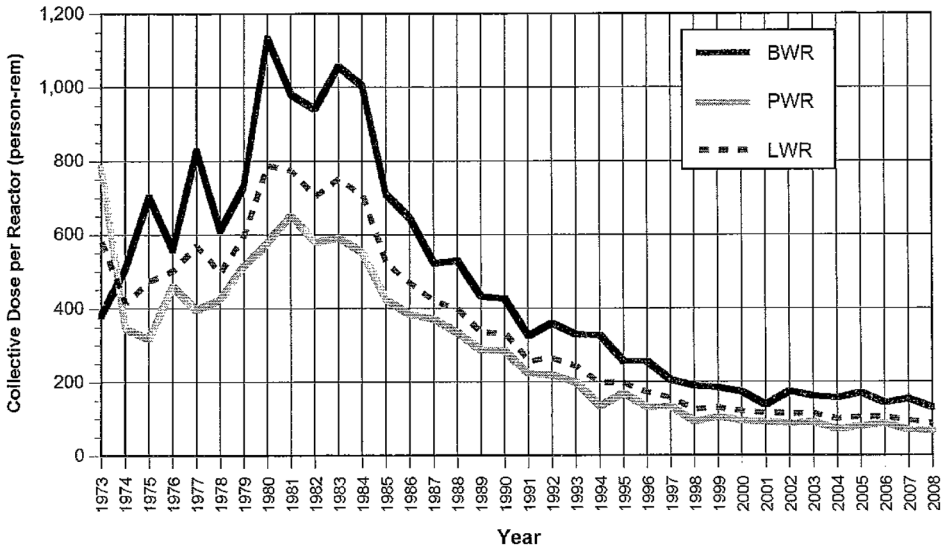


Fig. 81 - Average collective radiation dose at US power reactors

performed. Typical annual doses to persons currently living within a radius of 80 km (50 miles) of a U.S. nuclear power plant site are less than a thousandth of a millirem. This dose is equivalent to about 5 extra minutes of natural background radiation per year or would result from an increase in cosmic ray background due to an average living elevation increase of 2.5 inches (6 cm)!

In support of commercial nuclear power reactors, a number of fuel cycle facilities exist across the United States. Figure 82 lists the major facilities active or under

URANIUM HEXAFLUORIDE FACILITY:

Honeywell Inter. Inc., Metropolis, IL

MIXED OXIDE FUEL FABRICATION:

Shaw AREVA MOX Svcs, Aiken, SC

GAS DIFFUSION U ENRICHMENT:

USEC Inc., Paducah, KY

GAS CENTRIFUGE U ENRICHMENT:

USEC Inc., Piketon, OH

LES-URENCO, Eunice, NM

AREVA Enrichment Svcs, Idaho Falls, ID

URANIUM FUEL FABRICATION:

G N F Americas, Wilmington, NC

AREVA NP Inc., Richland, WA

Westinghouse Elec., Columbia, SC

Nuclear Fuel Services, Erwin, TN

AREVA NP Inc., Lynchburg, VA

B&W N. O. G., Lynchburg, VA

LASER SEPARATION ENRICHMENT:

GE-Hitachi, Wilmington, NC

Fig. 82 - Locations of U.S. fuel cycle support facilities

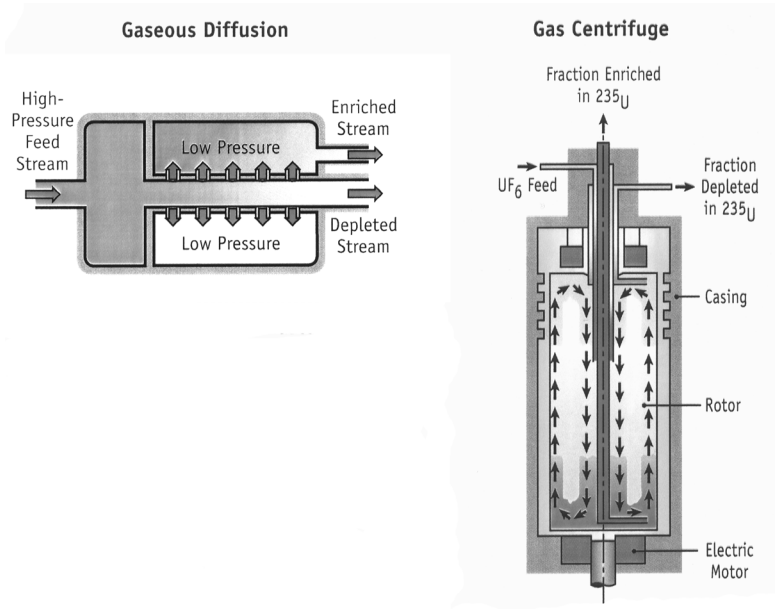


Fig. 83 - The two common methods of enriching uranium

construction as of 2011. The 13 facilities are distributed among 10 different states. These facilities either produce uranium hexafluoride from uranium ore, enrich the UF_6 gas (see Figure 83) or convert the UF_6 gas into fuel pellets and rod assemblies (see Figure 84).

In gaseous diffusion, uranium is enriched by passing the UF_6 gas though a porous barrier which lets the U-235 molecule through slightly more easily than the

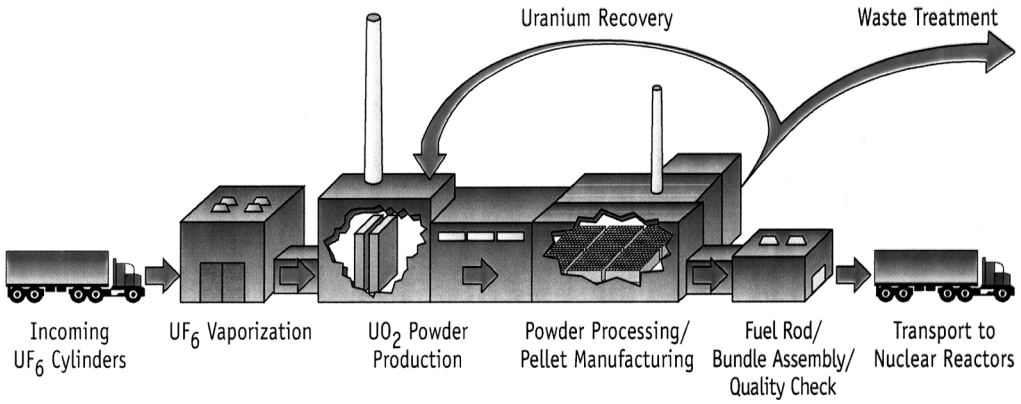


Fig. 84 - Diagram of a fuel fabrication plant

heavier U-238 molecule. Thus, the concentration of U-235 is increased relative to U-238. In a gas centrifuge, the UF_6 gas is rotated in a cylinder at high speed. The heavier U-238 molecules accumulate in higher concentration towards the outer wall of the cylinder leaving an enriched component nearer the cylinder axis where it is extracted. In 2010, AREVA received a $2 billion Department of Energy loan guarantee to expedite construction of their new Eagle Rock centrifuge facility in Idaho. Also in 2010, the USNRC authorized the URENCO facility in New Mexico to start commercial operations producing low enrichment uranium fuel for power reactors.

The GE-Hitachi laser facility uses a high power tunable laser to photoionize the U-235 in naturally enriched UF_6 gas. The U-235 is then chemically removed.

Advanced Power Reactors

Changes to the Licensing Process

A number of years ago, the U.S. nuclear power industry and the U.S. Nuclear Regulatory Commission realized that under current operating procedures, it was unlikely that any new nuclear power plants would be ordered. The construction and licensing delays created a situation where it was not cost-effective any longer. After study, it was concluded that two types of changes needed to be made. The NRC needed to speed up the lengthy licensing process and the nuclear power reactor manufacturers needed to commit to a small number of standard plant designs. As we entered the 21st century, both objectives had been realized!!

The U.S. NRC completed a series of steps which have created a whole new licensing process. By instituting Early Site Permits, Standard Design Certification and Combined Licenses, the entire chain of events of siting, construction and operating permission was greatly streamlined, costs were reduced and public confidence was increased. As of 2010, the US NRC had issued four “pre-approved” design certifications. The Combined Construction and Operating License (COL) process effectively eliminated the second step that provided considerable opportunity for opposition in the earlier licensing process. Now the COL process essentially provides for licensing certainty “up front.” A company can now make a decision whether to go forward with the very expensive construction program knowing that a license to operate the plant is in hand.

Generations of Reactors

If one considers the small early demonstration reactors Generation I then we can identify the current inventory of larger, more standardized reactors as Generation II. Early Gen I units had limited output and in some cases, limited operation. Some examples include Big Rock Point, an early Boiling Water Reactor in Michigan with 67 MW_e output that operated from 1967 to 1992. Humboldt Bay 3 in Northern California was another BWR, 63 MW_e that operated from 1963 to 1976. An early PWR was Shippingport in Pennsylvania that had an output of 60 MW_e and operated from 1957 to 1982. Peach Bottom 1, also in Pennsylvania, was a 40 MW_e High Temperature Gas Cooled Reactor that operated from 1967 to 1974.

Most of these Generation I reactors were not economically competitive with other sources of electricity generation. However, the scaling up to larger size and the

introduction of "turnkey" designs made the concept much more attractive. Generation II reactors began with larger units that utilized the advantages of the new pooled power distribution systems allowing energy to be moved from one location to another more easily. Nuclear plants are suited for this expanded "grid" because they operate best when running continuously at full power. The design of Gen II plants soon rose from 200-500 MW_e units to over 1000 MW_e. In the span of just three years from 1966 to 1968, utilities ordered over 65 nuclear units and estimated that as many as 400 units would eventually be in service.

The Three Mile Island (TMI) accident in March 1979 ended many of those projects. After TMI, opposition to nuclear power became intense and focused and many plants were cancelled or halted during early construction. Licensing became problematic because of the hurdles raised by opponents during the final Operating License phase of the process. Delays in the construction and licensing skyrocketed the capital costs. The result was that many of those units announced in the early 70s were cancelled. It was only in the 1990s that a last few units were completed or put back into service. Watts Bar 1 in Tennessee was the last unit placed in service that had been announced before the TMI accident. The Tennessee Valley Authority has reinitiated licensing for Watts Bar 2 so that it may also eventually be placed in service.

Generation III reactors are the next design evolution and are currently in operation or construction in several countries. U.S. utilities considering the next generation of plants will have several choices for an advanced reactor. There are presently five Generation III+ reactor designs available for utility ordering. Two are light water boiling water reactors designed and marketed by General Electric. The Advanced Boiling Water Reactor (ABWR) is a light water boiling water reactor with an output of 1350 to 1600 MW_e. The NRC has certified the ABWR design. See Figure 85. Four units are already operating in Japan and three more are under construction in Taiwan and Japan. Nine more are planned in Japan. GE Energy also designed and markets the 1520 MW_e Economic Simplified Boiling Water Reactor or ESBWR. The NRC is still

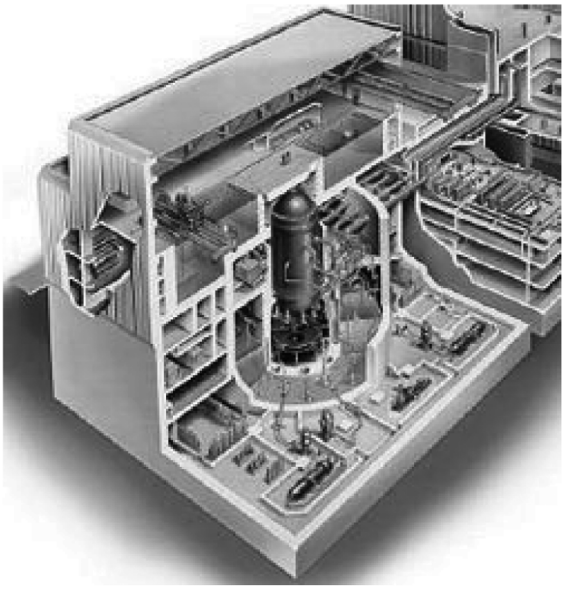


Fig. 85 - The Design Certified GE ABWR advanced reactor

reviewing the ESBWR for certification. The other three designs are all pressurized water reactors or PWRs. The U.S. EPR (Evolutionary Power Reactor) is an AREVA designed 1600 MW_e plant similar to units under construction at Olkiluoto, Finland and at Flamanville, France. Several utilities have expressed interest in construction of new units in the US using this design. It is currently being reviewed by the NRC for certification. Westinghouse offers the AP1000, an 1117 to 1154 MW_e PWR that has been certified by the NRC. See Figure 86. Mitsubishi Heavy Industries of Japan

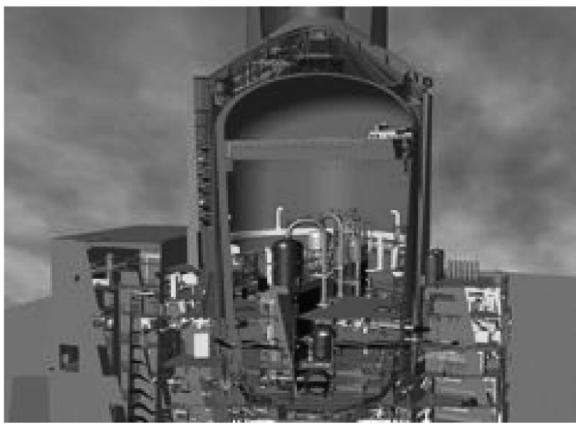


Fig. 86 - The Design Certified AP1000 advanced reactor

designed the US-APWR a large 1700 MW_e PWR also under review by the NRC for certification.

All three of these new designs are considered “advanced” (or Generation III+) in that the designs have been greatly improved over conventional power reactors in terms of accident safety, operations and maintenance. They make use of “passive” safety features which use gravity, natural circulation and convection to replace motor drives and pumps. The calculated core damage frequency, a measure of reactor safety, is 10^{-7} in the ABWR, a factor of 50 to 100 times better than conventional (or Generation II) plants. For example, the ABWR uses an internal reactor pump that reduces the main piping and eliminates the chance of water falling to a level that would expose the core. The ESBWR has 25% fewer pumps, valves, and motors while the AP1000 has 50% fewer valves and 80% less safety-related piping. Many of the newer designs are capable of using mixed oxide (MOX) fuel, a combination of uranium and plutonium that takes advantage of the plutonium production inherent in light water reactors or the downblending of plutonium-based weapons. These plants may also be designed to resist the impact of an aircraft. They typically promote higher thermal efficiencies as well as reduced operations and maintenance costs due to considerably fewer pieces of equipment.

Small Modular Reactors

The newest designs of reactors that are being marketed are small modular reactors (SMR). These have outputs of about 10 to 150 MW_e and are intended to be sold as units that are essentially installed as a single unit. Examples include:

- The Pebble Bed Modular Reactor is a 165 MW_e design that may be built in South Africa. It has an unusual fuel design that gives it the name and operates with helium gas for coolant. The gas coolant allows the unit to operate at considerably higher temperatures which raises the thermal efficiency. The pebble-type fuel design allows online refueling.
- The NuScale reactor will be housed underground. It is a 45 MW_e light water natural circulation reactor with both the reactor core and steam generator in the vessel. The project plan envisions up to twelve SMRs submerged in a pool inside a common underground building.
- Toshiba Corporation has been working with the town of Galena, Alaska promoting an SMR of 10 MW_e output called the 4S for Super-Safe, Small and Simple. The design features a liquid sodium cooled reactor with a 30 year refueling cycle, built primarily underground.
- Hyperion Power is another SMR with a lead-bismuth eutectic coolant. The entire reactor module is a very small unit, 10 MW_e, that is intended to be transported and placed underground as a single unit. When the fuel is exhausted the entire reactor module is to be replaced, approximately every 7 to 10 years.

Two larger designs include the General Electric PRISM or Power Reactor Innovative Small Module, a 311 MW_e liquid sodium system with two reactors for each turbine generator. It will be refueled every year or two. The second larger reactor is the B&W mPower pressurized water reactor. It, also, has the reactor and steam generator inside a single reactor vessel located underground. All of these SMRs are in the late design stage such that they are expected to submit documents for design certification by the end of 2012.

Fourth Generation Fast Reactors

The irradiated fuel from light water reactors contains about 94% uranium-238. If fast reactors are used, this "used" fuel actually becomes a source for plutonium-239 that is fissile and can be used as fuel in other light water reactor systems. This "closing" of the fuel cycle is the ultimate goal of the fission process for energy production and is being pursued by several countries. However, fast reactors and the reprocessing of irradiated fuel is a very expensive proposition that exceeds the capabilities for private ventures and must be addressed on a national or even international scale.

In the United States, two conceptual designs are under study by the Department of Energy. They are the Very High Temperature Reactor (VHTR) and the Sodium-Cooled Fast Reactor (SFR). The VHTR has a low power density while the SFR has a very high power density with liquid metal (sodium) coolant. Both reactors rely on fast neutron spectrums such that they "burn" much lower enriched fuel and greatly eliminate actinides from the waste, compared to today's current fleet of light water reactors.

At present, France, the United Kingdom, and Japan recycle irradiated fuel, extracting the plutonium that can be converted, along with uranium, into mixed oxide fuel (MOX). Nuclear plants in France are using MOX fuel now while the UK is storing the plutonium for eventual use in

recycling applications. The Japanese facility that is just starting up will provide MOX fuel for their light water reactors.

Whatever process is eventually used for future nuclear power plants, there will be some radioactive waste requiring disposal. The resulting volume and time of required isolation will depend on the type and effectiveness of the recycling process that is selected.

Department of Energy Weapons Production

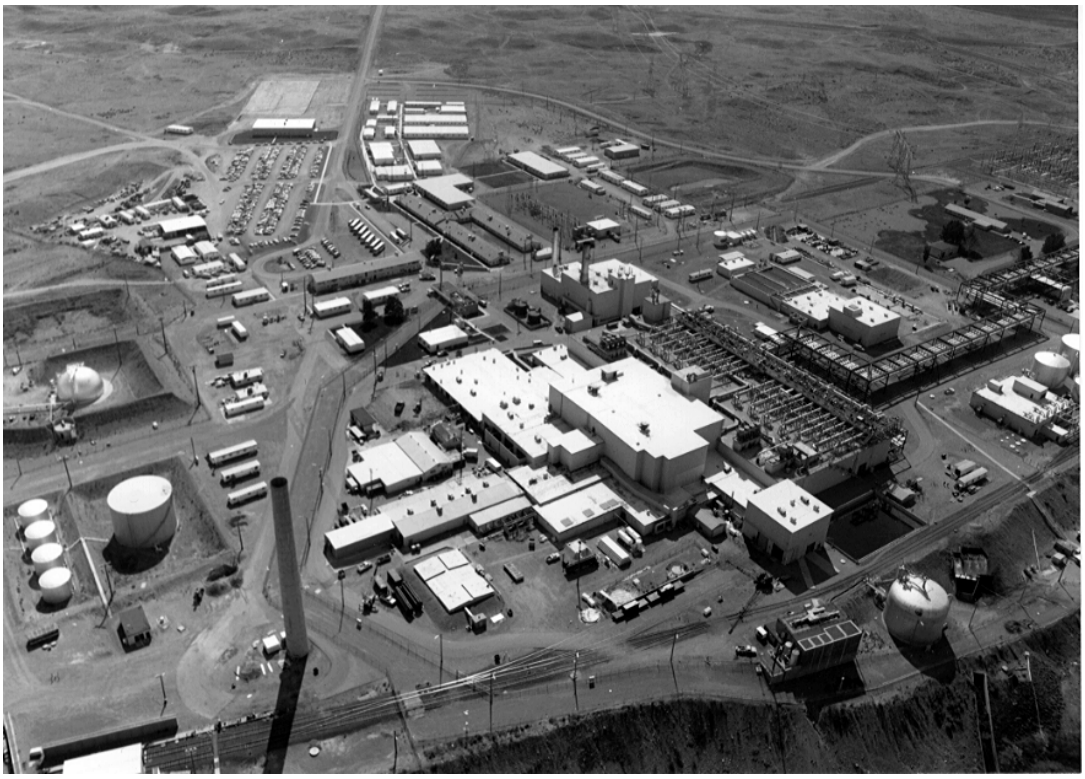
Moving away from the commercial and research sectors, there is still one remaining use of nuclear reactors - nuclear weapons production. In the U.S., this responsibility is given to the Department of Energy or DOE. The actual production is spread across the states in some 15 major facilities. Weapons grade plutonium is produced by neutron bombardment of U-238 in heavy water or graphite moderated, water-cooled reactors fueled with highly enriched uranium. Power levels are typically a few thousand MW thermal. Tritium, used to boost the power of virtually all U.S. warheads, is similarly produced except that the target material is lithium. Following fuel reprocessing, plutonium is extracted and machined into parts and assembled together with the detonators, high explosives and electronic triggers.

The plutonium production reactors were located on both coasts. Nine reactors operated at one time or another at the Hanford facility, Richland, WA. In addition, both tritium and plutonium could be produced in three reactors at the Savannah River Site in Aiken, SC. All of these weapons production reactors operated at lower pressures and temperatures than commercial power reactors. They did not have the containments characteristic of their commercial cousins.

At the present time (2010), all Hanford reactors have been shut down. The most recently operated "N-Reactor" (Figure 87) was the focus of intense review following the Chernobyl accident. The N-Reactor was built during the 60s and was expected to run for a 20 year life span. It was graphite moderated like the RBMK-1000 type at Chernobyl. As a result of the investigation into various safety problems and design features, the DOE temporarily shut down the N-Reactor in 1987. In 1988, the decision was made to make the shutdown permanent.

Around 1985, under the Freedom of Information Act, environmentalists were able to obtain records of radioiodine releases from the Hanford reactors. Apparently about 530,000 curies of I-131 were released into the air between 1944 and 1956. This is an enormous quantity. The 1957 Windscale reactor core fire, which will be discussed in Chapter 14, released 20,000 Ci. The Hanford releases were exceeded only by Chernobyl (about 7 million Ci of I-131).

Some 270,000 persons in ten downwind counties were exposed to these emissions. The Hanford Environmental Dose Reconstruction Project (HEDR) released its draft report in 1990. They estimate that 50% of this exposed population received thyroid doses over 16 milligrays and 5% were above 330 mGy (33 rad). The report also determined that 11 infants could have received thyroid doses over 25.3 Sv (2,530 rems)! It should be kept in mind that these radioiodine releases were made legally under regulations in force at that time. All the reactors were being operated under wartime



Courtesy Battelle Pacific Northwest Laboratories

Fig. 87 - The N-Reactor at Hanford in Washington State

conditions and under a "secret" security classification so the iodine releases were not in the public domain at that time.

The three Savannah River reactors, built in the 50s, had heavy water moderators. The emergency core cooling system used river water. The fuel was aluminum clad rather than the zirconium usual in the commercial reactor sector. The aluminum cladding melts at lower temperature and is potentially susceptible to hydrogen gas releases in a loss of coolant accident. The National Academy of Sciences set up a panel of experts to review the safety of the Savannah reactors. They issued recommendations in 1987 criticizing DOE on several major safety items. In 1988, all three reactors were shut down for safety modifications. Over $2 billion has been spent to add an additional emergency core cooling system, upgrade control rod housings, improve seismic safety and add backup diesel generators to a cooling system used if river cooling is prevented. In 1991, DOE announced that the K and L-Reactors would resume operations but the P-Reactor would be permanently retired (Figure 88 shows the K-Reactor). As of 2010, all three reactors have been permanently shut down.

This leaves the U.S. without a tritium production facility to replace the gas which is lost from our nuclear weapons through radioactive decay (12.3 year half-life). One solution being discussed is to build a nuclear particle accelerator at Savannah River specifically to make tritium. An alternative plan is currently being implemented. This involved modifying one of the commercial nuclear power reactors from the Tennessee Valley

Fig. 88 - The K-Reactor site at Savannah River in South Carolina

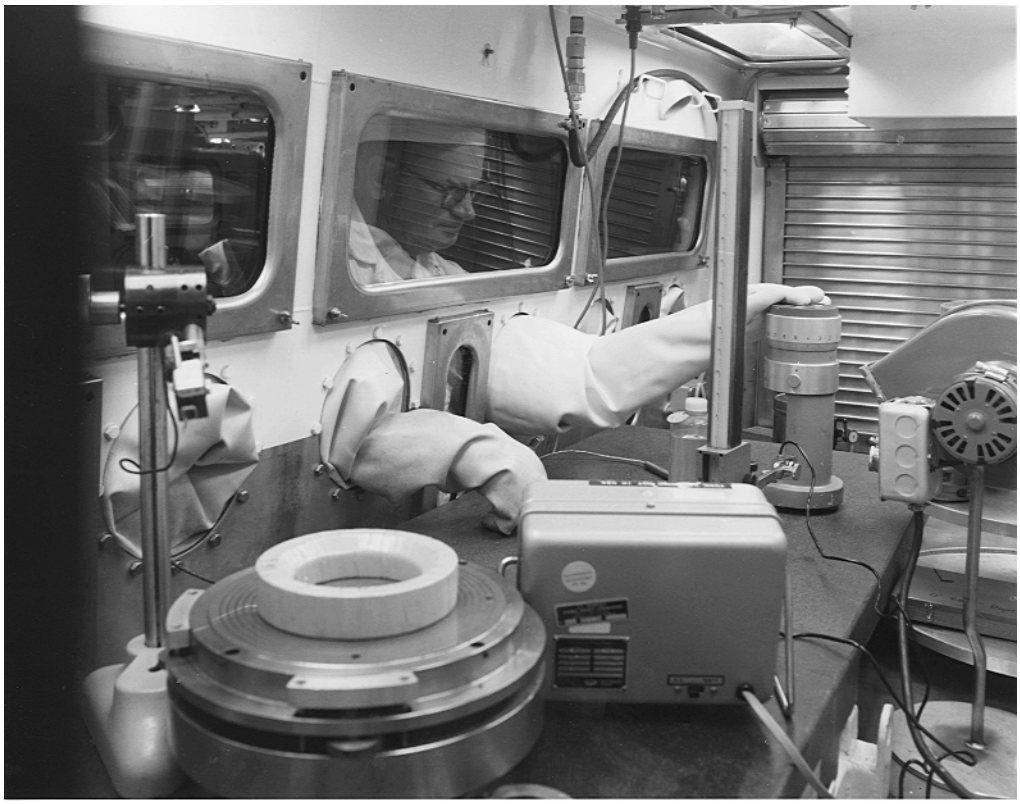
Authority to add a tritium production capability. Components were added to the TVA plant to allow exposure of lithium-6, a stable isotope, to the neutron field near the reactor core. Then, neutron capture leads to tritium from the reaction:

$$^{6}Li + {}^{1}n \rightarrow {}^{4}He + {}^{3}H.$$

After irradiation, the target material is returned to the Savannah River Site. A newly built tritium extraction facility on the site has been put in operation recently to recover the H-3.

With the shift away from weapons materials production, Savannah River was left without a mission. At present, the laboratory is gearing up to start converting weapons grade plutonium removed from U.S. warheads to mixed oxide reactor fuel, MOX fuel. This fuel can then be transferred to a commercial nuclear power reactor which has been upgraded to accept this MOX alternative. In addition, Savannah River is setting up a program on their site to remove plutonium pits from decommissioned nuclear warheads to cycle into the MOX fuel program. As of 2011, the project is progressing. The MOX facility is under construction. Unfortunately, the plutonium pit operation is still only in the planning phase. However, the shut-down K-Reactor site has been incorporated into the project. It has been converted into a facility for interim storage of the plutonium removed from the decommissioned nuclear weapons.

Plutonium metal components for weapons used to be machined at Rocky Flats in Golden, CO. Over the years, the plant was plagued by a number of glove box fires, caused by the pyrophoric nature of plutonium (Figures 89 & 90). In 1988, the facility was temporarily closed following exposure of a DOE inspector and two employees to small doses of plutonium when they entered an area where the radiation warning sign was



Courtesy EG&G Rocky Flats

Fig. 89 - Rocky Flats glove box in use

Courtesy EG&G Rocky Flats

Fig. 90 - Rocky Flats master-slave manipulator and glove boxes

blocked by an electrical panel. During 1989-90, the facility was in a standby mode. Then in January 1992, President Bush announced a scaling back of nuclear weapons production and cancelled the W-88 warhead, a major Rocky Flats project. Plutonium manufacturing operations were shut down and Rocky Flats redefined their mission to one of decommissioning, waste cleanup and environmental restoration.

Some of the remaining major facilities in the overall DOE weapons complex include the following:

- **Y-12 plant in Oak Ridge, TN which processes uranium**
- **Mound Laboratories in Miamisburg, OH which processes tritium**
- **Feed Materials Production Center in Fernald, OH**
- **Idaho National Engineering Lab, (INEL), in Idaho Falls, ID**
- **Pantex Plant, Amarillo, TX.**

The Fernald Ohio plant reprocessed uranium and made reactor fuel rods. INEL reprocesses spent reactor fuel and stores radioactive waste for DOE facilities pending transfer to the Waste Isolation Pilot Project in Carlsbad, NM. The final assembly of all nuclear weapons takes place in the Pantex Plant, Amarillo, TX. Pantex is also responsible for disassembly of outdated and retired nuclear weapons in order to reclaim and recycle usable materials.

Miscellaneous Sources

Occupational Exposures

Doses received by radiation workers, in contrast to the general population, are individually measured with radiation badges. Periodically, national summaries of badge data become available. The most recent, at the time of this sixth edition, was

Occupational Category	% of Radiation Workers	Ave. mSv/year
Medicine	60%	0.8
Industry	11%	0.8
DOE, DOD, Gov't.	3%	0.6
Nuclear Power	5%	1.9
Aviation	14%	3.1
Other	7%	0.7
ALL WORKERS	100%	1.1

Fig. 91 - Average doses to actually exposed U.S. radiation workers - 2006

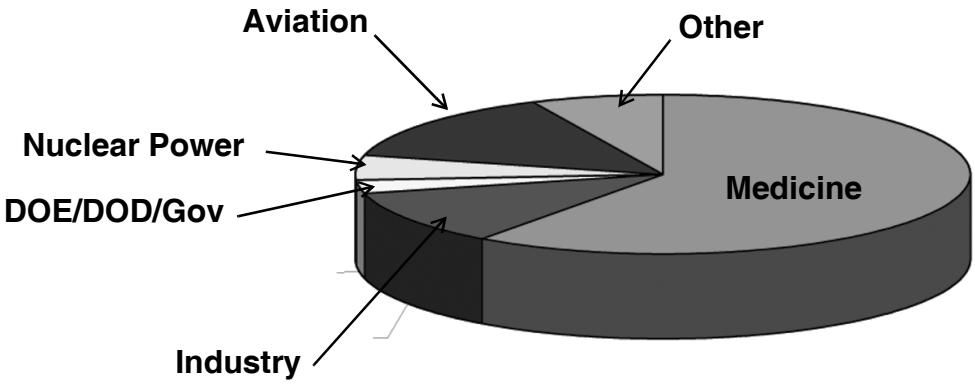


Fig. 92 - Distribution by category of U.S. radiation workers - 2006 data

NCRP Report 160. Occupational data was reported for the year 2006. A table and pie chart summarizing the effective dose results and occupational distribution for U.S. radiation workers are shown in Figures 91 and 92.

The data are divided into six occupational categories. Note that this summary includes data only for workers who received a measurable dose on the badge. Since 1985, the average effective dose per exposed worker fell 58%! The number of radiation workers employed in the medical field in the U.S. exceeds all other radiation worker categories combined.

Isotopic Neutron Sources

A number of disciplines make use of sealed neutron sources. Such sources come in two common varieties - spontaneous fissioning radionuclides and alpha emitting radionuclides + a target. Californium-252 is an example of the former while a Pu-Be source exemplifies the second type.

Cf-252 has a number of useful properties when used in this application. The half-life is 2.6 years, in sharp contrast to most spontaneous fission radionuclides which have half-lives less than a minute. It emits 3.7 neutrons per fission, with an average energy of 2.3 MeV. A Ci of ^{252}Cf emits 4.4 x 10^9 neutrons per second. The specific activity is very high – one gram of Cf-252 equals 20 TBq (537 Ci). A one-milligram source produces, at one meter distance, a neutron dose equivalent rate of 23.3 mSv/hr and a gamma dose equivalent rate of 1.6 mSv/hr.

Sources in the second category are manufactured by intimately mixing an alpha emitter with an appropriate target material to produce fast neutrons by (alpha, n) nuclear reactions. The three commonly used target materials are listed in Figure 93 along with the neutron yields. It is readily apparent why beryllium is so popular - the neutron yield is so high. Commonly used alpha emitters are listed in Figure 94.

In commercially available sources, neutron yields are typically around 10^7 n/sec-Ci for a Ra:Be source and 2 X 10^6 n/sec-Ci for Po:Be, Ac:Be and Pu:Be sources. Unfortunately, most alpha emitters do not decay exclusively by alpha emission. All of the useful alpha emitters listed have

Element	Neutrons per 10^6 Alphas
Lithium	2.7
Beryllium	77
Boron	22

Fig. 93 - Targets for isotopic neutron sources

associated gamma and x-rays that are also emitted by the finished source. The Ra:Be source has the highest ratio of gamma to neutron output (0.6mSv/hr @ 1 m per 10^6 n/sec) while the Po:Be source has the lowest (only 0.0004 mSv/hr @ 1 m per 10^6 n/sec). Changing to a lithium target reduces the average neutron energy significantly. Appendix A-3, Data for Neutron Instrument Calibrations, gives further information.

Besides some medical uses in radiation oncology, isotopic neutron sources have several industrial applications. They are used as start-up sources in a nuclear reactor. Figure 95 shows a neutron howitzer, a popular research tool at colleges and universities. The isotopic neutron source is visible just below the center in the plastic water-filled tank (howitzer). Objects to be irradiated can be lowered into the tank or inserted in drawers in the horizontal plastic tubes.

The moisture gauge is another industrial application. As discussed in Chapter 3, fast neutrons are scattered most efficiently by a particle having the same mass as a neutron. Since a water molecule contains two hydrogen nuclei (protons) with virtually the same mass as a neutron, water will readily scatter fast neutrons. The moisture gauge uses a collimated neutron beam from an isotopic source and also incorporates a neutron detector. The detector measures the fraction of the neutrons which are scattered back. Placing the gauge on soil allows the water content to be determined.

A similar application is a roof moisture gauge. This device is used industrially to map the moisture content of large roofs to isolate and

Radioisotope:	Alpha Energies: (MeV)	Half-Life:
Ra-226	4.7 to 7.7	1620 years
Pb-210	5.3	22 years
Po-210	5.3	138 days
Ac-227	4.9 to 7.4	22 years
Pu-238	5.3 to 5.5	88 years
Pu-239	5.1	24131 years
Am-241	5.3 to 5.5	432 years

Fig. 94 - Alpha emitters commonly used for neutron sources

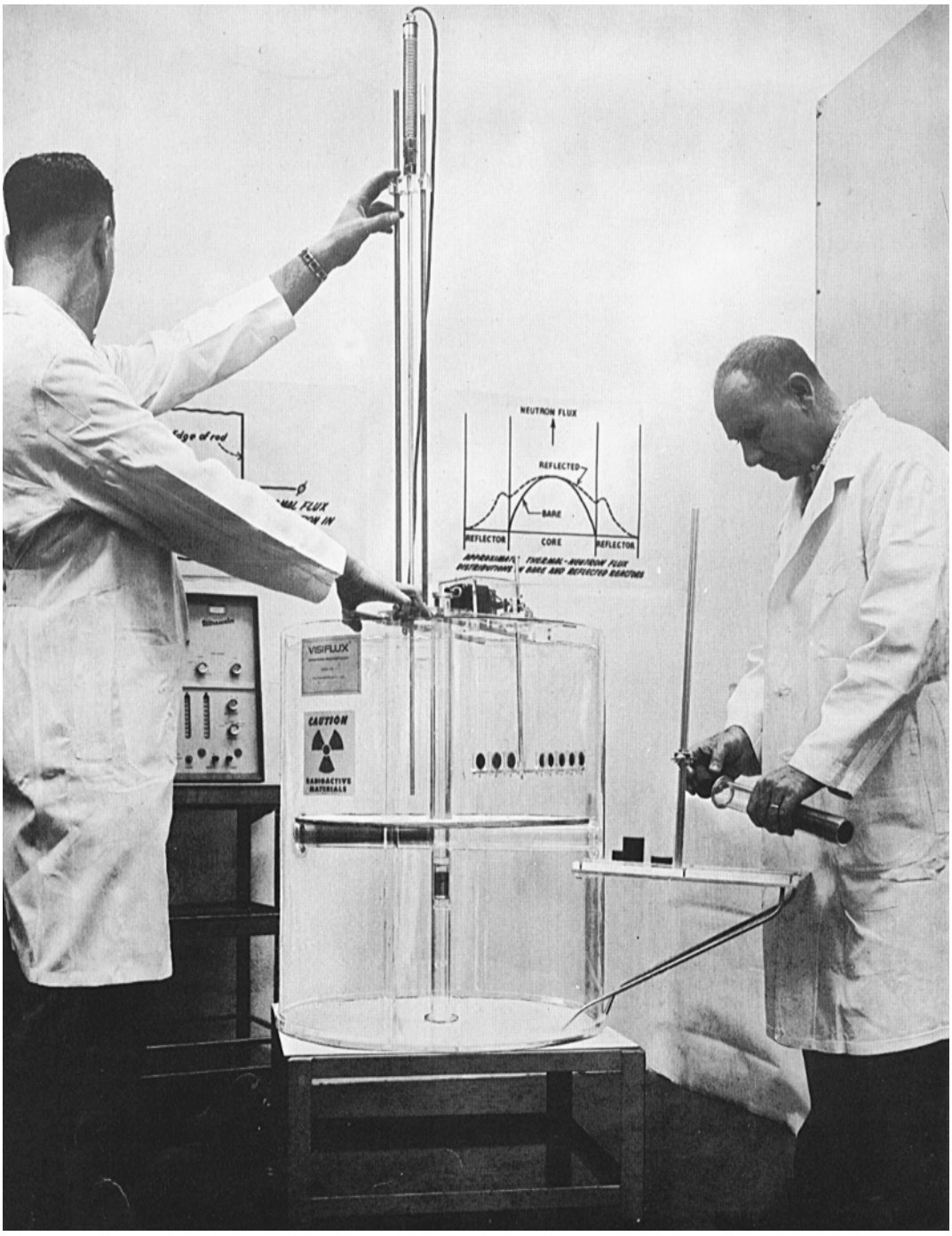


Fig. 95 - A neutron howitzer in use

Fig. 96 - A moisture gauge using an alpha-beryllium neutron source

repair water leaks. Such an instrument is illustrated in Figure 96. This gauge uses a 40 mCi Am-241:Be neutron source with a helium-3 thermal neutron detector. In this case, the roof moisture thermalizes the fast neutrons so the higher the reading, the greater the moisture level at that location.

Oil Well Logging

A final, closely related example is oil well logging. Companies engaged in oil exploration make use of probes lowered down a drill hole. There are three common techniques which have made use of this idea for decades. "Gamma logging" makes use of the natural radioactivity in the rocks which line the borehole. A sensitive gamma ray detector is slowly lowered to record the radiation level as a function of depth in the hole. "Gamma gamma logging" is a second method in which ^{60}Co or ^{137}Cs sources of a few mCi to about 2 Ci are lowered along with a gamma ray detector. The backscattered gamma ray level is recorded for analysis at various depths. "Neutron logging" is the third common procedure. Isotopic sealed ^{241}Am-Be neutron sources are attached to the probe which also contains a thermal neutron detector. The higher the petroleum content in the strata, the higher the reading of thermalized neutrons from the hydrogen in the oil. A graph of readings versus depth in the drill hole is used to evaluate the borehole. Source size is typically a few Ci of Am-241.

More recently, some additional logging methods have become available with advances in technology. Multichannel Pulse Height Analyzers (discussed in Chapter 7) allow measurement of underground gamma ray energies to identify radionuclides. Also, highly miniaturized nuclear particle accelerators can be lowered down the hole to produce high energy neutrons to induce gamma ray emissions and identify underground elements.

A second way of evaluating a borehole during oil exploration is to inject radioactive material into the hole, sometimes under high pressure. This radioactive tracer is especially useful for locating perforations in the well casing. The commonly used tracer radioisotopes include ^{192}Ir "frac sand," ^{192}Ir paint, and liquid solutions of ^{3}H (several curies per well), ^{57}Co, ^{60}Co and ^{131}I (tens of mCi to a few hundred mCi typically). On occasion, the high pressure ruptures the injection apparatus, causing dispersal of the radioactive material and gross contamination of the wellhead area.

The health physics problems associated with well logging or tracer procedures can be formidable. Field operations are usually in remote locations, far from the prying eyes of the company health physicist or regulatory personnel. Sources are transported to the wellhead in a shielded container chained to a truck. Transfer of sealed sources to the logging tool usually involves exposure to the operator for 3 to 5 seconds. Statistically, it has been determined, in a 1980 Texas study, that one source becomes stuck underground for each 10,000 borehole operations. Retrieval is fraught with headaches. The "long" handled tools used routinely in radiation protection technology don't measure up to a fishing task at 6,000 feet underground! If the source is lodged below the productive strata, a common practice is to pour cement down the borehole to fix the source permanently in place. Estimates put the number of cemented sources in the ground at around 1,000. Analysis of Canadian well logging dosimetry records for 925 loggers showed an average annual dose of 205 mrem per person in 1980.

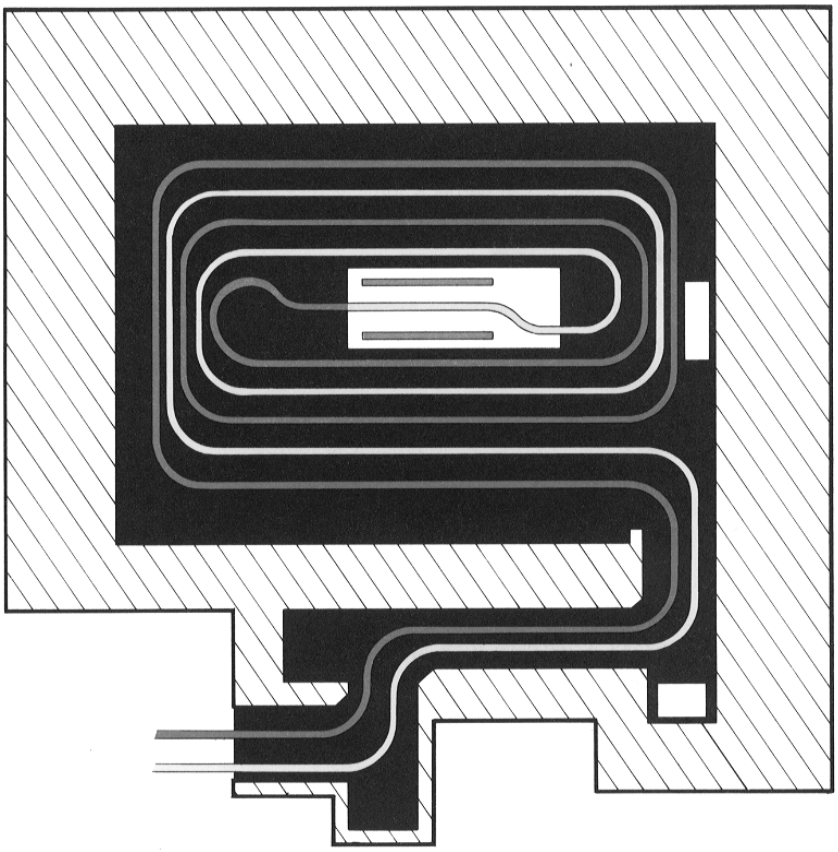
In the United States, national regulations governing the health physics aspects of well logging are found in Title 10, Code of Federal Regulations, Part 39 - Licenses and Radiation Safety Requirements for Well Logging. Subpart D (Radiation Safety Requirements) is particularly relevant for radiation protection technicians. It covers training, operating and emergency procedures, personnel monitoring, surveys and contamination control.

Radiation Sterilization

In 1986, the U.S. Food and Drug Administration approved the gamma irradiation of fruits and vegetables up to a dose limit of 1000 gray. In the mid 80s, dried spices, herbs and pork were approved. The first foods accepted for gamma irradiation were wheat and potatoes back in the 1960s. The objective is to reduce the need for pesticides and preservatives. Radiation kills trichina worms in pork and salmonella bacteria which produces food poisoning in humans. Large sealed sources of ^{137}Cs and ^{60}Co are typically used. There were 50 large gamma irradiators, as of 2010, in the U.S. and Puerto Rico.

In addition to food, gamma sterilization is also used on medical supplies, disposable diapers, milk cartons and a variety of other products. One of the two remaining alternatives, ethylene oxide gas sterilization, has decreased in usage since 1973. Originally, gas sterilization was the most economical method for non-food products as the gas is cheap and chambers could be added to expand capacity. However, it was eventually shown that ethylene oxide reacts with moisture when in the presence of

chloride ions to form a toxic product. Federal regulations have controlled worker exposure since 1984 and several states have limited atmospheric releases. In 1986, radiation sterilization was used for 40% of the sterilized medical supplies produced in this country. Figure 97 shows a schematic drawing of the world's largest facility operated by Radiation Sterilizers in Tustin, California. It utilizes ^{60}Co sources with an activity of around 6 million curies (222,000 TBq)! The shielding consists of 8 foot thick concrete walls with a 6 foot thick concrete ceiling. Figures 98 and 99 show the source pool which shields the sources when they are not raised up for irradiations and some of the mechanical components for moving product carriers through the facility. The typical sterilizer has an inventory of from 1.5 to 2.5 MCi. The typical radiation dose delivered to non-food products is 2.5 Mrad (25,000 Gy).



Courtesy Radiation Sterilizers, Tustin, CA

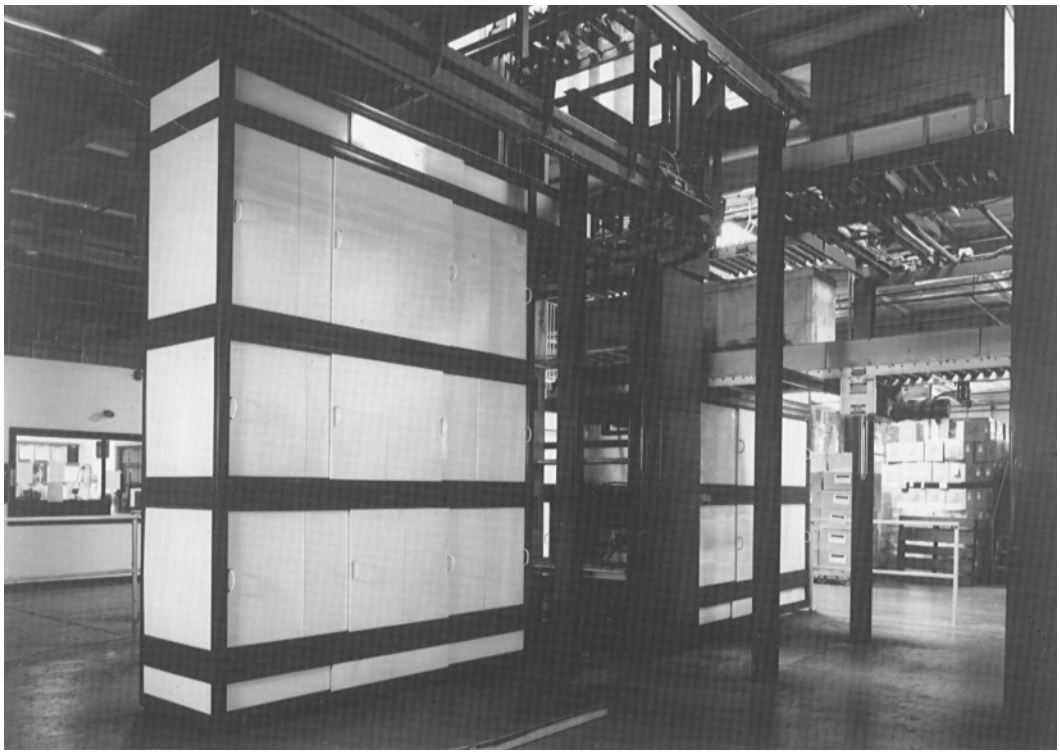
Fig. 97 - Floor plan of a radiation sterilizer facility

Conclusions

In the beginning of this Chapter, the various sources that expose the U.S. population were identified. Later in the Chapter, a case was made for the usefulness of the concept of GSD when discussing population exposures. The 1987 NCRP Report 93 estimated that the U.S. Average GSD from all radiation sources combined was 130

Courtesy Radiation Sterilizers, Tustin, CA

Fig. 98 - Source pool lit by Cerenkov radiation from intense gamma sources



Courtesy Radiation Sterilizers, Tustin, CA

Fig. 99 - Sterilizer handling equipment

Source	Contribution in mrem/year
Natural Background	
Cosmic Radiation	28
Terrestrial Sources	28
Internal Nuclides	36
Occupational	0.6
Nuclear Fuel Cycle	<0.05
Consumer Products	5
Medical	
Diagnostic x-rays	20-30
Nuclear Medicine	2
TOTAL	≈130

Reprinted by permission of the NCRP

Fig. 100 - Ave. GSD background radiation in the U.S. from all sources

mrem/year. The breakdown of that total, by category, is shown in Figure 100. More recent U.S. GSD data was not available due to the policy change mentioned earlier on the part of the NCRP.

This concludes the first Unit on Radiation Protection Theory. The next Chapter begins Unit 2 - Radiation Protection Instrumentation.

Problem Set

1. Name the three components of natural background radiation. How do they compare in size?

2. Why does the external terrestrial radiation received by a person depend on the part of the country they live in?

3. Define the terms “Genetically Significant Dose” and “U.S. Average Genetically Significant Dose.” What information is needed to compute each?

4. Name some isotopes that contribute to the natural internal dose of persons in the USA. What long-lived natural radioisotope contributes the largest internal dose to members of the general public?

5. Why is the composition of cosmic rays so different at sea level and at 50 km altitude? Why does the sea level dose rate vary with position above the equator?

6. What are HZE particles and why are they a hazard to space travelers?

7. With a half-life less than 4 days and a gaseous nature, why is radon-222 such a problem in the oil and gas production industry?

8. Which artificial source contributes the most to the U.S. average GSD? How does its magnitude compare to the various natural components of background radiation?

9. Respond to the statement, "Radiation from consumer products is dangerous."

10. How does the estimated annual dose to a person living near a nuclear power plant compare with a person living near a coal burning power plant? How do both doses compare to that received by cooking with a natural gas range?

11. Describe how an x-ray tube produces x-rays. What is the chief engineering design problem with the tubes? What are some solutions to this problem?

12. Calculate the change in dose rate if a copper target is substituted for the tungsten target of a 100 kVp medical x-ray tube.

13. Briefly discuss the radiation safety problems encountered in industrial radiography (NDE).

14. What is the difference in the medical x-ray procedures of general radiography vs. fluoroscopy? How do the doses to the patient compare in the two procedures?

15. What is a CAT Scanner (Hint: It is not used in veterinary medicine)? What type of radiation field does such a machine produce?

16. About how much medical radiation is received by the average U.S. citizen each year?

17. Describe a medical linac. What is it used for? What dose rate would be expected in the useful beam?

18. What is the chief hazard to operating personnel of an analytical x-ray machine? How might the accident risk be reduced?

19. Why are radioisotope cows so popular in hospitals? Estimate the exposure rate at 1 meter away from a freshly arrived UNSHIELDED Tc-99m generator. State your assumptions.

20. Name some radioisotope sources that might be commonly encountered in a radiation oncology department. What is each used for?

21. Calculate the energy that could be given to a bare boron nucleus if it were accelerated in a linear particle accelerator with a potential difference of 1.7 MV.

22. What is the chief design difference between a betatron and a cyclotron?

23. Why is the "prompt" radiation field not of concern to a radiation protection technologist assisting in maintenance inside a high energy electron accelerator vault?

24. What is the difference between the prompt fields of a medium energy electron accelerator compared to a proton accelerator of the same energy?

25. How many U-235 atoms must fission to produce a kW-hr of ELECTRIC energy from a PWR?

26. What is k_{eff}? How is it used? What effect does a reactor control rod have on it?

27. Name the various components of a nuclear reactor, the function of each and typical materials that might be used.

28. What are some uses for research reactors?

29. Describe the major design differences between a PWR and a BWR. What are the advantages of each?

30. Which occupational category of radiation workers receives the highest annual average doses? Which occupational category has the most exposed workers?

S-1. Why is beryllium often used as a target in isotopic neutron sources?

S-2. What are the chief design differences between DOE weapons production reactors and U.S. commercial power reactors?

S-3. Describe the process and purpose of oil well logging.

S-4. What is the most popular use of radiation sterilization?

S-5. Name three safety improvements used in the new "advanced" power reactors designed in the U.S.

S-6. How do the proposed Generation IV reactors differ from the Generation III units?

Other Resources

1. "Ionizing Radiation Exposure of the Population of the United States," NCRP Report 93, National Council on Radiation Protection and Measurements, Bethesda, MD, 1987.

2. "Ionizing Radiation Exposure of the Population of the United States," NCRP Report 160, National Council on Radiation Protection and Measurements, Bethesda, MD, 2009.

3. "U.S. NRC Information Digest," NUREG 1350, Nuclear Regulatory Commission, Washington, DC. (Latest volume available free at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1350/). This valuable report summarizes annual data on NRC licensee activities on reactors, rad materials, rad waste and U.S. & worldwide nuclear energy.

4. "Bulletin on Management of Naturally Occurring Radioactive Materials (NORM) in Oil & Gas Production," API Bulletin E2, American Petroleum Institute, Washington, DC, 1992.

5. "Radiation Protection for Particle Accelerator Facilities," Report 144, National Council on Radiation Protection and Measurements, Bethesda, MD, 2003.

6. "Operational Radiation Safety Program for Astronauts in Low Earth Orbit: A Basic Framework," National Council on Radiation Protection and Measurements, Bethesda, MD, 2002.

7. Updates on the status of the Standard Design Certification process for advanced power reactors can be found on the U.S. NRC website http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/next-gen-reactors.html.

Unit 2

Radiation Protection INSTRUMENTATION


M.J Kelly

7

Radiation Detectors

Outline of this Chapter

Chapter Summary

This chapter begins a new unit - Radiation Protection Instrumentation. It should be noted that this chapter deals only with the theory and operation of the actual radiation detectors themselves. The applications and use of complete instruments for radiation protection surveys are covered later, in Chapter 12. After a brief review of some of the many mechanisms which are potentially useful as a radiation detector, this chapter discusses practical detectors using each of the three states of matter – gases, liquids and solids.

The characteristic curve of a gas-filled detector illustrates all of the common commercially available gas-based detectors. The lower section of the curve is the operating region of the ion chamber. This member of the gas-filled family is noted for its energy independence, very low current level signal and large anode to prevent gas multiplication. The middle region of the characteristic curve is home to the proportional counter. This device requires gas multiplication, which is encouraged by a tiny diameter anode, and puts out voltage pulses proportional to the energy deposited for each ionizing event. The upper section of the characteristic curve is used for Geiger counter operation. Here, a large voltage pulse, with no energy information, is produced, which can be processed by rather simple electronic circuits.

Liquid scintillation counters use photomultiplier tubes to observe light flashes from vials containing the liquid sample, solutes and scintillating chemicals in solution. The intimate mixture of radioactivity with the scintillation chemical molecules means high efficiency, even for low energy beta sources where this detector is unchallenged. Superheated drops in liquid or gel suspensions are becoming practical as neutron dosimeters.

Solid scintillation counters are useful for gamma ray spectroscopy. NaI(Tl) is the most common, though the new lanthanum bromide counters are becoming increasingly popular. Although NaI(Tl) is hygroscopic, it has efficient light output and high density. Semiconductor counters are an alternative to scintillators, particularly where superior energy resolution is needed. HPGe detectors can be built with the same counting efficiency as 3” x 3” NaI(Tl) scintillation counters, but with much higher energy resolution ability. The chief disadvantage is the requirement for operation at liquid nitrogen temperature. An emerging semiconductor counter, CZT, has good energy resolution and operates at room temperature.

Detection Mechanisms

There are a variety of changes which can be caused by ionizing radiation interacting in a variety of materials. Each combination is potentially useful as a radiation detector. While this chapter will focus primarily on gas-filled and solid state detectors making use of the ionization or scintillation principle, the technologist should be aware of the existence of some other types of detection mechanisms. A short list, all of which have at least one "practical" application, is given in Figure 1.

1. Ionization – Release of ion pairs by radiation
2. Biological – Changes produced in a living system
3. Chemical – Release of free radicals in a solution
4. Heat – Temperature rise from deposited energy
5. Scintillation – Light flash in a special phosphor
6. Thermoluminescence – Light release on heating a phosphor
7. Superheated Drop – Bubble formation in a gel matrix
8. Radiochromic Dye - Color change after irradiation

Fig. 1 - Some radiation detection mechanisms

Ionization is probably the most commonly used phenomenon in the design of practical detectors for radiation protection applications. Instruments can be constructed to read both the exposure (or dose) and the exposure rate (or dose rate). Because the W value is so small (33.9 eV/ion pair in air) ionization based detectors are capable of high sensitivity, that is, they will read at very low exposure rates. In some detectors based on the ionization mechanism, single individual ionizing events can be distinguished. Biological mechanisms used for radiation detection are quite insensitive. Also, there is a significant time lag between the radiation exposure and the corresponding change. Several practical biodosimetry methods were discussed in Chapter 4.

As with biological changes, chemical changes following irradiation are also relatively insensitive to dose. A practical chemical detector would be the Fricke dosimeter in which a ferrous sulphate solution is oxidized to ferric sulphate by the free radicals released during the irradiation of the solution. This dosimeter is useful in the range of about 10 Gy to about 500 gray (50,000 rads).

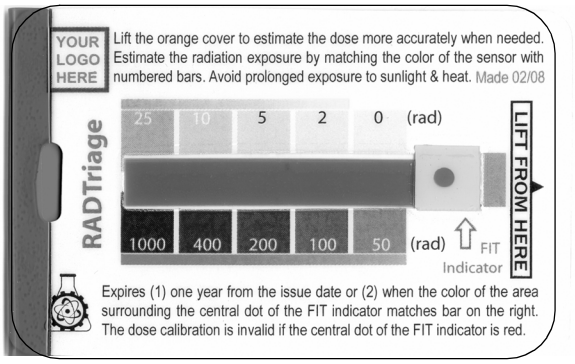
Another "practical" example is the antique U.S. Army Tactical Dosimeter illustrated in Figure 2. This unit contains tubes of chlorine solution in 5 different concentrations. In addition, each tube contains an "acid/base indicator" which changes color from purple to orange if the solution pH changes to acidic. The exposure to radiation of the dosimeter releases hydrogen free radicals into the solution where they combine with the Cl to make hydrochloric acid, HCl. This changes the pH (and hence, the color of the tube) when a sufficiently high dose has been delivered. The required dose depends on the Cl concentration in the tube. Note the dose range covered – 50 to 450 ROENTGENS. Obviously, this is not a device that would be suitable for routine personnel dosimetry in the mR range.

Fig. 2 - A chemical personnel dosimeter

Energy deposited in an absorber (i.e., dose) raises its temperature. This fact can be used to build a dosimeter. Through use of a laboratory "calorimeter," very small temperature changes can be determined. Since a temperature rise of the order of 10^{-5} degrees Celsius per rad is common, this system is useful for measuring doses in the kilogray to megagray range. The details of the scintillation mechanism will be explored further in this chapter. Thermoluminescence is the topic of Chapter 8. A recent development in neutron dosimetry makes use of the detection of bubbles formed from superheated drops in a gel. This topic is also covered later in the chapter.

Finally, color changing dyes are the basis of a new, emerging detector. The SIRAD, for Self-indicating Instant Radiation Alert Dosimeter, measures ionizing radiation over the range from 2 to 1,000 rads (0.02 to 10 Gy). The radiochromic dye is diacetylene which turns blue as it transforms to polydiacetylene during irradiation. The inventor, Dr. Gordhan Patel, introduced the commercially available SIRAD around 2003. A dye strip and reference color blocks are combined into a thin, low cost disposable dosimeter the size of a credit card or, alternatively, a postage stamp. Figure 3 illustrates both configurations. The dose is determined by comparing the color of the central strip with the color blocks on either side. The color change occurs within one minute so the information is available immediately. Dose uncertainty is about ±20%. No processing by a badge company is needed! Each SIRAD has a useful life of about 1 year at room temperature. The life can be extended by cold storage.

These dosimeters are particularly designed for the emergency first responder audience. More information is listed in the "Other Resources" section for this Chapter.



Wallet Card

Sticker

Courtesy, JP Labs, Inc.

Fig. 3 - SIRAD radiochromic personal dosimeter

One general point should be discussed before proceeding to specific detectors. Seldom will a technician encounter a detector that responds to every ray striking it, i.e., has 100% counting efficiency. Instead, only a fraction of rays interacting with the sensitive volume of the detector will actually result in a "count." On the other hand, recording just the count rate (cpm) on a wipe test or leak test sample gives very little information. State and federal regulations require measurement of the disintegration rate or activity (dpm or μCi). To compute the disintegration rate from the counting rate, it is necessary to know the counting efficiency.

Disintegration Rate (dpm) = Count rate (cpm) /Efficiency (c/d).

Sample Problem 1 illustrates the use of this important detector relationship.

Sample Problem 1

GIVEN:
A wipe test counter has a stated beta detection efficiency of 34% (counts/disintegration). A 100 cm^2 area is wiped for test purposes.
FIND:
The contamination level on a beta wipe with a net count rate after background subtraction of 286 cpm.
SOLUTION:
From the above formula, Dis. Rate (dpm) = Count Rate (cpm) / Eff. (c/d) so here we have Dis Rate = 286 cpm / (34% (c/d) x 1.00 / 100%) = 841 dpm per 100 square cm. We will see, in Chapter 11, that this is an acceptable level of beta contamination.

Gas-Filled Radiation Detectors

Characteristic Curve

A gas-filled detector is constructed by filling an electrically conducting receptacle (often cylindrical) with an appropriate counting gas and inserting a collecting electrode (very often a wire down the axis) which is electrically insulated from the receptacle. By placing a positive potential difference on the collecting electrode relative to the outside receptacle, ion pairs formed when the gas is irradiated will move under the Coulomb force to the charged conductors. This constitutes an electrical signal which can be processed to give information on the radiation field. A sketch of a typical gas-filled detector arrangement is shown in Figure 4. If the tube shown in the

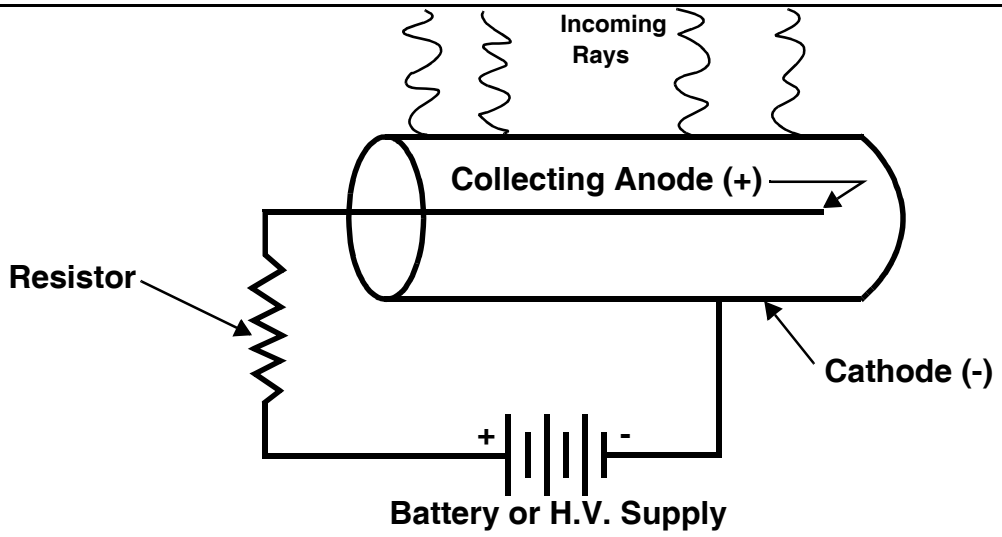


Fig. 4 - The gas-filled detector electrical circuit

figure is exposed to a radiation field of constant intensity and the applied voltage from the battery raised slowly from zero to a large value, the "characteristic curve" for a gas-filled detector results. A typical curve of this type is shown in Figure 5. The three regions which are labeled on the figure represent the usable conditions for common commercial radiation detectors.

At zero applied voltage (potential difference, if you insist!) the electrons released from the filling gas molecules will not experience any forces and hence won't move. As the voltage is raised, the Coulomb force increases and the electrons slowly accelerate toward the central collecting anode. The positive ions of the ion pairs, of course, move toward the outer, negatively charged cylinder (cathode). At low voltages, the ion motions are relatively slow and so an electron heading toward the wire may pass quite close to a positive ion moving toward the cylinder and the two ions will experience a strong mutually attractive Coulomb force which "overrides" the force moving them in their initial directions. They collide and re-combine to form a neutral

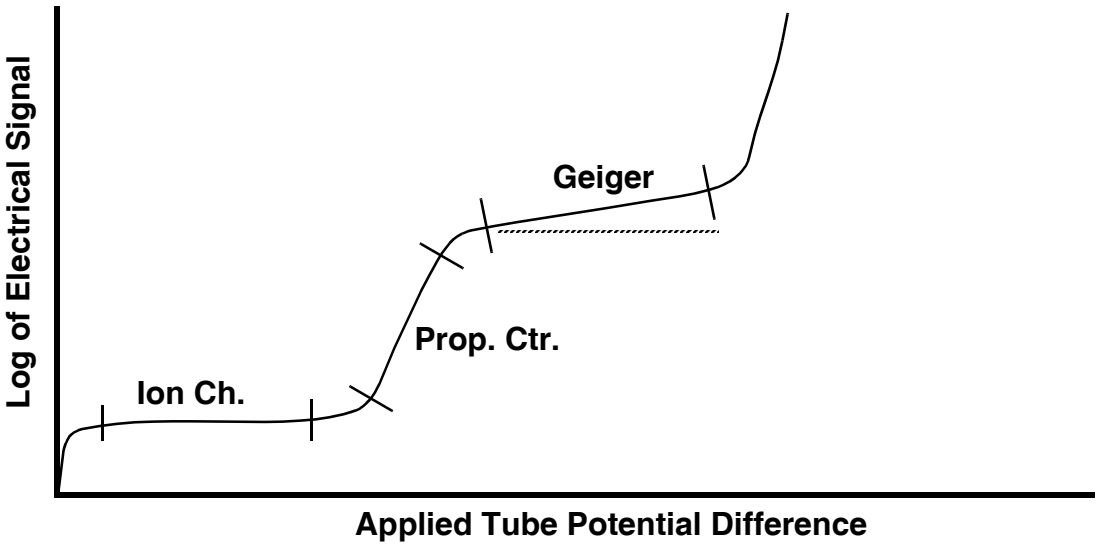


Fig. 5 - The gas-filled detector characteristic curve

gas molecule. This removes their electrical charge from the final signal as they never reach their destinations. As the voltage rises, the attractive forces of the wire and cylinder get progressively stronger, while the mutual attractive force remains the same, so more and more ions can evade other ions and make it all the way to the anode or cathode. Thus, the signal increases with increasing voltage in this region, called the "recombination region." Eventually, with high enough voltage applied, all of the ions formed by the initial primary ray are able to avoid recombination. Then, 100% collection results. The signal reaches a plateau and becomes constant even though the voltage continues to rise. This is the ion chamber region.

Next the signal goes through a sharp rise with voltage, moving through the conditions for proportional counter operation. This is the result of additional "secondary ionization" being produced due to gas multiplication. The individual ions move with such high energy and velocity that they are capable of causing the ionization of gas molecules which they strike. Finally, a second plateau is reached in which Geiger counter operation occurs. In this plateau, a single ion pair injected into the counter is enough to cause complete discharge of the counter. This process will be discussed in detail below.

At the end of the Geiger plateau, the signal rises rapidly and becomes unusable. This is caused by the extremely high voltage leading to the breakdown of the insulating properties of the filling gas. The gas is no longer an insulator, but has become a conductor. This effectively causes a short circuit between the anode and cathode. The battery discharges across the tube. This is called the continuous discharge region. Note that this condition results even if the tube is not in a radiation field. Many tubes will be damaged by operation under the conditions of continuous discharge, so this region should be avoided. This brief description of the characteristic curve will now be expanded by considering each of the three usable regions.

Ion Chambers

The incoming radiation releases electrons, called primary electrons, and positive ions from the chamber wall or the filling gas. The operating potential difference is sufficiently low in the ion chamber region so that no gas multiplication takes place to introduce extra electrical charge, beyond the primary ion pairs, into the detector. Recalling that the W value for air (and many commonly used counter gases) is 33.9 eV/ion pair, the number of ion pairs and the amount of electrical charge deposited in the detector by the incident "ray" is easily calculated as follows:

 # Ion Pairs = Energy Deposited / W [Eqn. 1]

and

Charge Deposited = # Ion Pairs X e [Eqn. 2]

where e represents the electronic charge, i.e., the charge on one electron, 1.6×10^{-19} coulomb. The collected charge resulting from numerous rays interacting in the detector produces a small electric current in the external resistor (see Figure 4). This current is due to multiple rays hitting the chamber. The charge deposited by a single ray is so small that individual rays cannot be distinguished in an ion chamber. One of the unique characteristics of an ion chamber is the extremely small size of this current signal. Sample Problem 2 illustrates this. The current in a typical ion chamber is smaller than the electrical "noise" signals found in most electronic circuits. Therefore, the electrical signal of an ion chamber must be amplified to produce a usable signal.

The circuit commonly used to amplify and detect this small signal is called an electrometer circuit. Based on Ohm's Law (Chapter 1), in order to get a reasonable potential difference generated from an almost impossibly small current, the current should be passed through an unbelievably large resistance. The electrometer circuit thus presents an extremely high input resistance for the ion chamber signal. It amplifies the signal which is then capable of producing a deflection on a conventional meter movement. The addition of a power supply gives a complete instrument. The active device in the electrometer circuit is an "electrometer tube" or the modern solid state equivalent, the metal oxide semiconductor field effect

Sample Problem 2

GIVEN:
An alpha source emitting 6000 Pu-239 alphas per minute is placed inside an air filled ion chamber.
FIND:
The current that will be generated from this source?
SOLUTION:
From Appendix A-1, the Pu-239 alpha carries about 5.15 MeV. Using equation 1 just above, the number of ion pairs formed per second will be 5.15 MeV x 10^6 eV/MeV x 6000/min x 1 min/60 sec ÷ 33.9 eV/ion pair = 1.5×10^7 i.p./sec. The charge deposited will be 1.5×10^7 i.p./sec x 1.6×10^{-19} Coul/i.p. = 2.4×10^{-12} Coul/sec or 2.4×10^{-12} Amp since, from Chapter 1, 1 Amp = 1 Coul/sec.

transistor, abbreviated MOSFET. These devices commonly have input resistances exceeding 10^{14} ohms (a hundred million megohms!!).

For the detector to operate in the ion chamber region of the characteristic curve, the manufacturer must provide enough potential difference to assure 100% collection of the primary ion pairs (i.e., prevent recombination) AND must limit the applied voltage to a value below which gas multiplication takes place. There are two methods commonly used in the design of an ion chamber to prevent gas multiplication. The first depends on the fact that the W value amount of energy must be supplied to produce ion pairs. In the case of ion pairs formed in the ion chamber, each ion carries one electronic charge, e. Thus, the maximum energy that the ion can pick up in accelerating through the chamber to the collecting electrode is (from Chap. 6, Fig. 50) the product of the charge, e, and the applied potential difference between the anode and cathode. A 100 volt applied difference can lead to ions with 100 eV of energy, maximum. But the design objective is to limit ions to less than 33.9 eV. Then, they will not be able to produce secondary ionization since they carry less than the W value. This is most easily accomplished in practical ion chamber designs by using a battery or power supply of less than 33 volts in the chamber circuit. One commonly used battery in portable ion chamber survey meters is the rectangular 22 1/2 volt size.

The second method of preventing gas multiplication is also based on physics. It is necessary to have an alternative method due to the fact that some specialized ion chamber designs (e.g., in-core reactor instrumentation) require over 1000 volts potential difference across the chamber. These chambers still operate with no gas multiplication as is required of an ion chamber. This can be done if no ion can acquire more than 33.9 eV of energy between collisions (over one mean free path). If the electric field inside the chamber is kept small and the gas pressure relatively high, the ions will collide with gas molecules frequently enough to keep their average energy below the critical W value. In practice this is done by making the collecting electrode with a large physical diameter. As was discovered by Benjamin Franklin in designing lightning rods, a small diameter electrical conductor creates large electric field strengths and, conversely, a large diameter leads to small electric fields. In the ion chamber, the electric field must be small so the manufacturer uses a large diameter collecting rod in the middle of the chamber, effectively preventing gas multiplication. Figure 6 is a photo of a traditional ion chamber instrument named a Cutie Pie. Note the large 6 mm (1/4") diameter rod used for the anode.

Herein lies one of the great mysteries of Health Physics. Where did the name Cutie Pie really come from?? One version of the story says the name was coined in the Manhattan Project based on the assigned circuit designation, CP, standing for "capacitor" which the detector approximated. Another story says the name was purely whimsical, selected to confuse enemy agents as to the purpose of the circuit.

In the end view of this same chamber (Figure 7), the electrometer tube is visible as the peanut shaped object. An inside view of another instrument is seen in Figure 8. The holder used for the ion chamber battery is labeled 22.5 volts. Note also that the electrometer tube does not have a plug that mates with a tube socket as was conventional in vacuum tube circuits. This was not an economy move by the manufacturer. The resistance between the pins of a tube socket, while high enough for

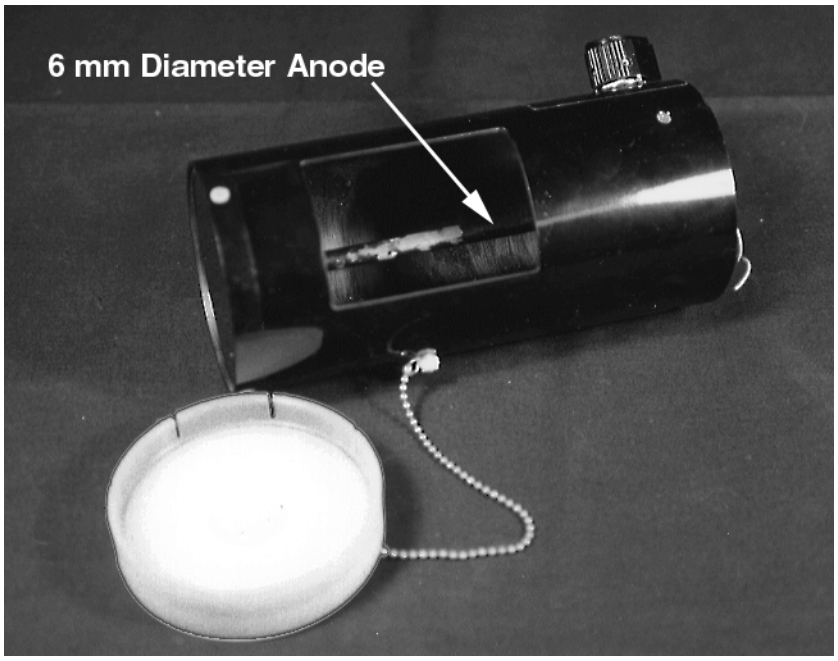


Fig. 6 - A Cutie Pie ion chamber with a side cutout to show the anode rod

ordinary circuits, is much too low for the impedances presented by the electrometer tube or by a MOSFET. The socket would effectively short out the signal if it were used. Notice also the "high meg" resistors shown in Figure 7. These require special manufacturing techniques. In an ordinary

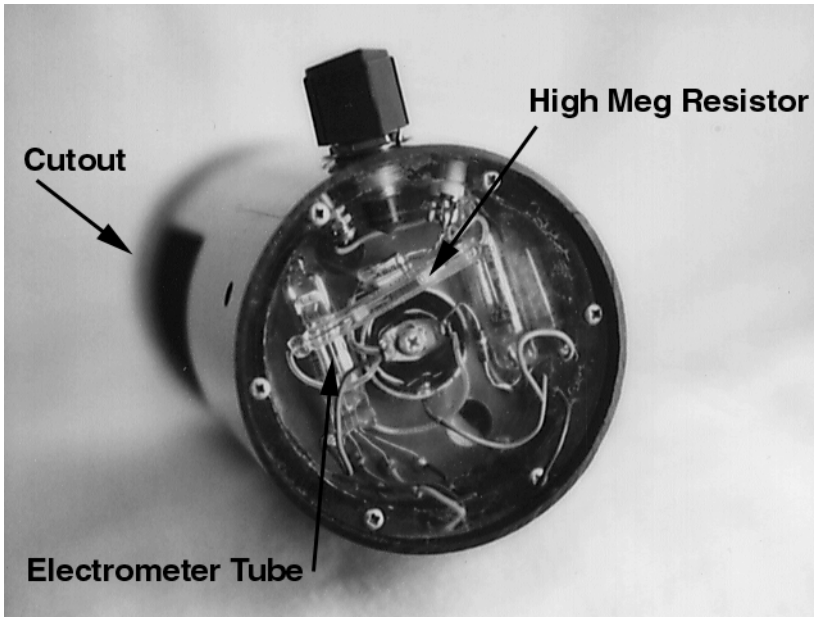


Fig. 7 - End view of a cut open Cutie Pie instrument

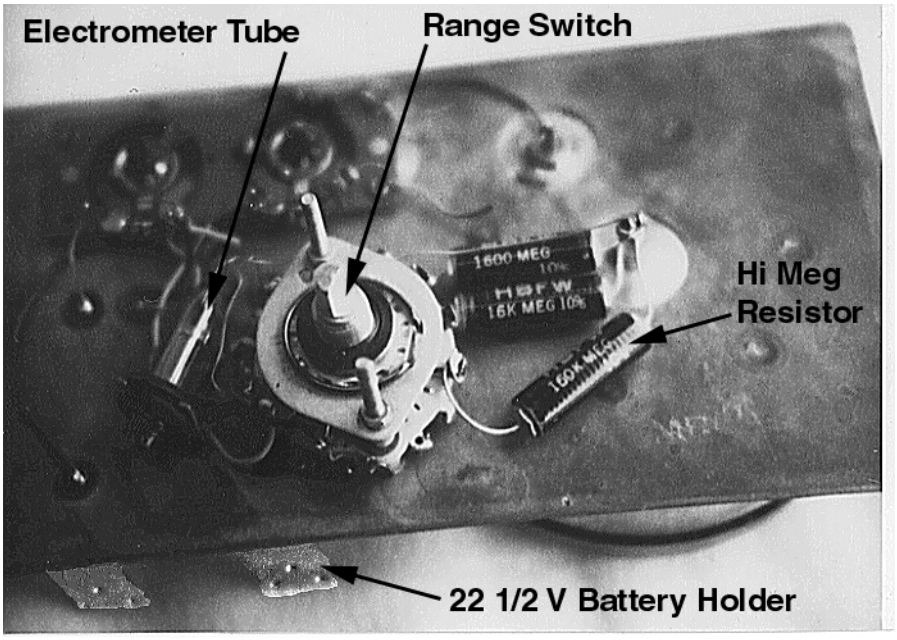
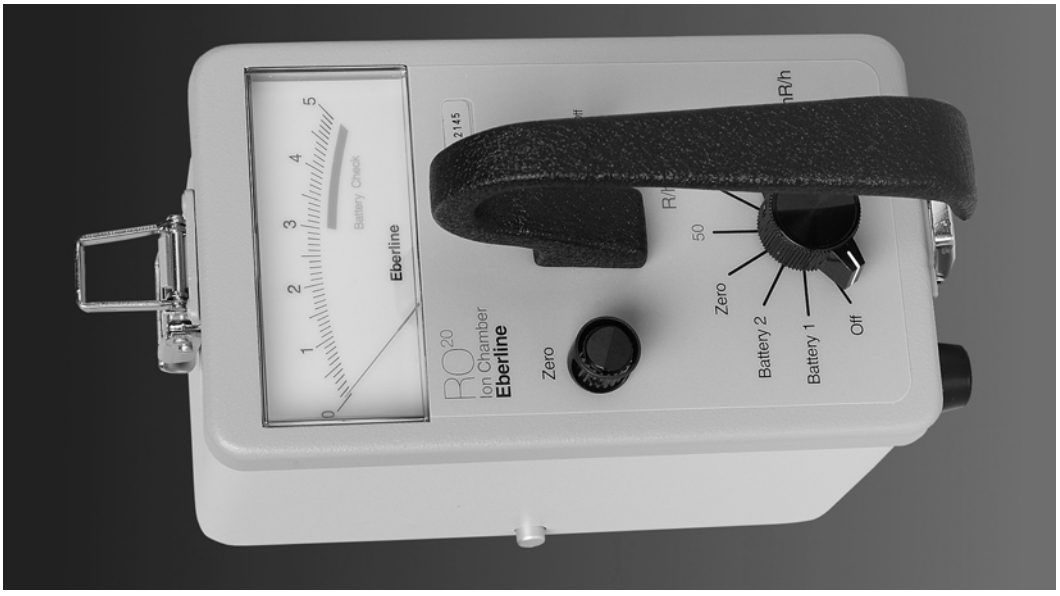


Fig. 8 - Inside view of another ion chamber instrument

electronics supply house, the largest resistors available are about 20 megohms.

A newer version of the old Cutie Pie that has become extremely popular at nuclear power plants is the Eberline RO-20 (RO = "Rad Owl"). Figure 9 is a photo of



Courtesy, Eberline Instrument Co.

Fig. 9 - The RO-20 ion chamber instrument

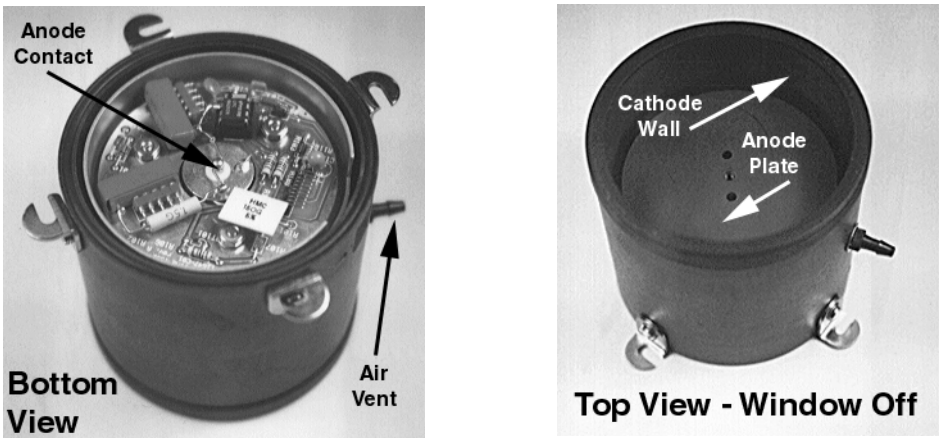


Fig. 10 - The internal ion chamber itself from the RO-20

the complete instrument. The internal ion chamber itself is shown in Figure 10. The RO-20 chamber wall is constructed of low atomic number materials and the cylindrical shape has been "squashed" in length compared to the more traditional Cutie Pie. These features give the instrument a very flat energy response (Figure 11) and result in a beta ray "correction factor" which is smaller than in many similar units. (The beta response will be discussed in Chapter 12). The meter reads in five linear ranges from 0-5-50-500 mR/hr and 0-5-50 R/hr.

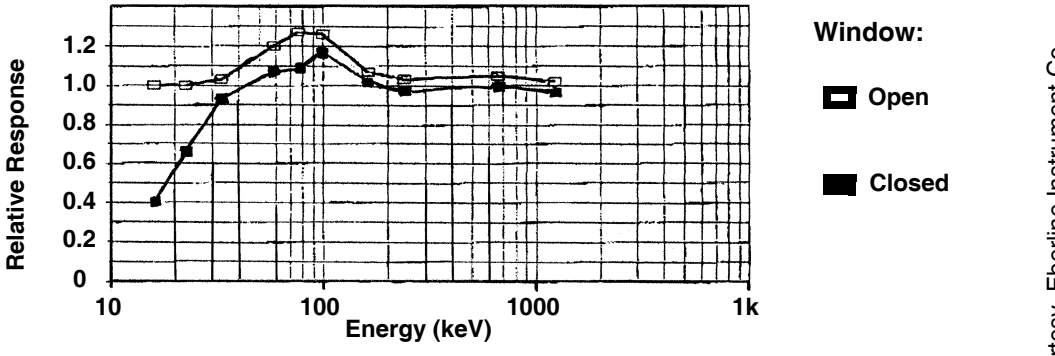


Fig. 11 - The photon energy response of the RO-20

Figure 12 shows a pocket ion chamber instrument for military applications. It has color coded scales which indicate when to "take cover" and when "travel is permissible." It is of interest to note that, in this application, travel is permitted in exposure rates up to 20 R per hour! The designer has placed the electrometer tube inside the hermetically sealed ion chamber on this unit. This is common practice to prevent tampering with the sensitive tube.

An instrument that has been slowly growing in popularity is the Extrapolation Chamber. Originally developed as a standards laboratory instrument, the commercial availability has made them more accessible.

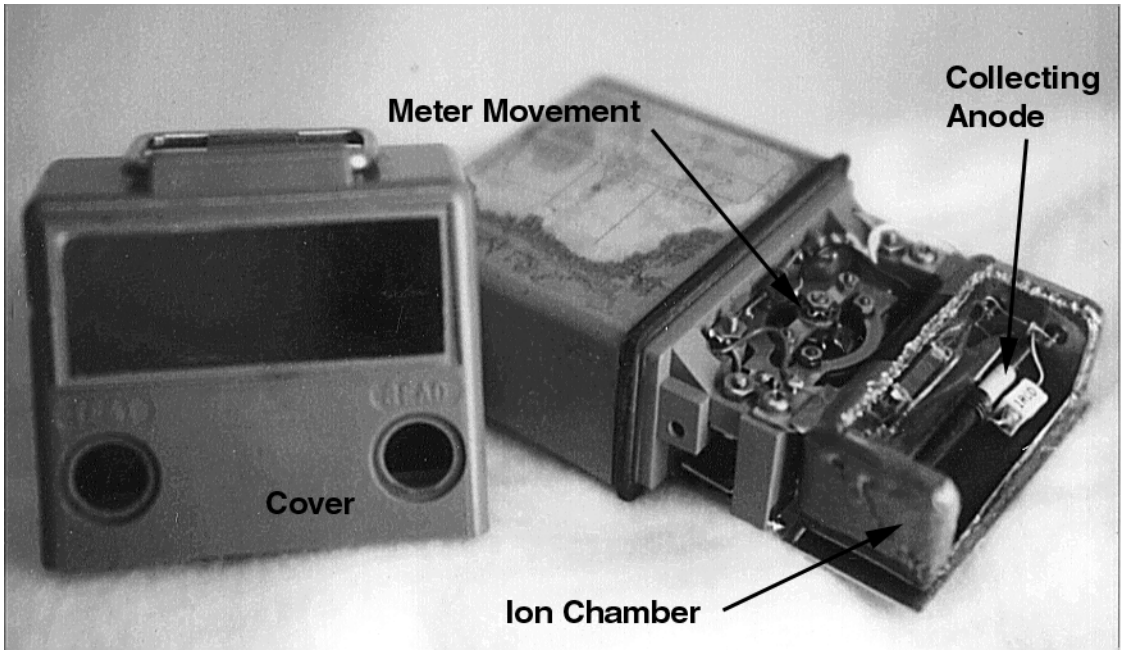


Fig. 12 - Cut away view of another ion chamber instrument

Their chief use in radiation protection is for beta dosimetry. As discussed in earlier chapters, most of the signal inside an ion chamber is due to electrons generated in the wall of the chamber. In an extrapolation chamber, the wall effect is eliminated by measuring the change in the ionization for a small change in chamber volume, caused by moving a piston which defines the back wall of the chamber (see Figures 13 and 14).

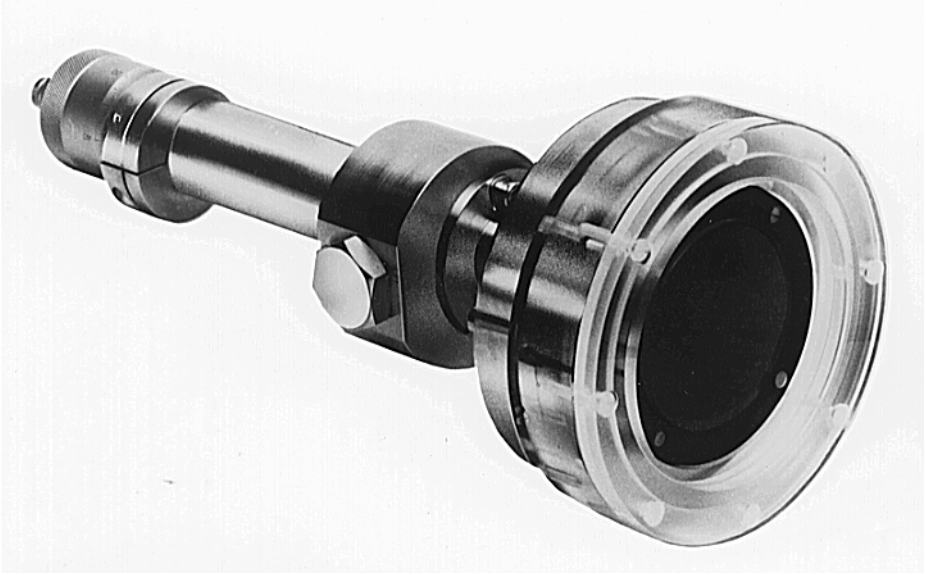


Fig. 13 - A complete extrapolation chamber instrument

Courtesy Nuclear Associates, Carle Place, NY 11514

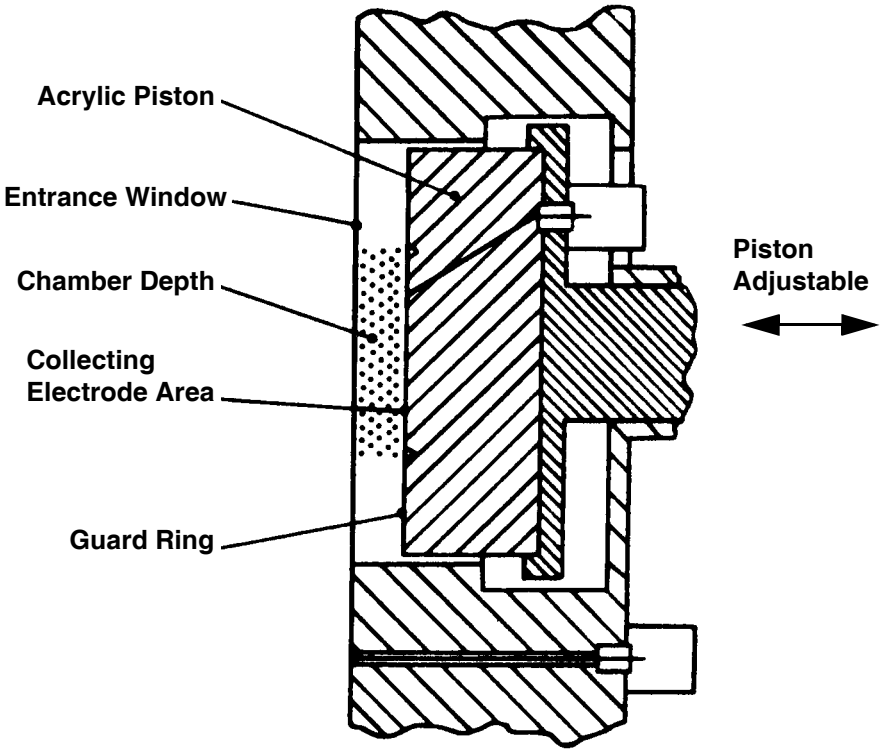


Fig. 14 - Cross-sectional drawing of an extrapolation chamber

The next photo (Figure 15) is a final example of a pocket chamber instrument operating in the ion chamber region with no gas multiplication. The side is cut away to show the inner workings. The instrument consists of a quartz fiber electroscope in the ion chamber section, a charging pin assembly and a microscope. When connected to a power source, the electroscope is charged and the quartz fiber is repelled up away from the horseshoe. When exposed to radiation, the ion pairs formed in the chamber section will be attracted to the electroscope which cancels some of the charge. This allows the quartz fiber to move back down closer to the horseshoe. This movement appears as an upscale reading on the microscope which is focused on the top of the fiber.

Note that some pocket chambers have a boron coating applied inside the metal wall ion chamber section. Thermal neutrons can be captured in the boron, releasing a charged alpha particle which exposes the ion chamber. Thus, the boron lined pocket chamber is sensitive to thermal neutrons. Chambers can be made sensitive to fast neutrons by adding a plastic wall inside the metal wall. Protons released by elastic scattering interactions will expose the air causing a reading.

Unsealed ion chambers, which include all the chambers discussed previously in this chapter, are actually vented to the atmosphere by a small hole drilled in the chamber wall. This means that the chamber air pressure will vary with changes in barometric pressure over time and will vary depending on the altitude the meter is being used at. If it is calibrated at sea level and then used at 10,000 feet, the meter

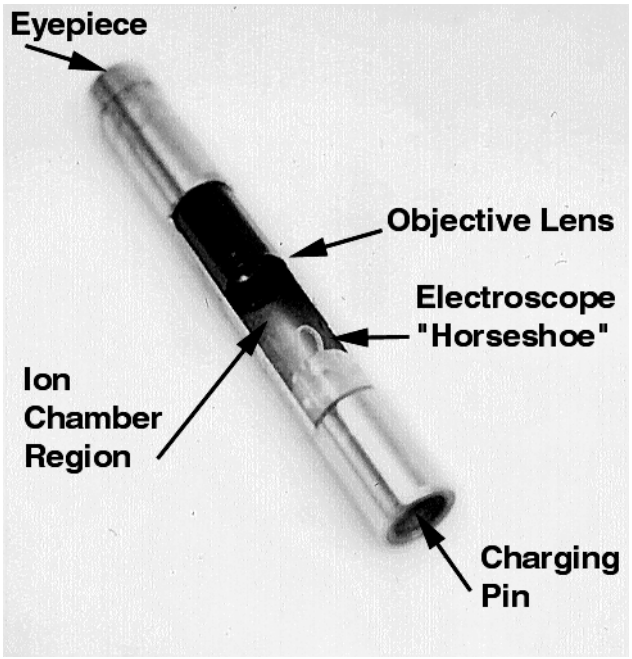


Fig. 15 - A pocket chamber with the side cut away

will read 30% below the true dose rate! (This is due to the 30% reduction in air pressure between sea level and 10,000 ft.) The ambient temperature also affects the reading. Remember that the probability of a gamma ray interaction (i.e., the chamber reading) depends on the number of atoms in the gamma ray's path.

You may recall from chemistry classes that the Kinetic Theory of Gases tells us that an ideal gas obeys the equation P V = n R T where n is the mass of the gas in moles and R is the universal gas constant. Since the sensitivity of a gas-filled ion chamber is directly proportional to the mass of contained gas, then, a simple correction factor can be applied to calculate the sensitivity of a chamber which was calibrated at one temperature and pressure but is now being used at a different temperature and pressure. Usually a calibration laboratory will state it's calibration at standard temperature (273° K) and standard pressure (760 mm Hg.). Then, the correction factor (with T in degrees C) that multiplies the uncorrected reading to give the corrected reading is as shown.

$$\left(\left(\frac{T + 273}{273}\right) X \frac{760}{P}\right) = \text{Correction Factor}$$

Occasionally a technician may encounter a pressurized ion chamber (PIC) instrument. In these meters, the ion chamber is sealed from the atmosphere so they do not exhibit the problems caused by pressure and temperature variations and humidity entering the chamber through the vent hole. They have some additional attractive features. The chamber can be made much smaller. By filling the chamber above atmospheric pressure, the effective volume is increased. Thus, a 20 cc chamber pressurized to 10 atmospheres (about 150 psi) would have the gamma ray sensitivity

of a 200 cc unsealed chamber. A chamber at 10 atmospheres pressure will put 10 times more atoms in a cm of path length than a chamber at 1 atmosphere of pressure.

Another advantage of the sealed pressurized ion chamber is that it can be used in an environment that includes radioactive gases. In the unsealed chamber, radioactive gas will leak into the chamber through the vent and can lead to very spurious readings as the signal will now be a combination of the external field plus the internally trapped gas. A disadvantage of the PIC family of meters is that the chambers sometimes lose their seal. Then, the chamber gas leaks out and the sensitivity of the meter is greatly reduced. (It will be reduced by the ratio of the sealed gas pressure to the leaky gas pressure.) Unfortunately, the technician has no way of knowing if this has occurred. The meter will still read, although well below the true rate. When using this type of meter, it is most important to perform frequent tests with a check source to verify that the meter is at full sensitivity.

Figure 16 shows a recent commercial entry in the PIC family, the Ludlum 9DP. It reads to 5 R/hr, is autoranging and has a built-in USB port.

Fig. 16 - The Ludlum pressurized ion chamber, Model 9DP

Courtesy, Ludlum Measurements, Inc.

Proportional Counters

As the applied voltage is increased, the next usable region of the characteristic curve is the steep slope for operation of proportional counters (review Figure 5). The increased potential difference now provides a strong enough Coulomb force to accelerate ions to energies above the W value. Thus, gas multiplication takes place. The

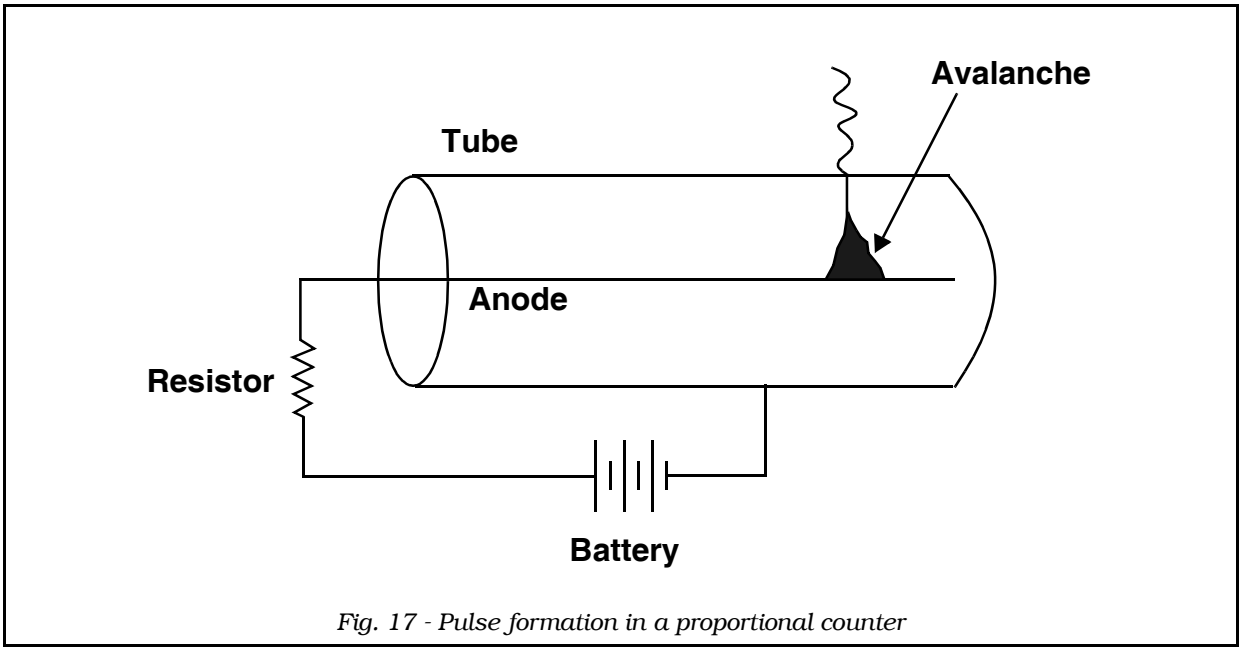


Fig. 17 - Pulse formation in a proportional counter

speeding ions collide with neutral filling gas molecules and rip electrons off, releasing more ion pairs in the gas in a sort of chain reaction phenomenon. At the central collecting electrode, the swarm of electrons converging there is termed an avalanche. The size of the avalanche is dependent on the gas multiplication factor, M, which is defined as the number of electrons collected divided by the number of electrons produced by the original ionizing event (the primary ray). In a proportional counter, the gas multiplication factor can be as large as 10^6. For each electron produced by the primary ray, one million electrons are collected. (See Figure 17 and Sample Problem 3.) Because of gas multiplication, the proportional counter is able to detect individual ionizing rays. This is the reason for the word counter in the name. Recall that the ion chamber can only measure the cumulative effects of a field of many ionizing rays per second.

An important concept related to radiation counters that distinguish individual events is the "dead time." This is defined as the minimum length of time that must elapse between two ionizing events occurring in the counter such that they are distinguished by the counter, that is, recognized as TWO events instead of as ONE event. (If two events arrive separated by less than the dead time, the counter forms only one pulse instead of two). In the proportional counter, the avalanche is limited to a small section of the collecting electrode (see Figure 17). The tube is therefore able to "clear out" the collected charge quite rapidly and the dead time is short. In a typical proportional counter design, the dead time is about a half a microsecond. As will be seen, this is much shorter than the dead time in a Geiger counter. This means the typical proportional counter can operate at higher counting rates (a stronger radiation field) than the typical Geiger counter.

In discussing signal formation in the ion chamber, it was concluded that large diameter collecting electrodes were needed to keep the electric field strength small enough to prevent gas multiplication. The proportional counter is the "other side of

Sample Problem 3

GIVEN:

A beta particle deposits 800 keV in a proportional counter.

FIND:

If 4 x 10^{-10} C. of charge are collected for this event, what is M for this counter?

SOLUTION:

# of electrons collected = Q/e = 4 x 10^{-10} C/1.6 x 10^{-19} C/elect. = 2.5 x 10^{9} electrons. The primary particle deposited 800 kev x 1000 eV/keV ÷ 33.9 eV/electron or 2.36 x 10^{4} electrons. Thus, the multiplication factor is 2.5 x 10^{9} / 2.36 x 10^{4} = M = about 10^{5}.

the coin." In this counter, large gas multiplication is desired. Very small diameter collecting electrodes are used in the design of proportional counters to meet this objective. The counter shown in the photo in Figure 18 illustrates this fundamental design difference.

The size of the electrical pulses developed across the resistor in the proportional counter circuit is of the order of 10s of millivolts. This is still not big enough for direct use in most electronics counting circuits. A preamplifier is commonly used to increase the pulse height by about a hundred times. These larger, amplified pulses can be more easily utilized. An example of a commercial counter using a preamp will be shown later in this Chapter.

Recall that the proportional counter operates on the steepest section of the entire characteristic curve for gas-filled detectors. This means that a small change in the applied operating potential difference will

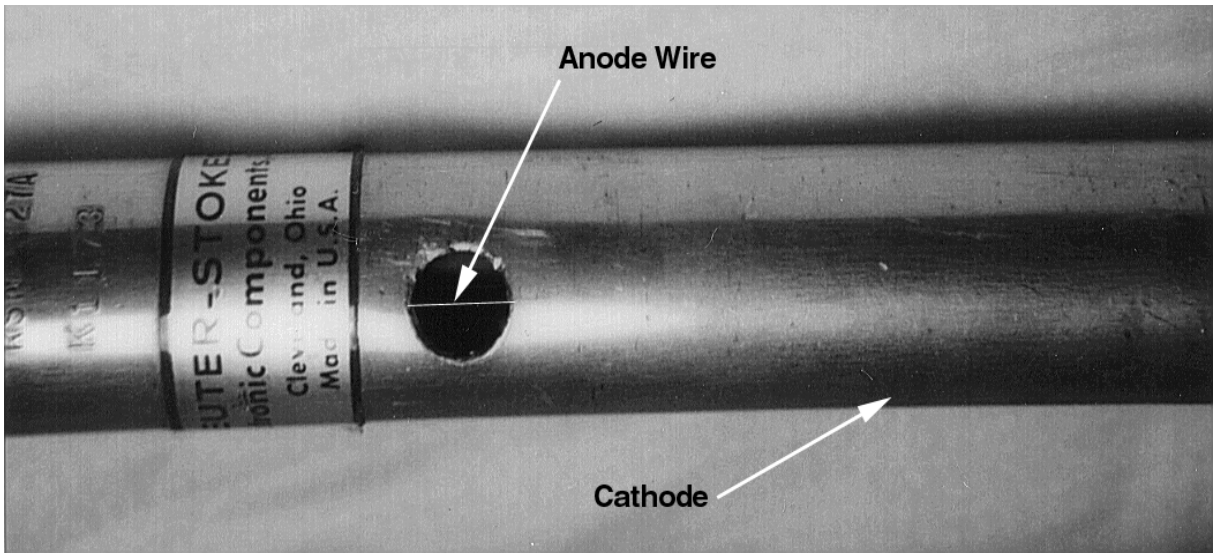


Fig. 18 - A proportional counter showing the small anode wire

cause a large change in the electrical signal. In many applications this will cause large changes in the calibration factor for a particular counter. The "bottom line" is that most proportional counter circuits must be equipped with tight voltage regulators to prevent changes in the tube voltage due to line voltage fluctuations or aging components or batteries.

Consider the word "proportional" in the name of the detector. This too is related to the physics of the pulse formation process. Although several different parameters are proportional to each other in the device, the name comes from the proportionality between the output pulse voltage amplitude and the energy deposited by the primary ionizing event. This proportionality is due to a chain of events. The energy deposited in the filling gas is directly proportional to the number of ion pairs released (the proportionality constant is, of course, 1 ÷ W value). The number of ion pairs collected is proportional to those released (the proportionality constant being the gas multiplication factor). Finally, the pulse height is directly proportional to the circuit resistor (Ohm's Law). If the external electronic circuit is designed to respond to different pulse heights, then, the proportional counter can be used to measure the energies of the incoming particles.

This principle of proportionality is often used in radiation protection for telling alpha counts and beta counts apart. For example, this is necessary in counting wipe test samples. The radiation protection limits for leaking sources or contamination on working surfaces are very different for alpha emitters in comparison to beta emitters. Thus, a detector is required that can distinguish these two different radiations. The proportional counter is able to do this due to several factors. Because of stopping power differences, common alphas have ranges of only a few centimeters in air while common betas will travel a meter or more. Thus, an alpha will give up almost all of its energy to ion pairs in the gas of a common size proportional counter while the beta expends only part of its energy in the gas. The remainder is absorbed by the counter wall. In addition, alpha particles virtually always carry much higher energy than beta particles. This fact was stressed in Chapter 2. Finally, remember that on the average, a beta particle carries only one third of the decay energy, E_{max}.

Because of all of these factors, the electrical pulses produced by alpha particles are significantly higher in voltage amplitude than pulses from beta particles. If a mixed alpha–beta source was counted on a proportional counter, an oscilloscope connected across the tube resistor would show a display like the sketch in Figure 19. The alpha pulses tend to have the same amplitude while the beta pulses spread out below them due to the variable energy shared with the accompanying neutrino. By using an electronic counter (scaler) which has a voltage discriminator (a circuit that will reject pulse heights below an adjustable minimum) alphas can be counted separately from betas on the same sample in a proportional counter. Since the gas multiplication factor is directly dependent on the applied counter voltage, a low voltage will allow only the alpha pulses to be sufficiently large to be counted. Raising the applied counter voltage increases the height of all pulses, allowing now both alpha and beta pulses to be counted. By taking the difference in the counts at two voltages, the alpha counts and beta counts can both be obtained. The response curve for such an instrument in the presence of a mixed alpha–beta field is shown in Figure 19.

In more complex modern instruments, it is possible to SIMULTANEOUSLY count alphas separately from betas. This is done by incorporating

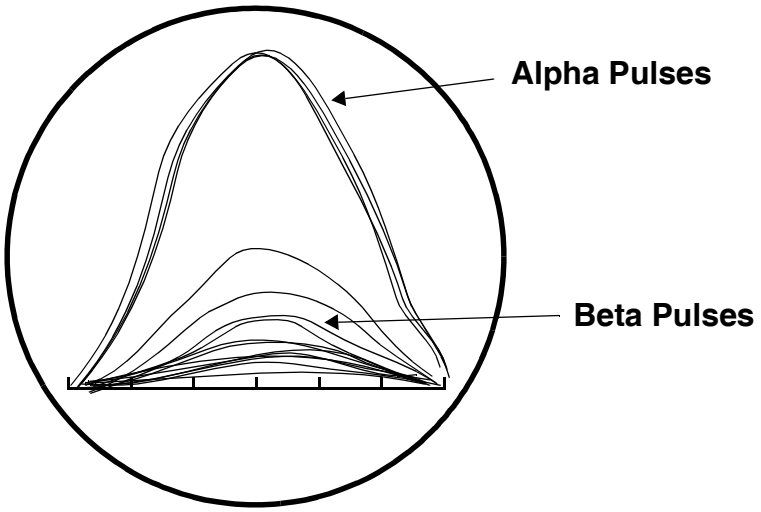


Fig. 19 - Oscilloscope display for a mixed alpha/beta source

two "single channel analyzers" in the electronics. A single channel analyzer allows a pulse to be sent to an electronic counter or rate meter only if the pulse height falls in the "voltage window" set by the adjustable controls. The pulse must exceed the lower level setting but not exceed the upper level setting. This is illustrated by Figure 21. One voltage window is adjusted to record only the alpha pulses and the second window (set below the first) to record only the beta pulses.

The counter used for counting alphas and betas on wipe samples is usually a bench-mounted gas flow proportional counter, although some manufacturers recommend a liquid scintillation system for this task. A gas flow proportional counter has the counter gas continuously replenished from a gas cylinder, pressure regulator and

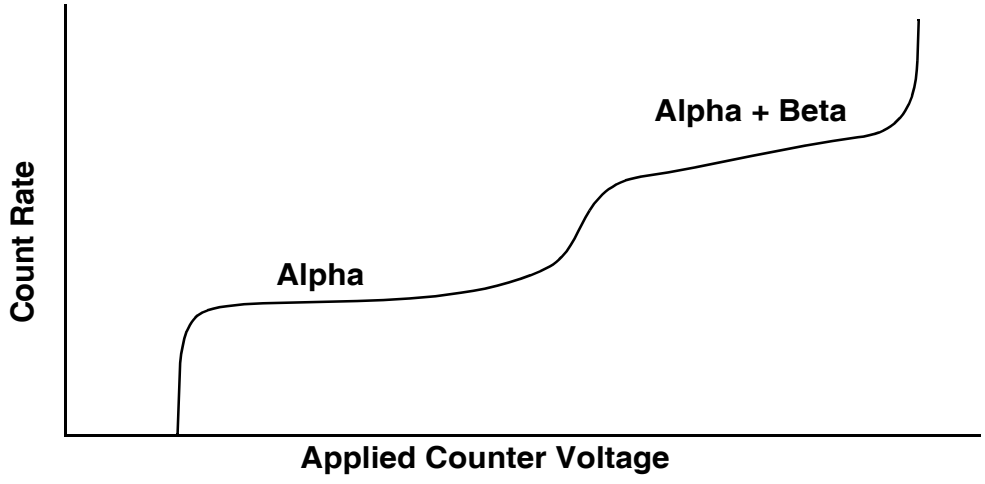


Fig. 20 - Counter response curve for a mixed alpha/beta source

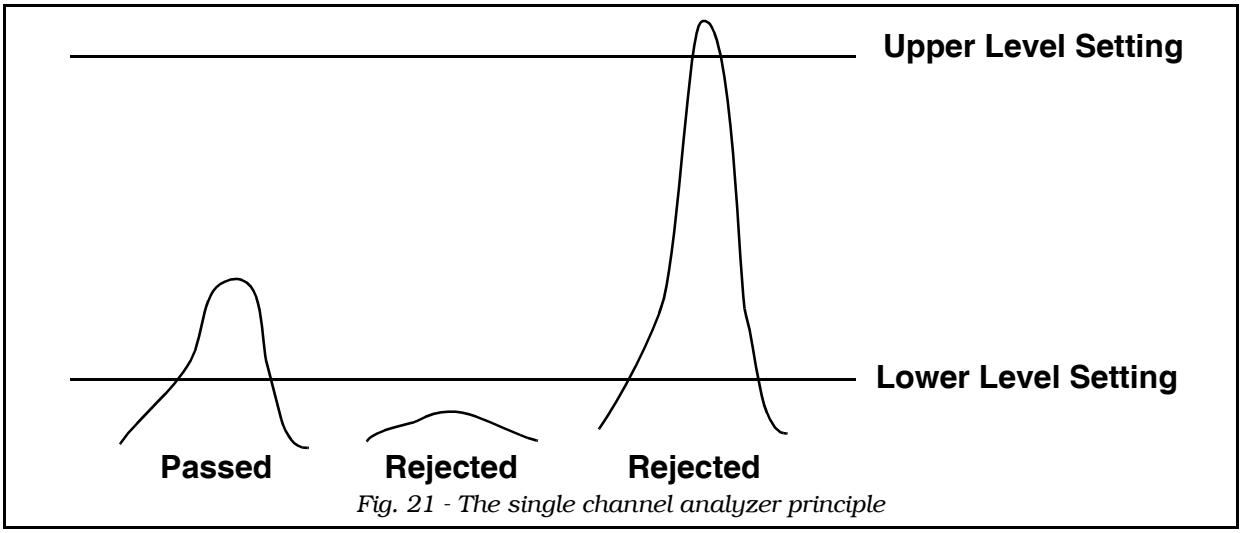


Fig. 21 - The single channel analyzer principle

valve assembly. A common gas mixture used is called P-10. It consists of 10% methane and 90% argon. These counters must have an extremely thin entrance window (or none at all, i.e., the windowless flow counter) to allow the short range alpha and beta particles to enter. Figure 22 shows a commercially available model of a bench mounted gas flow proportional counting system. See Sample Prob. 4.

Gas flow proportional counters need a little extra care and handling compared to a sealed proportional counter. If the gas lines must be removed, be sure to cap the detector gas connections to reduce the chance of contaminants or moisture from entering. Water vapor inside the counter will cause arcing and erratic behavior when it is reconnected to the gas and high voltage is applied. When starting up a flow counter that has been shut down, it is important to purge the counter with counting gas. This will gradually clean out any water vapor or atmospheric gases that may have leaked in during the downtime. At regular flow rates, the counter may need as much as 30 minutes to produce stable operation. If the flow gas pressure can be raised, a shorter flush time can be tolerated.

These instruments also have the same problem as unsealed ion chambers - the sensitivity varies with operating temperature and with

Sample Problem 4

GIVEN:
A proportional counter detects 23 net cpm in the alpha plateau and 450 net cpm when the applied voltage is raised to the beta plateau. Alpha and beta efficiencies are 0.40 ct/dis and 0.27 ct/dis respectively.
FIND:
What are the alpha and beta activities, in Bq, on this sample?
SOLUTION:
Alpha activity = 23 cpm/0.40 ct/dis = 57.5 dpm x 1/60 Bq/dpm = 0.96 Bq.
Beta activity = (450-23) cpm/0.27 ct/dis = 1581 dpm x 1/60 Bq/dpm = 26 Bq.
Since in the beta plateau, alphas are also counted, then, the 23 alpha cpm is subtracted from the 450 cpm total rate before correcting for counter efficiency.

Courtesy of Canberra Industries

Fig. 22 - A commercial alpha/beta wipe counter

barometric pressure changes (such as elevation changes). Usually the operating voltage should be readjusted if the unit is to be operated at an elevation difference of more than 1000 feet between the calibration altitude and the operating altitude.

The operating voltage determines both the sensitivity to radiation and the background count rate. If the voltage is too high, the background will be higher than acceptable. If the voltage is too low, the counter will lack sensitivity.

An example of a thin window portable proportional counter hand probe for alpha surveys is shown in Figure 23. The thin window is shown removed in this view. The proportional counter consists of the four parallel channels shown. Each channel has a wire a few ten thousands of an inch in diameter (smaller than a human hair) stretched along the length. The small diameter collecting wire gives the counter a huge gas multiplication factor, M, to allow the alpha pulses to register above background interference. The superfine collecting wires are invisible in this photograph due to their small diameter.

A final popular application of counters that are operating in the proportional region of the characteristic curve is a neutron detector. Since neutrons are not charged, they will not directly produce ionization in a detector. However, it has been discovered that if a boron-10 nucleus captures a thermal neutron, an alpha

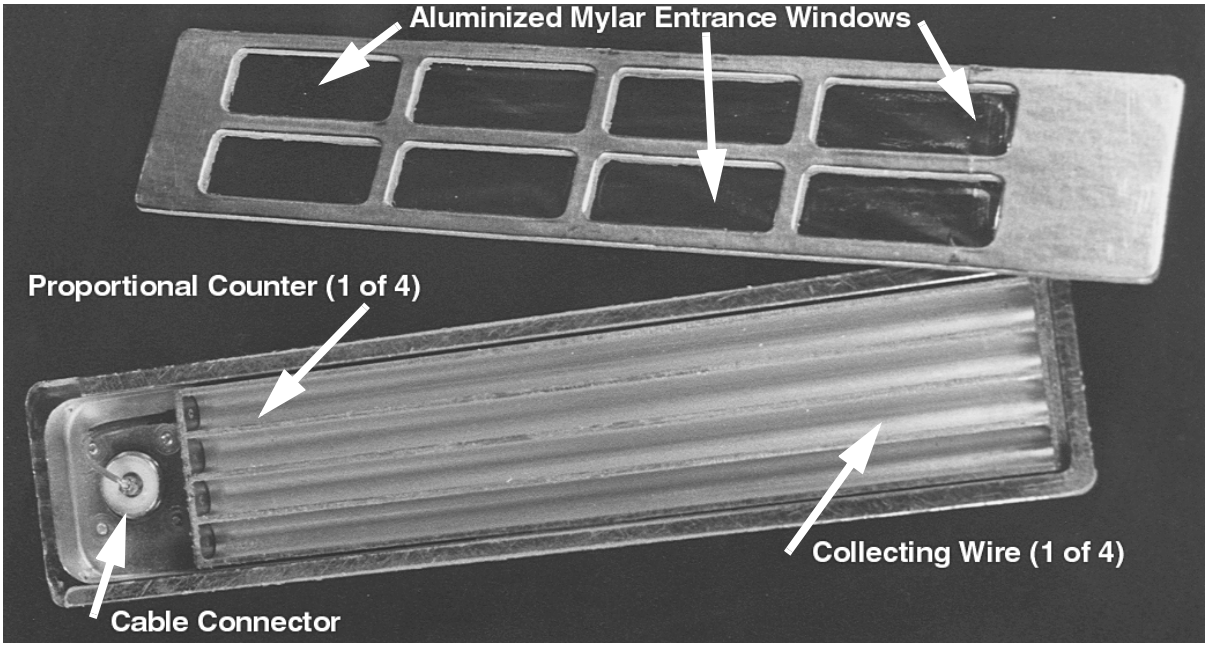


Fig. 23 - An air-filled proportional counter probe for alphas

$$^{10}_{5}B + ^{1}_{0}n \rightarrow ^{7}_{3}Li + ^{4}_{2}He$$

Fig. 24 - The reaction for detection of thermal neutrons

particle (a directly ionizing radiation) is produced. This reaction is shown in Figure 24. If the reaction takes place inside a proportional counter, the resulting alpha particle can be easily counted. Typically, the proportional counter designed for neutron detection has a rather thick metal wall which prevents any external particulate radiation from entering the counter gas. The boron is incorporated in one of two ways. The counter filling gas can be boron trifluoride, BF_3, in which the boron-10 atoms are part of the gas molecule. Alternately, a solid thin layer of boron can be coated on the inside wall of the counter. In either configuration, a sensitive, low background thermal neutron detector results. Note that the probability (cross section) of fast neutron capture by the ^{10}B is very small. Only thermal neutrons are caught with high probability. On the other hand, fast neutrons become thermal neutrons when their energy is dissipated. The basic counter can be made sensitive to fast neutrons by surrounding it with a good neutron moderator, i.e., something with lots of hydrogen such as wax or plastic. An example of a counter operated in this way is the “fast/slow neutron detector” shown in the photos of Figures 25 and 26.

Note the transistorized preamplifier built right into the probe handle to increase pulse amplitudes. The moderator used by this manufacturer is wax. Because of problems with wax at elevated temperatures, most designs now use polyethylene moderators. Finally, note the thin sheet of cadmium metal wrapped outside of

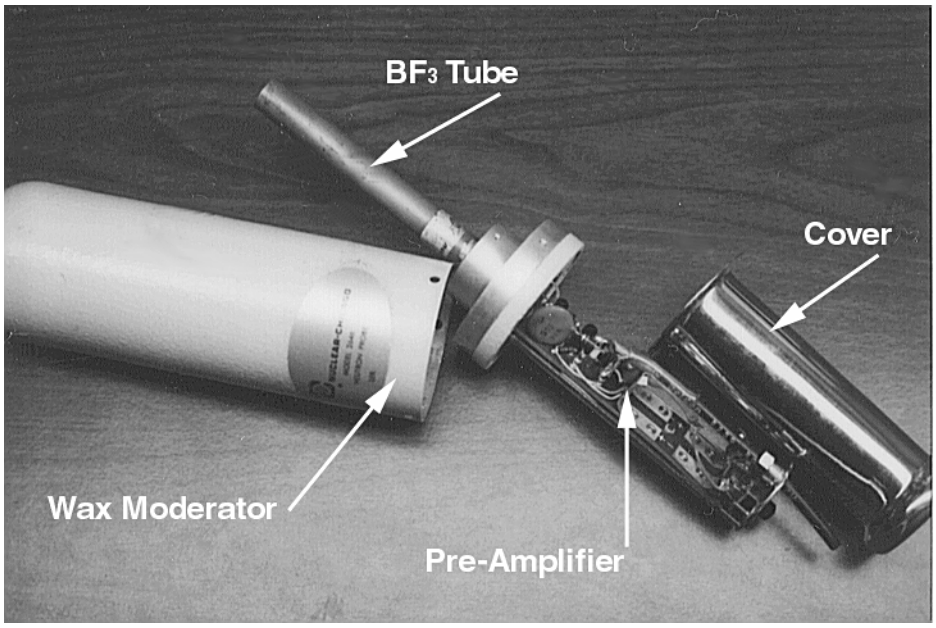


Fig. 25 - Disassembled fast/slow neutron proportional probe

the moderator. Cadmium strongly absorbs thermal neutrons. This prevents confusion by allowing the fast neutrons to be read separately from the thermal neutrons. To use the counter for thermal neutron surveys, the moderator cap assembly is slipped off

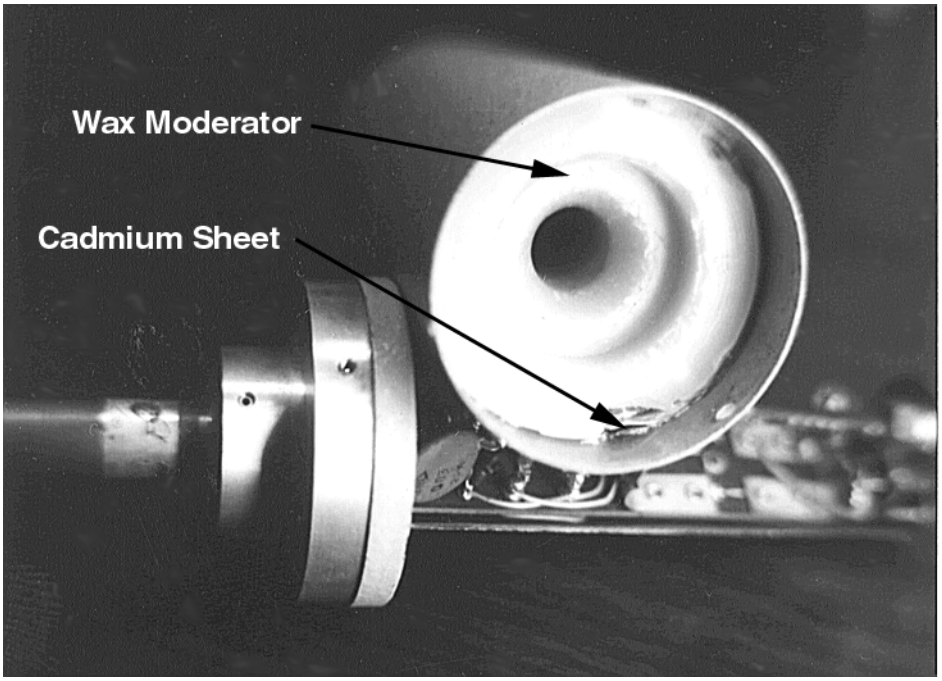


Fig. 26 - Close-up of moderator with cadmium thermal neutron shield

and readings taken. The instrument records only thermals. Then, to measure the fast neutrons, the moderator with the cadmium thermal neutron shield is placed over the tube. Fast neutrons are able to easily penetrate the cadmium sheet. They enter the moderator where they are slowed down and detected. On the other hand, ambient thermal neutrons present at the measurement location are almost totally stopped by the cadmium which has a tremendously large cross section for THERMAL neutron capture. Thus, any counts recorded are due only to fast neutrons with the cap on.

Geiger Counters

The Geiger counter (or Geiger-Mueller counter or GM counter) operates in the last usable region of the characteristic curve, Figure 5. In this region, the output pulse amplitude is of constant height regardless of the energy deposited in the counter. Thus, the Geiger counter is unable to distinguish the energies of the incoming rays as the proportional counter could. In the Geiger region, the gas multiplication factor is between 10^8 and 10^{10} so that the output pulses are of the order of a few volts in height. No preamplifiers are usually required for GM circuits.

The pulse formation takes place in stages over a relatively long period of time. The initial energy deposited by an incoming ray releases ion pairs in the gas and forms an avalanche as was the case with the proportional counter. The chief difference, at this stage, is that the Geiger avalanche contains more ions since the multiplication factor is larger. When this avalanche reaches the collecting wire, the local energy density is so high that ultraviolet light photons are emitted. These interact with the filling gas or tube wall to produce photoelectrons. The photoelectron, being charged, initiates another avalanche at some other location on the collecting wire. This process repeats several times until the collecting wire is eventually completely enveloped by ions. (See Figure 27). To turn off the process and "reset" the tube, a

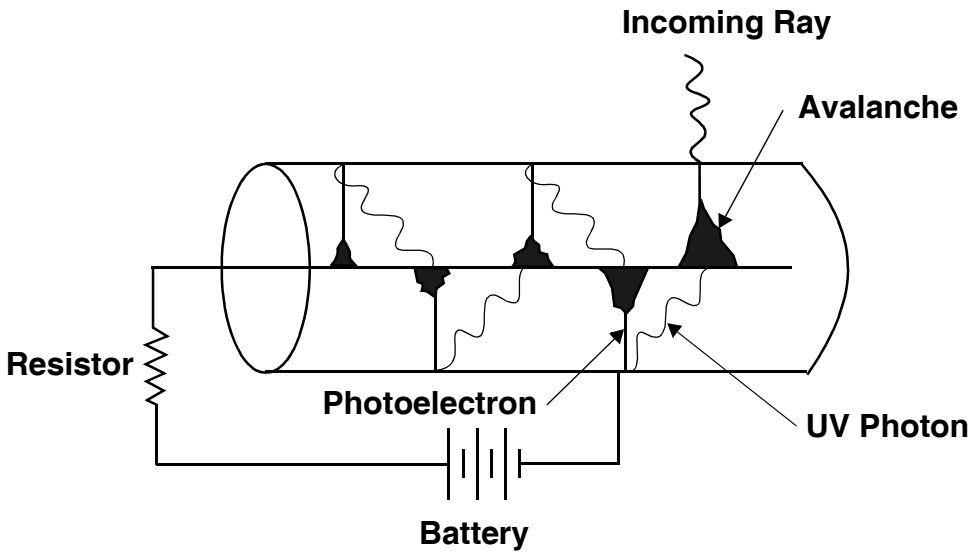


Fig. 27 - The spread of the GM discharge by multiple avalanches

quenching gas is added during manufacturing along with the regular filling gas (usually argon or neon). Common quenching gases are alcohol or chlorine.

An organic quench gas (e.g., alcohol) is used up in the process while an inorganic quench gas (e.g., chlorine) recombines to provide a continuous supply. The quenching gas molecules pick up the positive ion charges and head toward the cathode wall where they neutralize and dissociate into neutral species. In addition, the ions around the collecting wire act as an electrostatic shield which reduces the electric field strength and, consequently, the gas multiplication factor drops dramatically as the discharge spreads. All of these factors cause the succession of avalanches to cease, and the ions are then cleared from the tube. Since this takes much more time than the pulse formation in the proportional counter with a single avalanche, the Geiger counter dead time is much larger. A typical value would be of the order of 300 microseconds, about 600 times longer than the proportional counter. The dead time of a given counter can be experimentally determined.

There are two other terms which are closely related to dead time. These are resolving time and recovery time. Both the dead time and the recovery time depend on the Geiger tube. The resolving time depends on the electronic circuitry. Their interrelationship is illustrated by the sketch in Figure 28.

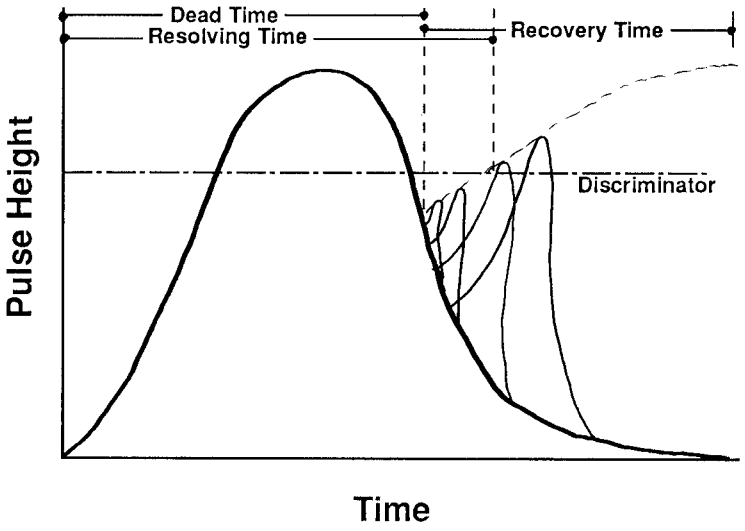


Fig. 28 - Dead time and related parameters

The recovery time is measured from the point on the tail of the pulse when a second tiny pulse is just distinguishable as arriving at the end of the dead time. It is measured out to the time when a second detected ray produces a full amplitude pulse. The resolving time is the minimum time that elapses from the moment of detection of a first ray until the electronics connected to the tube are able to count a second ray. It is longer than the dead time because the electronics package always includes a pulse height discriminator set higher than the background noise pulses. Thus, the resolving time includes enough time beyond the dead time for the second, partially formed pulse to grow big enough to trip the electronics into recording a second event.

Saturation is a problem in Geiger counters. It is related to dead time and refers to the behavior of some GM survey instruments when exposed to a very high exposure rate. A conventional instrument will show a momentary upswing of the meter needle followed by a return of the needle to a point near zero, even though the instrument is still in the high radiation field. In such a high field, the ionizing events are interacting with the counter tube at an average separation in time much closer together than the counter dead time. Most of these rays will be missed since the tube is "dead." The problem occurs near the end of the dead time while the last ions are being cleared from inside the tube. If a new event is detected during the dead time, the tube still has not fully recovered so the gas multiplication factor will still be depressed. This produces a much smaller pulse than normal. In fact, the pulses formed under these conditions are usually so small as to be at the same level as the background electronic noise. Since the noise pulses are discriminated against by the electronic circuit, this real count will be missed along with all the following counts that continue to trigger the tube before it can recover. Thus, the instrument reads "background" while in fact the operator is in an extremely hazardous radiation field. This problem can be eliminated by using only the "non-saturating" type of Geiger counters now commercially available. If in doubt, check the instrument specifications to make sure it will not saturate in fields which might be possible at your facility, even under worst case accident conditions.

A final property of interest in the GM counter is its energy dependence. It does not produce the same pulse output rate when exposed to the same exposure rate produced by gamma rays of different energies. Figure 29 illustrates a typical response. At low energies, the tube overresponds, indicating a higher exposure rate than is actually present. The reason for this behavior is the physics of the interactions (review Chapter 3 if necessary). At low energies, the gamma rays undergo a photoelectric interaction while at medium energies (a few MeV) the interactions are Compton scattering. The Geiger counter is calibrated to read milliroentgens per hour and the roentgen is defined only for air as the absorber. But the Geiger tube is constructed of aluminum, steel or glass. Every one of these materials has a higher atomic number, Z, than air. Therefore, for low energy photons, the tube will be more efficient than air in stopping gamma rays because the probability per gram for the photoelectric effect is

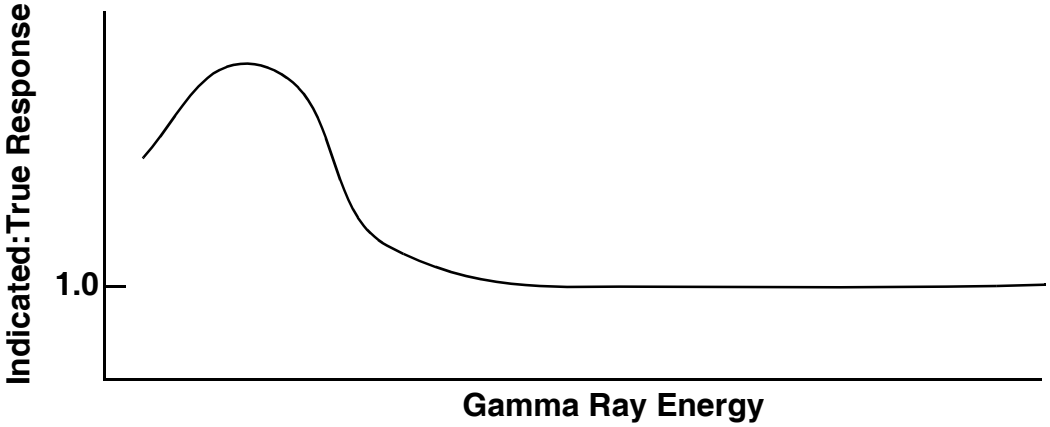


Fig. 29 - Energy dependence in a Geiger tube

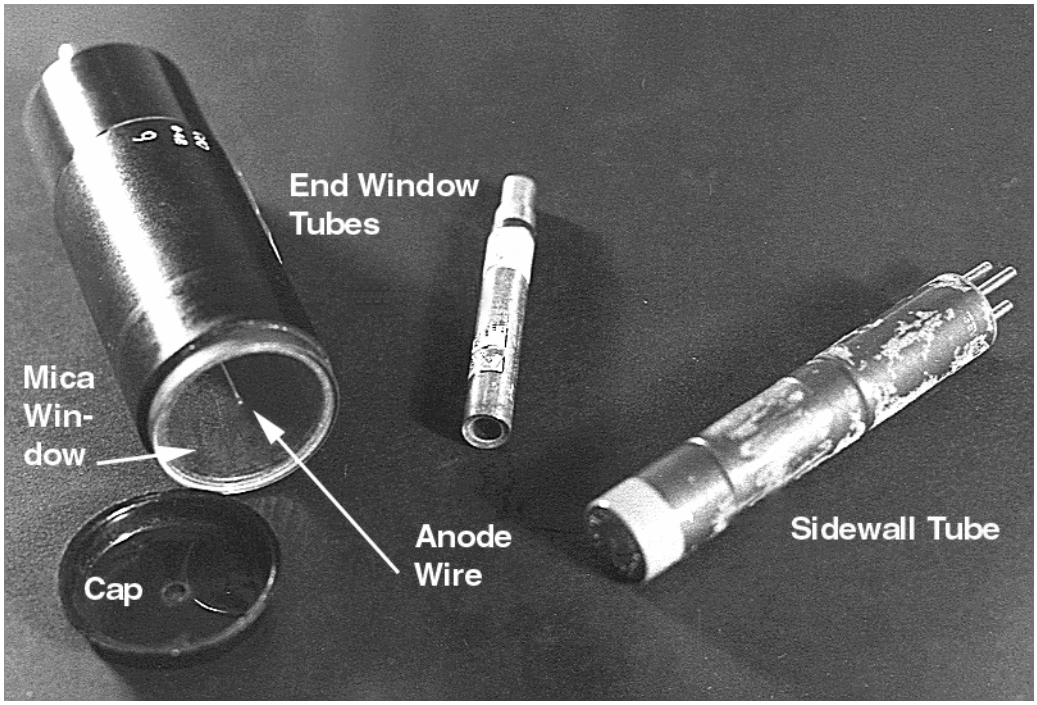


Fig. 30 - Some commercial examples of Geiger counter tubes

proportional to Z^3. The tube will read high. At medium energies, the tube will read correctly since the probability of Compton interactions is independent of Z. Geiger counters used to measure low energy X-ray fields or radioisotopes emitting photons below about 150 keV (for example, Am-241) will need to be calibrated specifically at those energies.

"Tissue Equivalent" detectors are slowly beginning to appear in the radiation protection marketplace. This means that the detector responds the same as human soft tissue would if placed at the same point in the radiation field. Recalling that tissue, water and air have essentially the same effective atomic number, then, it should make sense that any detector which can read photon exposure rates correctly in mR/hr, over a wide range in photon energies, will be tissue equivalent. Generally, since detectors are constructed of materials with Z higher than tissue, correction factors must be applied at high and low energies and, consequently, the detector is not tissue equivalent. Special purpose detectors are being constructed out of plastics with an effective Z of 7.5 and filled with a gas mixture of the same effective Z so they will be tissue equivalent.

Figure 30 shows some of the various commercially available types of tubes. The side wall GM tube is made to detect photons through the side. It is normally housed inside a metal probe with a "rotating beta shield." If the shield is closed, the gamma ray energy response is much flatter due to the filtering effect of the metal housing. The end window GM tubes are made primarily for use with alpha and beta particles. The thin window is made of mica or conducting mylar plastic with a density thickness of about 1 to 2 mg/square cm.

Figure 31 shows the popular 2-inch pancake counter. These probes are useful for alpha and beta contamination surveys (frisking). Also shown is a subminiature sidewall tube for pocket GM instruments. Its cathode diameter is only 5 mm.

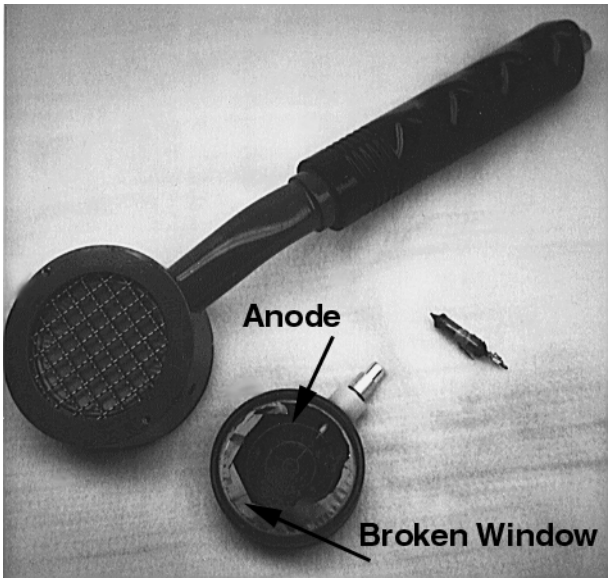


Fig. 31 - Pancake and sub-miniature geiger tubes

It is of historical interest to note that the "Geiger-Mueller Counter" was actually a joint invention, in 1908, by Ernest Rutherford and Hans Geiger. The original design was published in the Proceedings of the Royal Society. Geiger was a laboratory assistant working under Rutherford at the University of Manchester in England.

Liquid Radiation Detectors

Liquid Scintillation Counters

Although most of the instruments used in radiation protection employ a gas-filled or solid state detector, there is one instrument that is common in the field that has a liquid for the counting medium. The liquid scintillation counter, (LSC), is most often resorted to for measuring low energy beta emitters on wipe and leak test samples. In the case of tritium, the lowest known energy beta emitter, there is virtually no other type of detector that has the necessary sensitivity.

The LSC instrument has a very high counting efficiency due to the intimate mixing of the radioactive atoms being counted with the detector atoms in liquid form, the so-called scintillation cocktail. This solution is made up of two components – the solvent and the scintillating solute. The job of the solvent is to dissolve both the source radioactivity and the scintillating solute. Common solvents include toluene, dioxane or one of the new "third generation" biodegradable, non-toxic solutions. The

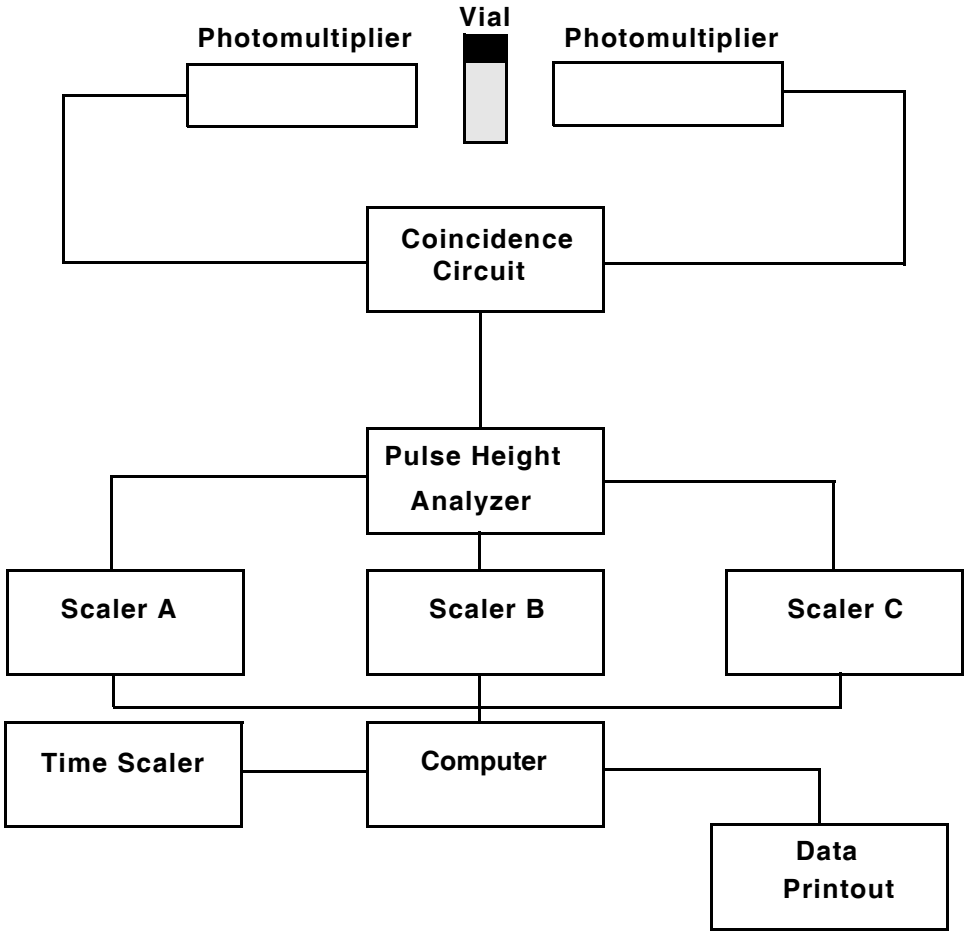


Fig. 32 - Block diagram of a liquid scintillation counter

source might be an organic chemical used in research studies, animal tissue or, in the case of radiation protection uses, a liquid urine sample or wipe test medium.

The solute absorbs the decay energy from the solvent and re-emits the energy as light. Often a secondary solute is added which shifts the wavelength of the emitted light to a more desirable wavelength in terms of the sensitivity of the photomultiplier tubes used. The intensity of the light flash produced following absorption of radiation energy is directly proportional to the energy deposited in the cocktail. Thus, beta energies (and, for the record, alpha energies) can be measured and some energy discrimination is possible. (Remember that beta emitters release a whole range of energies from zero up to the E_{max} value.)

Quenching is the main problem to be dealt with in a liquid scintillation system. This term is applied to any process which reduces the light output that would normally be expected. There are two general sub-classes – chemical and optical quenching. Frequently the sample being counted contains atoms which trap some of the emitted energy and release it as thermal energy rather than light. This is called chemical quenching. Optical quenching (or color quenching) involves absorption of some of the light before it leaves the solution. Again, many samples, such as urine, contain

molecules which strongly absorb certain wavelengths of light, thus, degrading the signal. The quenching problem is dealt with by measuring the amount of quenching and applying a "quench correction" to the counter results. Many commercial instruments can perform these corrections automatically.

The electronics is relatively simple to understand. As illustrated in Figure 32, the counter actually contains two photomultipliers aimed at the transparent scintillation vial holding the cocktail. This arrangement greatly reduces the background count. Random counts caused by thermionic noise pulses in the photocathodes of the two PM tubes are rejected by the "coincidence circuit." This electronic unit allows the pulse to pass through only if it receives a pulse simultaneously from both tubes. Light flashes from actual radiation interactions in the cocktail meet this requirement of occurring coincidentally in time. They are thus passed through to the pulse height analyzer which sorts the pulses according to their amplitude (i.e., their energy) and routes the different energy pulses to the correct scaler where a count is recorded. Typically the liquid scintillation counter has three energy channels. The lowest is usually set for tritium, the center one for ^{14}C or ^{35}S and the highest energy channel for ^{32}P. A timer controls the duration of the count for the sample and the computer handles data analysis such as quench correction and background subtraction. Figure 33 is a photo of a complete commercially available counter.

Some manufacturers now offer liquid scintillation systems which separate alpha counts from beta counts. For example, several Packard TriCarb models employ "Time Resolved Pulse Decay Analysis" that automatically applies a sophisticated electronic analysis of the trailing edges of the decaying pulses. The TriCarb model claims a crossover rejection of up to 99.5%. This means that when an alpha emitter falls at the same energy as a beta emitter, as little as 0.5% of the alpha counts are mistaken by the instrument as being beta counts.

Courtesy, Beckman Coulter, Inc.

Fig. 33 - A commercial liquid scintillation counter

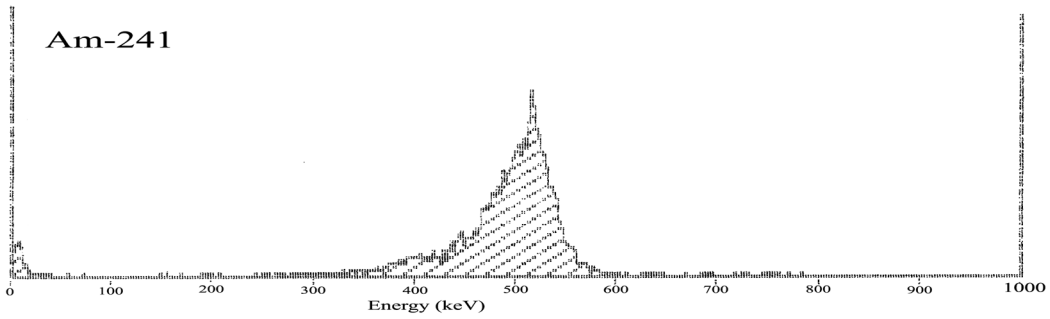
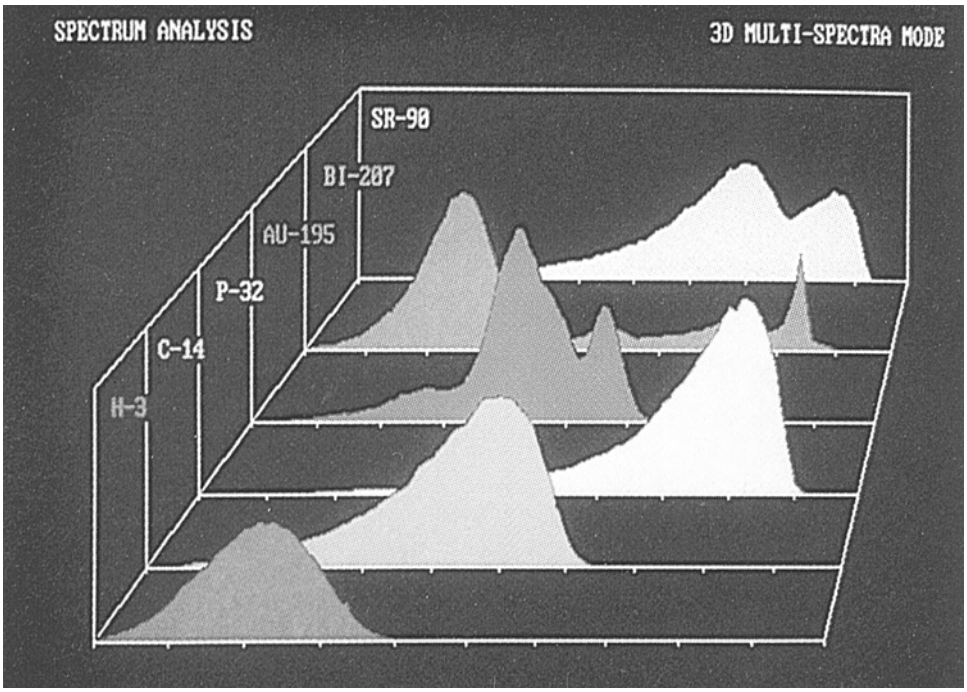


Fig. 34 - Am-241 alpha liquid scintillation spectrum

This is important because alpha particles interact differently than betas in the liquid scintillation cocktail. About 90% of the alpha energy goes to ionizing the cocktail molecules (ionization quench) and does not contribute to light output. Thus, a "typical" 5 MeV alpha particle will produce the light output of a 500 keV beta (10% of 5 MeV = 0.5 MeV or 500 keV). Without the decay time discriminator, the alpha pulses fall into the same energy channel as numerous common beta emitters.

Another unique benefit of LSC counting of alpha emitters is that, due to the nature of the interaction, alphas are counted with almost 100% efficiency in a liquid scintillator. Furthermore, the efficiency is basically unaffected by both chemical and color quenching. What happens instead is that the alpha spectrum peak gets moved to lower energy and peak width is broadened, but the total counts in the alpha peak are unchanged. Note, also, that since alpha particles are emitted without the neutrino that accompanies beta decay, a single alpha emitter produces a well resolved peak in a LSC spectrum rather than the broad distribution from zero to E_{max} characteristic of beta emitters. See Figure 34. If a technologist is faced with the problem of counting many wipe samples with low level mixed alpha/beta activity, the ability to simultaneously count the two activities with high efficiency and negligible sample preparation makes the LSC system an attractive alternative to conventional alpha/beta counters.

Most modern liquid scintillation counters have the ability to display energy spectrum information. For example, one commercial unit can view, on a separate PC, up to six different beta energy spectra simultaneously (see Figure 35). This is particularly useful in health physics applications to identify unknown beta emitters. This is often the case on wipe test samples taken in laboratories that are authorized to use more than one different beta emitting radioisotope or in counting urine samples from workers handling multiple radioisotopes. The maximum energy and shape of the spectrum determines the identity of the beta emitter.



Courtesy, Beckman Coulter, Inc.

Fig. 35 - Beta ray energy spectra from a liquid scintillation counter

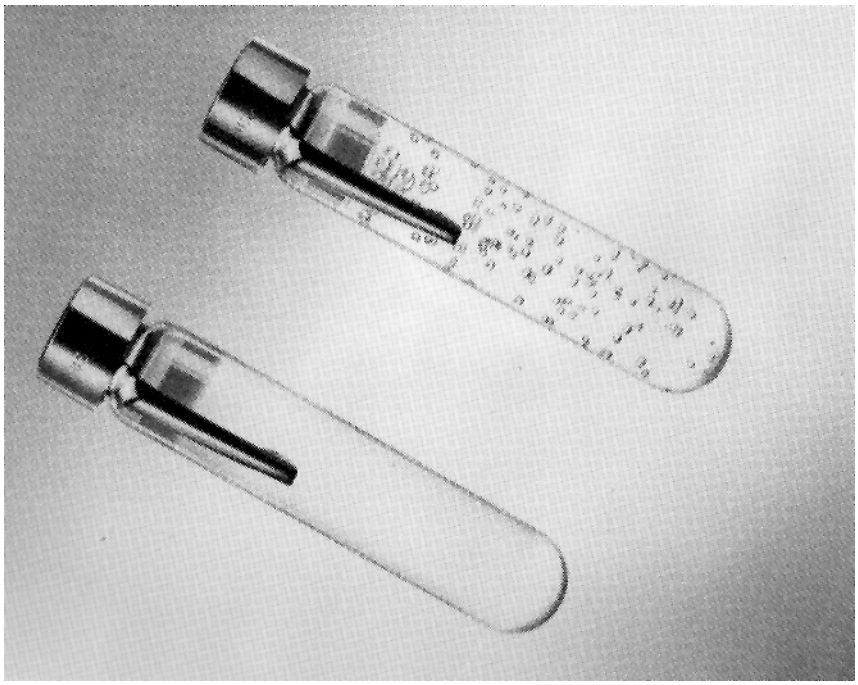
Superheated Drop Detectors

While perhaps not strictly qualifying as a liquid, the superheated drop detector, SDD, comes close in that the active detector is supported in a liquefied gel medium. Invented by Richard Apfel of Yale University in 1979, a decade or so of development resulted in a practical neutron detector. SDDs are particularly useful in the difficult mixed gamma-neutron field as the material is gamma insensitive. The dosimeter is composed of a small container of gel holding the superheated drops, each about 0.15 mm in diameter, in suspension. Proton recoil tracks caused by exposure to neutrons trigger the superheated drops to vaporize, producing an audible sound pulse which can be recorded. Alternatively, the amount of vapor in the gel can be determined (by counting the bubbles) to measure neutron dose. Figure 36 compares an unexposed and an exposed vial, demonstrating the visibility of the vaporized drops. These detectors show a flat energy response from 200 keV to 14 MeV. The manufacturer claims a sensitivity of 1 microsievert.

Solid Radiation Detectors

Scintillation Counters

The first of the two broad types of detectors using a solid medium rather than a gas as the sensitive volume is the solid scintillation counter. Many different crystals

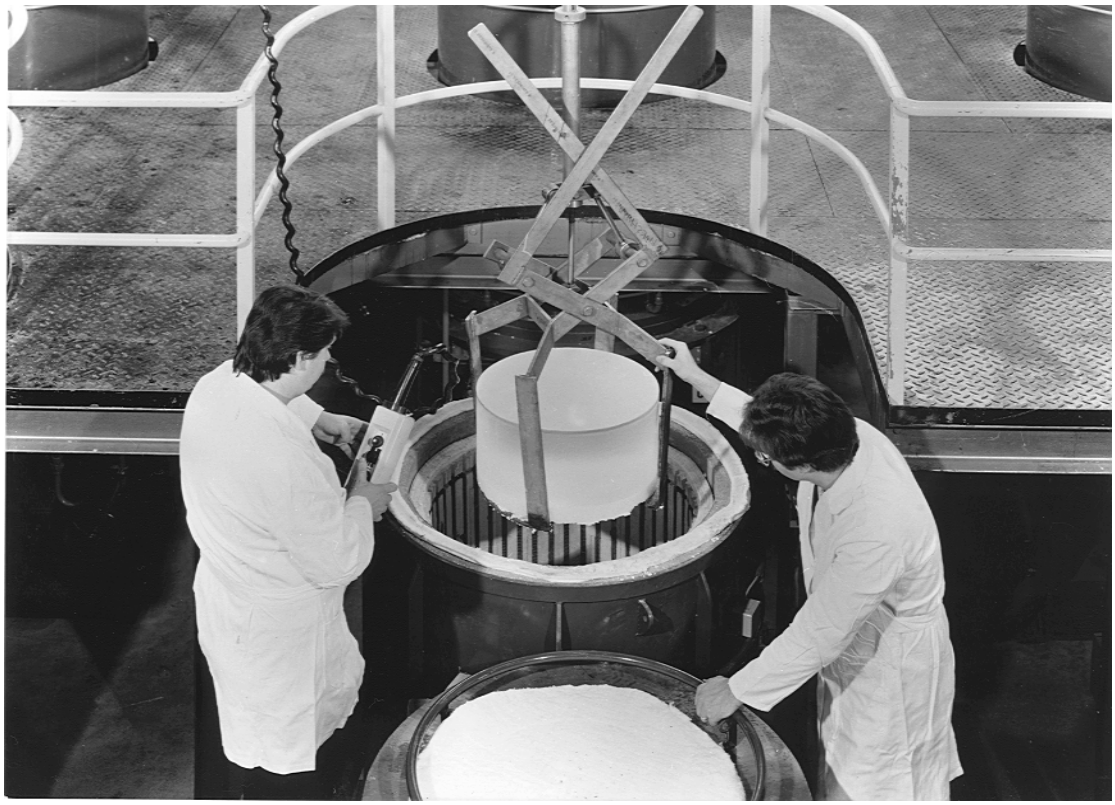


Courtesy of Siemens Dosimetry Services

Fig. 36 - The superheated drop neutron detector

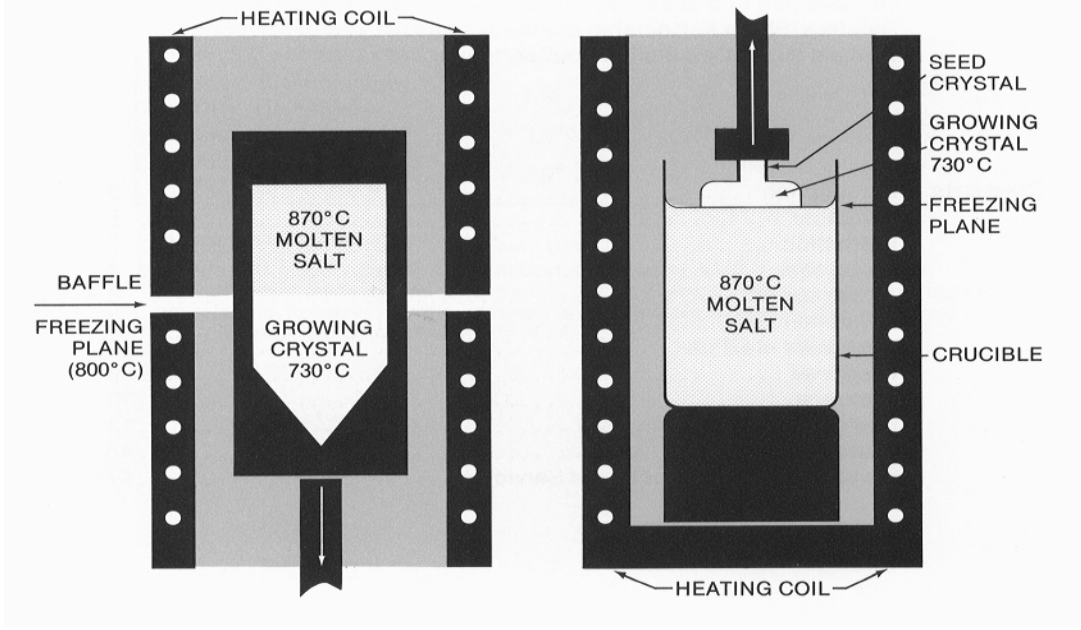
have been used as the active "phosphor" but at present, most scintillation counters use artificially grown sodium iodide with a trace amount of thallium as an "activator." The chemical symbol is written NaI(Tl). Figure 37 shows a crystal-growing oven at Harshaw/Filtrol Partnership. Figure 38 is a sketch demonstrating the two principal methods for growing crystals. The thallium (Tl) activator increases the output light intensity at room temperature. When energy is deposited in the form of photoelectrons, Compton electrons or an electron–positron pair, the phosphor converts it into a light flash with a time duration of about a quarter of a microsecond. The individual light photons are in the blue region of the optical spectrum (about 4100 angstroms or 410 nanometers wavelength). Because of the short pulse formation time, the counter can handle high counting rates. The intensity of the light flash is directly proportional to the energy deposited by the gamma ray in the crystal.

NaI(Tl) counters are used almost exclusively to record gamma rays. A major consideration is their hygroscopic nature. The crystal readily absorbs moisture out of its environment. An uncovered NaI crystal left for a week on a tabletop will "self destruct" by dissolving into a sticky puddle. This means that the crystals must always be sealed hermetically to prevent contact with humid room air. The preferred packaging technique is to use an aluminum cylinder with an aluminum cap at one end to form a close-fitting "cup." The inside is then coated with a white reflective paint to increase light output. The NaI(Tl) crystal is machined to fit snugly inside the cylinder and a glass or quartz window is then permanently sealed over the open end. The metal container effectively prevents alpha or beta particles from reaching the enclosed crystal (see Figure 39).



Courtesy of Harshaw/Filtrol Partnership

Fig. 37 - NaI(Tl) ingot being removed from a growing oven



Courtesy of Harshaw/Filtrol Partnership

Fig. 38 - The two common types of crystal growing ovens

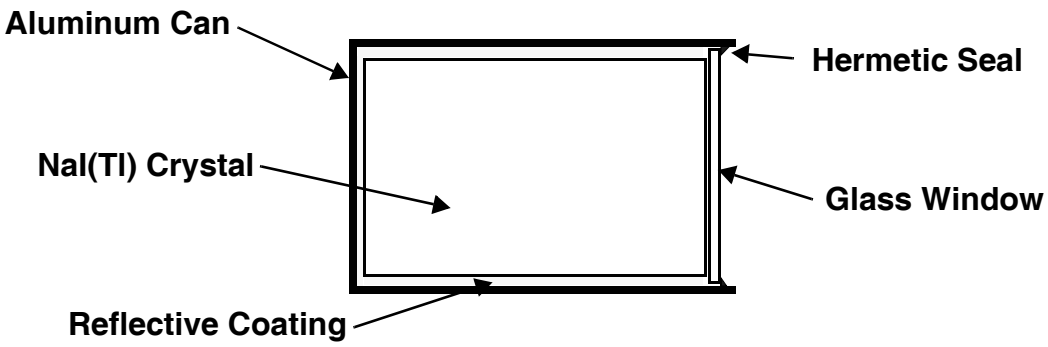
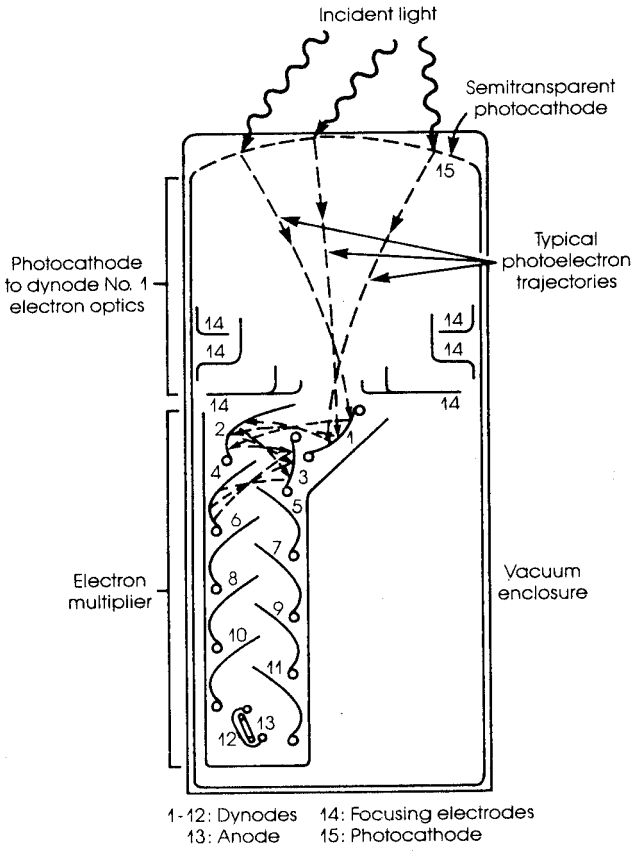


Fig. 39 - A demountable NaI(Tl) crystal assembly

To construct a complete scintillation detector, the sealed crystal is cemented to the entrance window of a photomultiplier tube (PM tube), an electronic device that amplifies weak light pulses into a large electrical signal. The operation of the PM tube is clarified by the sketch of Figure 40. The inside surface of the entrance window is coated with a material which releases electrons when struck by light photons, thus becoming a photocathode. These electrons are attracted to metal elements called dynodes by an externally applied potential difference. When they collide with the dynode,



Courtesy of Harshaw/Filtrol Partnership

Fig. 40 - Basic elements of a photomultiplier tube

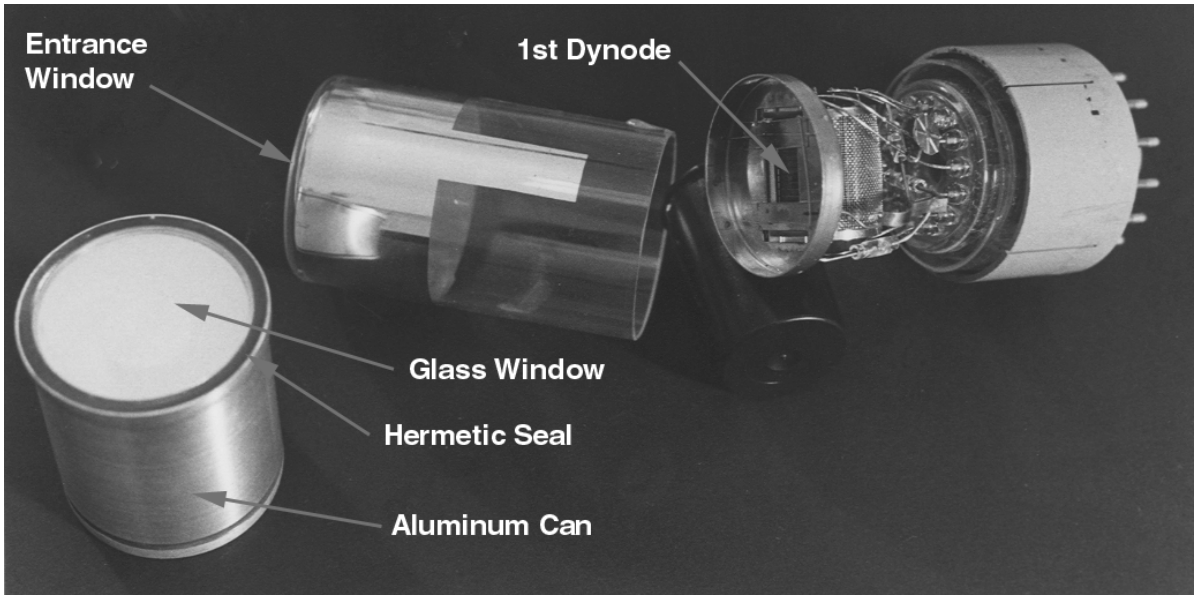


Fig. 41 - A NaI(Tl) crystal and sawed open photomultiplier tube

they knock off an average of three electrons each. Thus, at each "stage," the total number of electrons gets multiplied by three. Actual PM tubes usually have from 10 to 14 dynodes, so an overall gain (amplification factor) over 1 million is possible (Sample Problem 5). The anode cup collects the final swarm of electrons and produces a voltage pulse whose height is proportional to the intensity of the light flash at the photocathode and, hence, the energy deposited by the gamma ray. A photograph of a disassembled scintillation counter with the light-tight housing removed is shown in Figure 41. The PM tube has been cut open to better show the dynodes. These days it is common to purchase a scintillation counter already pre-assembled inside a permanently sealed aluminum envelope. Such counters are called integral assemblies in contrast to the demountable type shown in the previous photo. Some scintillators of this type are illustrated in Figure 42. The "hole thru" design improves counter efficiency by almost completely surrounding the sample with the detector.

A major advantage of the scintillation counter over a gas-filled detector is the much higher counting efficiency for gamma rays. This is due to two factors: density

Sample Problem 5

GIVEN:

A photomultiplier tube has 14 dynode stages.

FIND:

What is the theoretical electron multiplication possible in this tube?

SOLUTION:

The gain per stage, from the text, is three. So, the overall gain achieved by "stacking" 14 stages together is 3 raised to the 14th power. Finally, the

multiplication = 3^{14} = 4.78×10^6, i.e., well over one million!

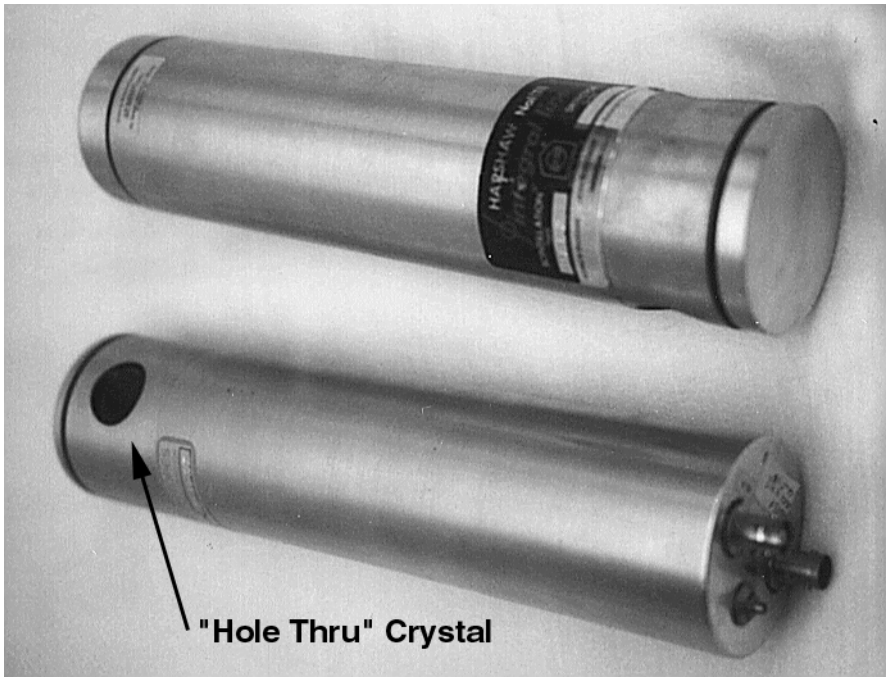


Fig. 42 - Examples of integral assembly scintillation counters

and atomic number. Sodium iodide has a density of 3.7 grams/cubic cm while gases have densities in the range of 0.001 g/cubic cm. Thus, comparing NaI(Tl) to a gas detector, there are several thousand more atoms per unit volume in the path of a gamma ray which greatly increases the chance of an interaction. Secondly, the iodine component (Z = 53) produces an overall higher average atomic number (49.7) than a typical filling gas such as argon (Z = 18). This increases the low energy photoelectric cross section significantly due to its Z cubed dependence. Figure 43 is a table listing some of the physical properties of a variety of commercially available scintillators.

Phosphor	Pulse (μsec)	Density	Hygroscopic?	Zeff	Rel. Efficiency
NaI(Tl)	0.23	3.67	Yes	49.7	100%
CsI(Na)	0.63	4.51	Yes	54.0	85%
CsI(Tl)	1.0	4.51	No	54.0	45%
$CdWO_4$	5.0	7.90	No	61.0	40%
^{6}LiI(Eu)	1.4	4.08	Yes	52.0	35%
$LaBr_3$(Ce)	0.016	5.20	Yes	45.2	165%
Plastic	0.002 to 0.02	1.06	No	Varies	30%

Fig. 43 - Physical properties of some scintillators

Cerium activated lanthanum bromide ($LaBr_3(Ce)$) is a fairly new scintillator. It was discovered in 2001, and, as of 2011, is available from vendors including St.-Gobain Crystals & Detectors in Newbury, OH and ORTEC in Oak Ridge, TN. It has several properties which are making it very competitive when compared to NaI(Tl). The chief disadvantage is cost. At presstime, a 2" X 2" LaBr3(Ce) costs about $31,000 vs. only about $1,200 for sodium iodide for complete detectors. A 3" X 3" is $60,000!

The chief advantage of $LaBr_3$ over NaI is in energy resolution. This parameter measures the narrowness of the detected spectral peaks for gamma rays and will be discussed fully later in this section. The narrower the peaks, the easier to identify the "fingerprint" of specific gamma emitting radioisotopes. $LaBr_3$ has a resolution of 2.9% for 662 keV gammas compared to 7% for the comparable NaI scintillator. The lanthanum detector also can count at higher count rates (higher activity sources) and has a higher temperature stability. Finally, this detector has a relatively high effective atomic number of 45.2 and high density (5.2) which again make for a high gamma ray detection efficiency.

Figure 44 shows how μ, the linear attenuation coefficient, varies for different photon energies in sodium iodide crystals. The efficiency of a given thickness of sodium iodide crystal can be estimated from the attenuation coefficient as discussed in Chapter 3. Sample Problem 6 illustrates the calculations.

The "steepness" of the μ versus energy curve in the low energy (photoelectric) region in Figure 44 gives us insight into some specialized scintillator applications. As shown by Sample Problem 6, low energy gammas are counted with high efficiency in small thicknesses of NaI(Tl). Conversely, to achieve comparable high efficiency for a higher energy gamma such as the 1.25 MeV average for Co-60, it can be calculated that a crystal thickness of 15 cm (6 inches) would be needed. This is the reasoning behind the use of thin, large diameter NaI(Tl) crystals in certain applications. For nuclear medicine studies with Tc-99m, for example, the gamma energy is 141 keV, about the same as the Co-57 used in Sample Problem 6. Thus, a crystal of 1/2" thick by 14 inch diameter is used to image human organs. The thickness assures almost total probability of gamma ray capture, but the 1/2" thickness also means that the crystal is unlikely to stop many high energy gamma rays commonly found in the ambient background. This produces a high "signal to noise" (background) ratio and improves the readability of the images. Similarly, a popular radiopharmaceutical label is the nuclide I-125. The main photon emissions are at 27

Sample Problem 6

GIVEN:

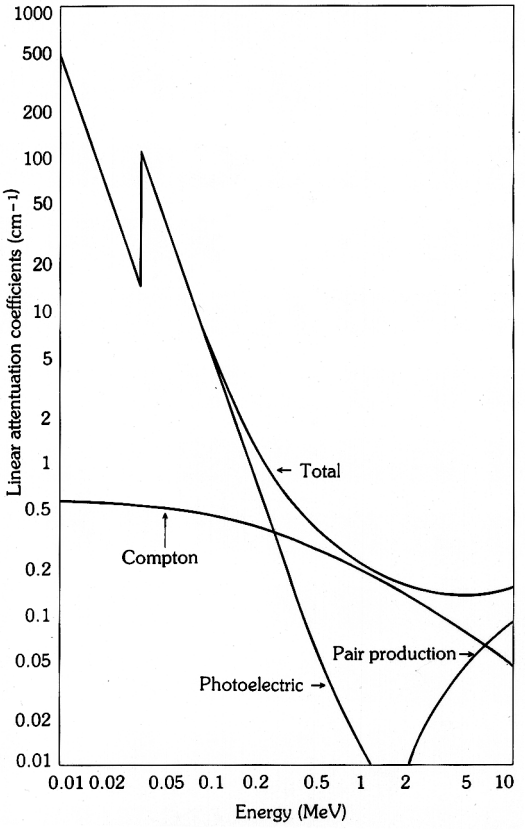
A scintillation crystal in a counter is 1 cm thick.

FIND:

What counting efficiency would be expected for the dominant Co-57 gamma?

SOLUTION:

From Appendix A-1, Co-57 emits a 0.122 MeV gamma. At this energy, from Fig. 44, μ = about 3/cm. Since each interaction produces a "count", the efficiency is just the fraction interacting, or, Eff.(%) = 100% – $e^{-\mu x}$ = 100% – e^{-3} = (100 – 5)% = 95%.



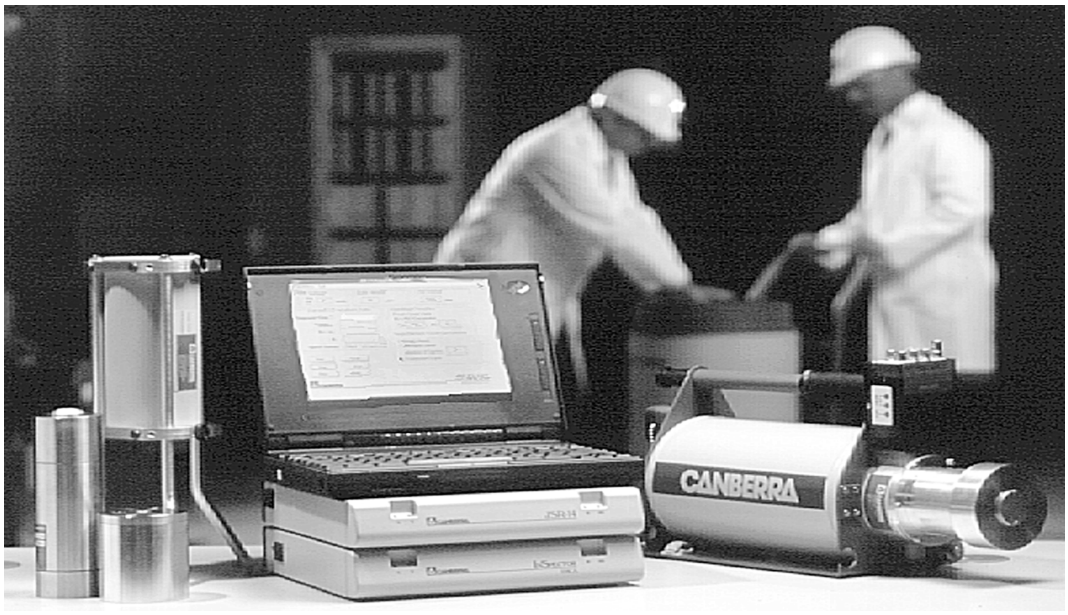
Courtesy of Harshaw/Filtrol Partnership

Fig. 44 - Variation of μ in NaI(Tl) with energy

and 35 keV. A number of manufacturers make I-125 probes with a one millimeter thick by 2 inch diameter NaI(Tl) scintillation crystal for radiation protection applications. Again, the probe has a high counter efficiency (some 94% absorption for the 35 keV gamma) but an extremely low background rate as higher energy gammas pass right through the 1 mm crystal without interacting.

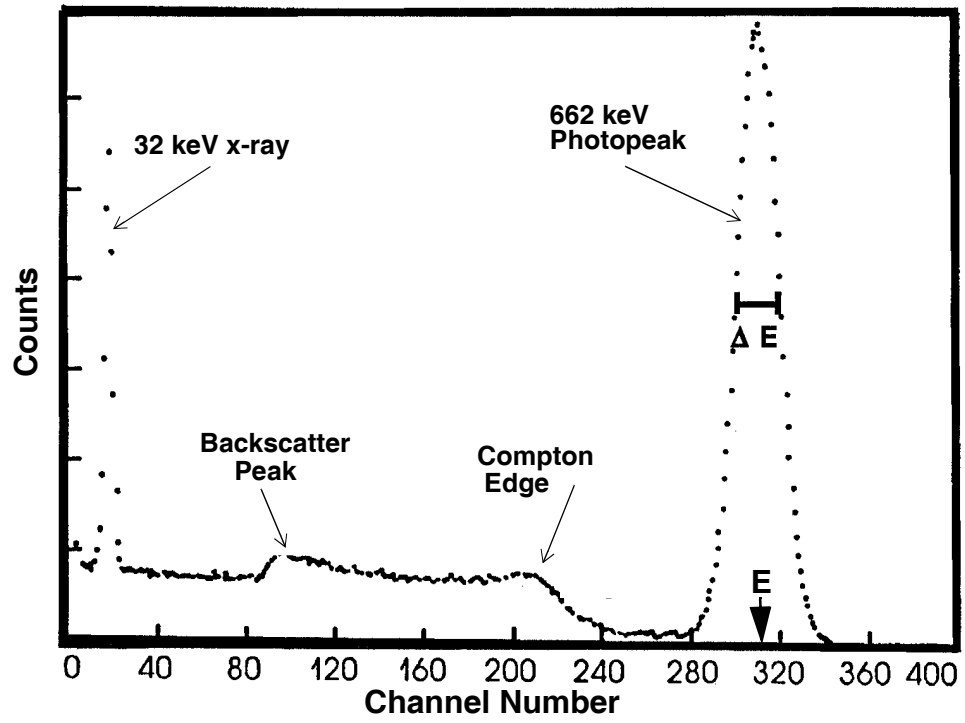
A final application of this principle is the FIDLER (Field Instrument for Detection of Low Energy Radiation). It consisted of a thin crystal survey meter with two single channel analyzers. One was set to detect plutonium and the other measured the impurity americium. It surveyed ground contamination following accidents involving rupture of nuclear weapons, for example, resulting from a military aircraft crash.

Since the output voltage pulse height is proportional to the gamma ray energy deposited, it is possible to use the scintillation counter for gamma ray spectroscopy. In this process, the amplitudes of the pulses from the photomultiplier tube are sorted by an instrument called a multichannel pulse height analyzer (or multichannel analyzer or MCA). Figure 45 is a photo of a commercially available MCA. The stored information on the numbers of pulses at each amplitude is then displayed on an computer screen or printed out to give the gamma ray pulse height spectrum. The features of such a spectrum for a gamma ray source emitting only a single energy gamma ray is



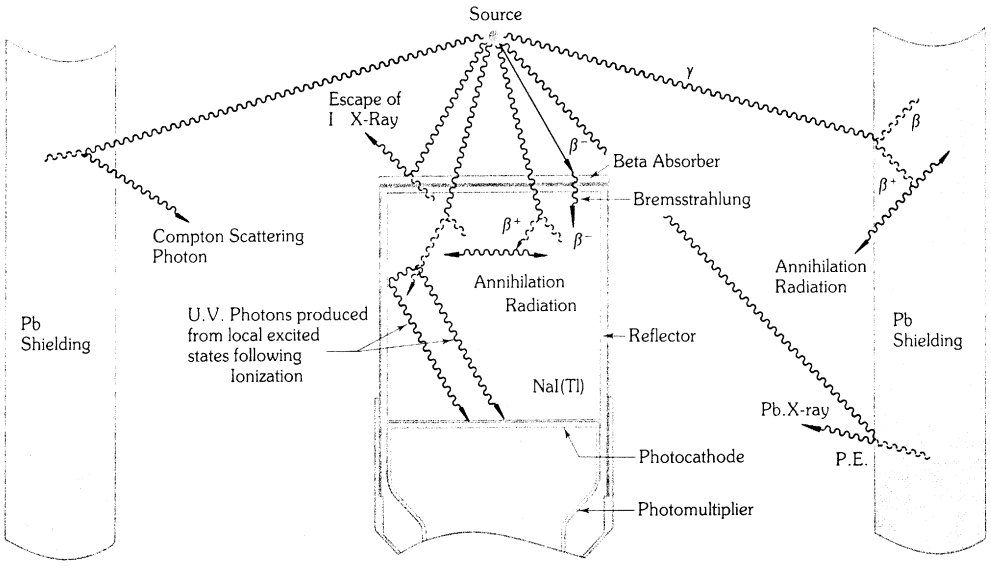
Courtesy of Canberra Industries, Inc.

Fig. 45 - A commercial multichannel pulse height analyzer



Courtesy of Harshaw/Filtrol Partnership

Fig. 46 - A NaI(Tl) scintillation counter pulse height spectrum



Courtesy of Harshaw/Filtrol Partnership

Fig. 47 - Physical processes in a scintillation detector

shown in Figure 46. The energy of the gamma ray is determined by the horizontal location of the photo peak (also sometimes called the total absorption peak), the identified tall spike in the figure. The name is due to the photoelectric interactions which produce this peak. The scale factor or energy calibration is obtained experimentally by measuring spectra for sources with known energies.

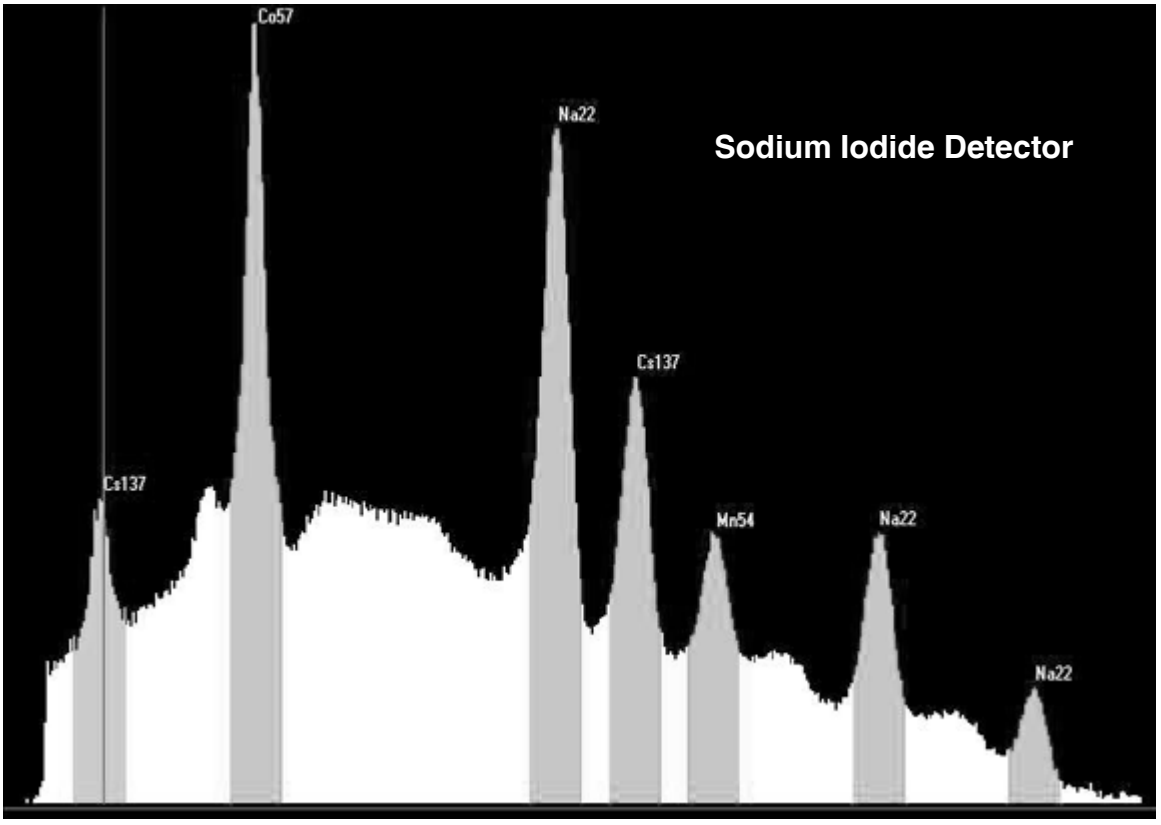
It is instructive to compare this gamma energy spectrum of Figure 46 with the beta energy spectra of Figure 35. The gamma spectrum is potentially much more useful because, as stressed in Chapter 2, beta emitters release particles with a range of energies, from zero up to E-maximum. So, beta spectra show a broad peak corresponding to a range of energies. In contrast, gamma emitters release discrete, unique energy rays. By observing the energy locations of the photopeaks in a gamma spectrum, it is possible to identify a large number of different gamma emitters with a single measurement.

Some of the many physical processes that contribute to the pulse height spectrum are illustrated in Figure 47. The "backscatter peak" is a result of capture in the crystal of photons which have Compton-scattered from shielding or other objects near the detector. Since they have lost some energy already, they show up with energies less than the photo peak (full energy peak).

The "energy resolution" of the scintillation counter is also illustrated by figure 46. This is defined as the width of the peak at half-amplitude (Δ E in Fig. 46) divided by the energy, E, times 100%. For sodium iodide, the energy resolution is typically 7% to 9% at an energy of 662 keV (Cs-137 gamma ray). At lower energies, the energy resolution is better (smaller %) and at higher energies it is worse (larger %). This is an important parameter for a gamma ray spectrometer because it determines how close in energy two gamma rays can be before the instrument gives a single broad peak

instead of two distinctly separate peaks. The energy resolution is very important when unknown gamma emitters are being recorded since photons differing in energy by less than the energy resolution will not be separately detectable. Unfortunately, the energy resolution of NaI(Tl) is not adequate for many applications in radiation protection. (A semiconductor instrument will be discussed in the next section of this chapter which does have adequate gamma energy resolution). The $LaBr_3$(Ce) scintillator mentioned earlier has a resolution of around 3%, much better than sodium iodide.

A breakthrough solution to dealing with the variable energy resolution in NaI(Tl) scintillators was patented in 1997 by Dr. William Hardy of Princeton Gamma-Tech, Inc. They named it Quadratic Compression Conversion™. The signal processing electronics are able to create multichannel analyzer channels which vary in width exactly in proportion to sodium iodide's energy resolution. Therefore, low, medium and high energy photo peaks are all displayed with the same peak width! This greatly increases the ratio of the peak height to the background for high energy peaks, making them more readily visible in the spectrum. In addition, low energy peaks are spread out wider so they can be more easily resolved. Figure 48 illustrates this.



Courtesy of Princeton Gamma-Tech, Inc.

Fig. 48 - Illustration of the Quadratic Compression Conversion™ process

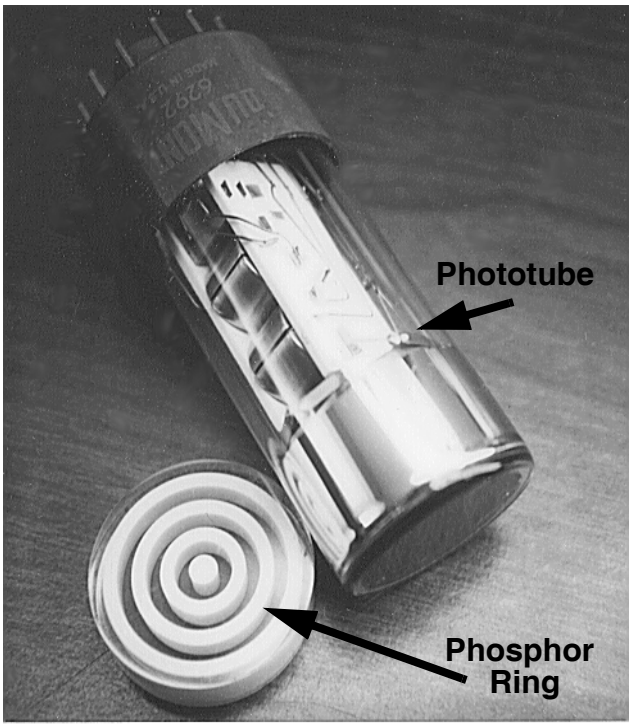


Fig. 49 - A plastic scintillator for fast neutron detection

Another use of the scintillation principle is for neutron detection. Thermal neutrons are detected by using lithium or boron to produce an alpha particle and then detecting the alpha with a scintillator. The most common phosphor used as an alpha scintillator is silver activated zinc sulphide, ZnS(Ag). Fast neutrons can be detected by counting the protons knocked out of plastic by elastic scattering. Figure 49 is a photo of a fast neutron scintillator. The clear regions are the plastic which is the source of hydrogen (protons) and the opaque rings are ZnS(Ag) phosphor which scintillates under proton bombardment. The disk is cemented to the entrance window of a PM tube and sealed with a light-tight housing to make a useful detector.

The Phoswich is a special type of scintillation detector used under certain conditions. It consists of two different scintillation crystals bonded together on the same photomultiplier tube. One of the crystals is very thin and the other thick. This "phosphor sandwich" allows one type of radiation to be counted in the presence of another type. The usual application is counting low energy photons, such as the X-rays from plutonium, in the presence of higher energy background gamma rays. It is able to perform this clever task due to the fact that the two scintillation phosphors have <u>different</u> pulse decay times and can thus be sorted out with a special electronic circuit called a pulse shape analyzer. This gadget separates out the pulses from the two scintillators and can be programmed to accept the low energy pulses from the front (thin) scintillator while simultaneously rejecting the high energy background from the thick scintillator. The

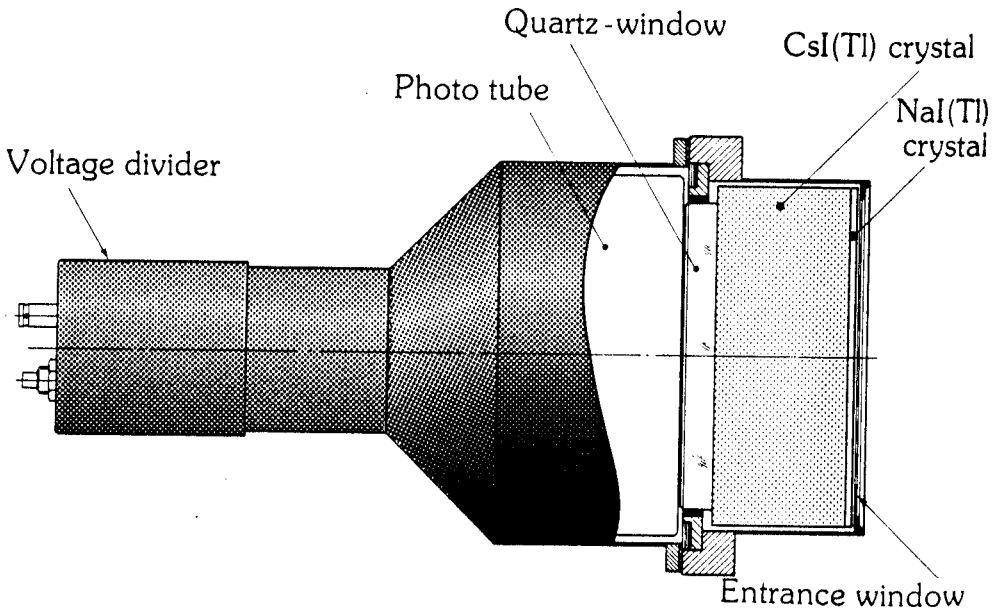


Fig. 50 - A phoswich detector

drawing in Figure 50 illustrates the principle with a thin NaI(Tl) crystal mounted over a thick CsI(Tl) crystal.

Historically, scintillation counters were used in the 1950s for a relatively unique application – prospecting for uranium ore. This required a light weight, ruggedized unit. The advantage of a scintillator over a Geiger counter was in the inherent sensitivity of a solid NaI detector. Figures 51 and 52 show a very popular 1950s era commercial instrument designed for this job, a Precision Model 111B "Scintillator." The battery pack could be attached to the prospector's belt. The analog meter could be read out in percent of uranium in an ore sample. Three time constants adapted to different travel speeds over the terrain.

One final scintillation counter application is in microR meters. These instruments are used to make quantitative readings at background levels and thus require the sensitivity of a solid state counter. Commercial microR meters usually employ NaI(Tl) detectors. While this gives good gamma sensitivity, the energy response is so poor, due to the high effective atomic number of the scintillator, that they are virtually useless for measuring actual dose equivalent rates. One neat solution is to substitute a tissue equivalent organic scintillator as is done in the Bicron Corporation MicroRem™ model described in detail in Chapter 12.

Fig. 51 - Precision "Scintillator" for uranium prospecting

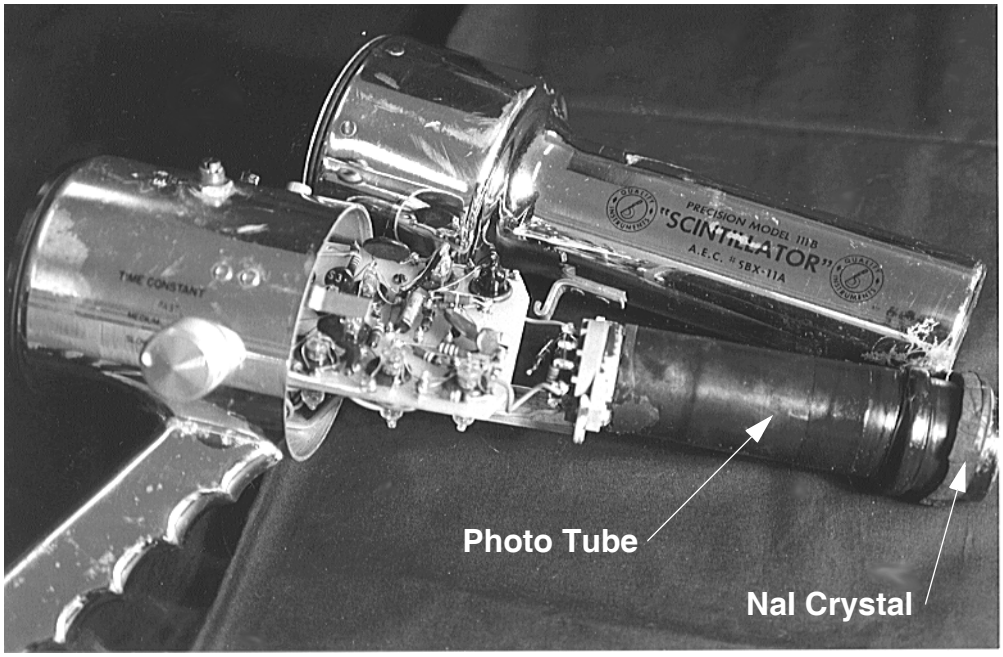


Fig. 52 - Precision "Scintillator" internal layout

Semiconductor Counters

The semiconductor counter is a recent addition to the tool kit of the radiation protection technologist. Three types have found application in the field. These are the surface barrier diode detector, the germanium detector family and the newest member, cadmium telluride (CZT).

The drawing in Figure 53 shows the basic operating principle of the surface barrier diode counter. The "P-type" semiconductor slab (in which the current is carried by positively charged holes) and the "N-type" semiconductor slab (in which the current is carried by negatively charged electrons) form the anode and cathode. By connecting the battery polarity in the arrangement shown (called reverse bias), the electrons and holes are both attracted away from the central region of the slab to leave this region "depleted" in charge carriers. This depletion region plays the same role as the filling gas in an ion chamber. The incoming ray interacts and causes ionization which produces electron–hole pairs rather than + and - ion pairs as would be the case in a material which is not a semiconductor. The surface barrier diode is useful only for particulate radiation such as alphas and betas. To measure their energy, they must expend all of the kinetic energy they carry in the depletion region. This means that the entrance window and the P-type slab must both be ultra thin, which puts the depletion region near the surface. The window is usually made by evaporating a thin film of gold on top of the P-type face of the diode.

Figure 54 shows a commercial counter. The chief advantage of this "solid state ion chamber" is that the energy resolution for alpha and beta particles is very good. This is directly traceable to the low "W value" for semiconductors. In silicon, W = 3.6 eV per electron–hole pair. This is only about 1/10 the value in non-semiconductors.

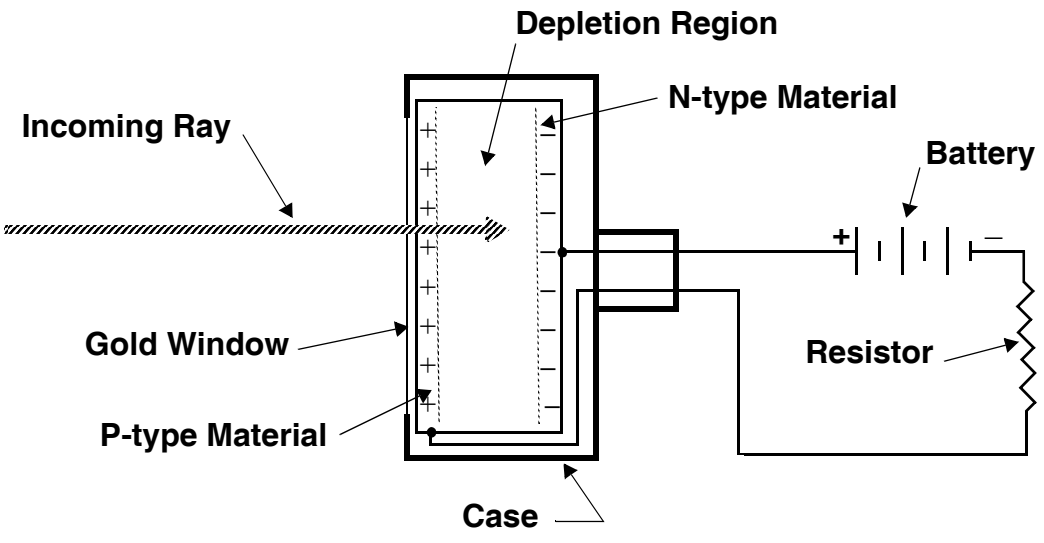


Fig. 53 - Schematic representation of a surface barrier diode counter

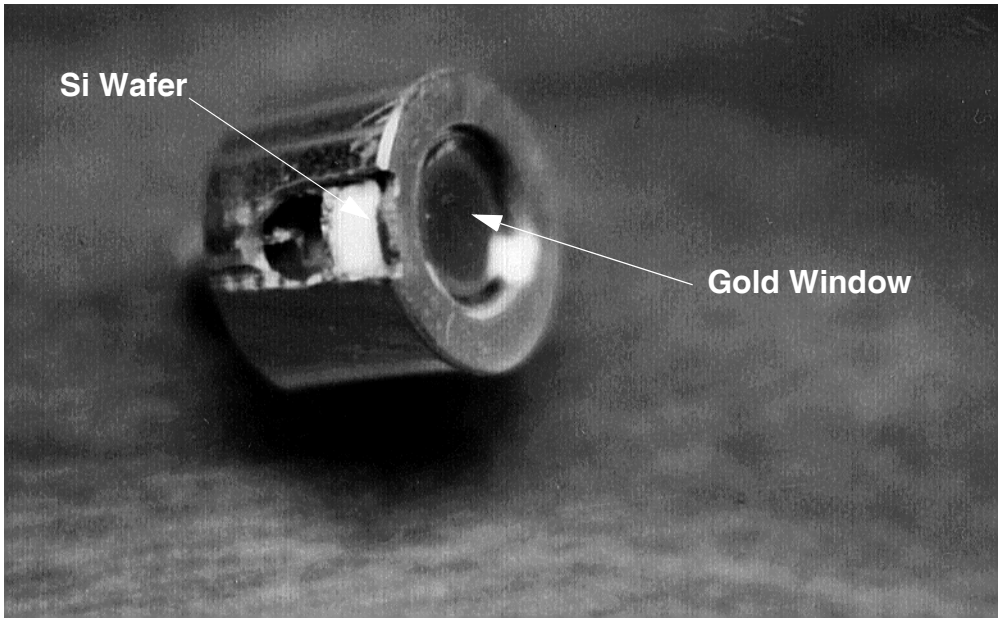


Fig. 54 - A silicon surface barrier diode counter with the side cut away

Surface barrier detectors are commonly used in modern alpha air samplers to distinguish alpha contamination from the radon background activity.

The germanium counter is another popular kind of semiconductor counter. It currently exists in two different types: the lithium drifted germanium counter [abbreviated Ge(Li)] and the high purity germanium detector (abbreviated HPGe). They are

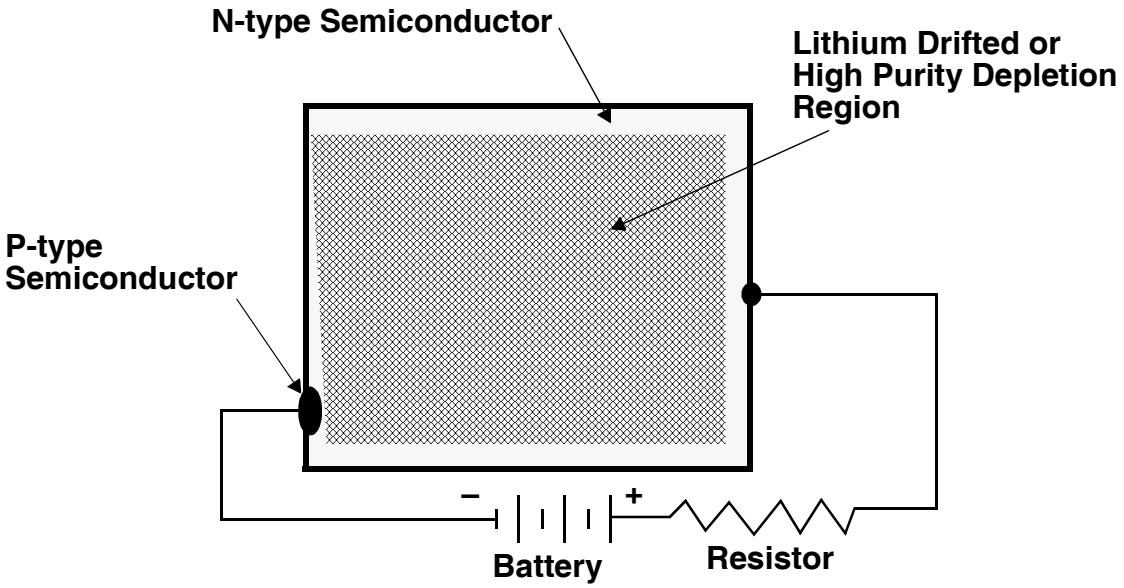


Fig. 55 - Design details of a germanium detector

both used for gamma ray spectroscopy. The operation is almost identical to the surface barrier diode described above, but the design configuration is different. Figure 55 illustrates a common cylindrical design. Most of the central volume is depleted in charge carriers through the use of either high purity germanium or by cancellation of the carriers through lithium drifting. The outside shell of N-type and the small P-type spot form the two electrical elements (cathode and anode). The gamma rays interact in the depletion region where they expend their energy in forming electron-hole pairs. These are collected by applying a relatively high potential difference, usually over 1,000 volts. The collected charge pulse of electrons and holes passes through the resistor and produces a voltage pulse (Ohm's Law). There is no multiplication of the charge produced by the photon. Since the pulse height is proportional to the energy deposited, the output signal is commonly fed to a multichannel analyzer to produce a gamma ray energy spectrum. Since the W value is so small, only 2.9 eV/electron–hole pair in germanium, much better energy resolution is obtained with this device compared to a NaI(Tl) or a $LaBr_3$(Ce) scintillation counter. A typical Ge counter will have an energy resolution of less than 2 keV for Co-60 energies (a % energy resolution of only 0.15%). For purposes of comparison, the NaI(Tl), $LaBr_3$(Ce) and Ge(Li) spectra of the two high energy peaks from Co-60 are superimposed on top of each other in Figure 56.

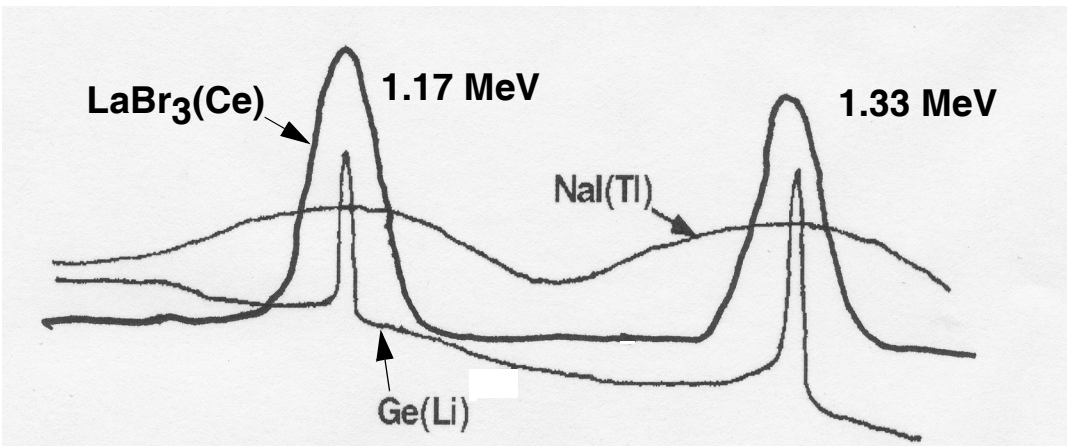


Fig. 56 - A comparison of the energy resolution of Ge(Li) vs. NaI(Tl) vs. $LaBr_3$(Ce)

A complete gamma ray energy spectrum from a germanium counter is shown in Figure 57. It illustrates the phenomenal energy resolution for this type of detector when exposed to multiple gamma ray emitters.

A comment is in order on the meaning of "high purity" germanium. In the form of single crystals, detector grade ingots of Ge are the purest substance known to mankind. The typical impurity level is one part in 10^{15}!

For many years, the difficulty in manufacturing germanium ingots of the required purity meant that the sensitive volume of the detectors was fairly small. This, in turn, meant that the counter efficiency was relatively low. A small mass can't stop as many gamma rays as a large one! The detector makers began comparing the efficiency of their germanium

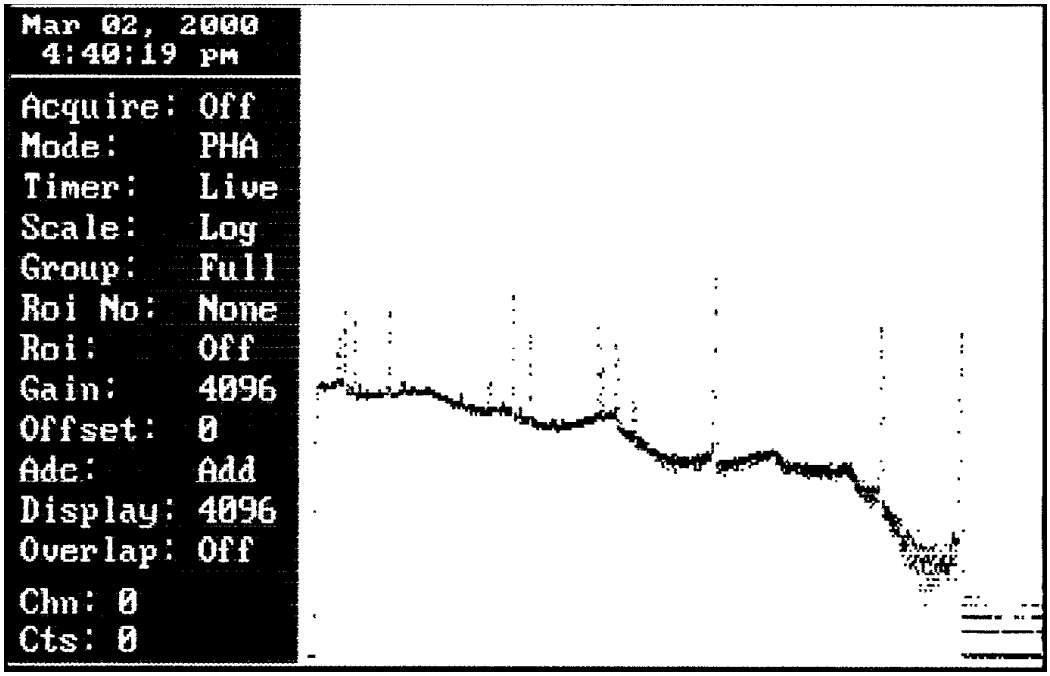


Fig. 57 - A germanium counter MCA gamma ray spectrum

counters to that of a "standard" 3" x 3" NaI(Tl) scintillation counter. This practice continues today. With time and experience, the efficiencies of germanium counters began to go up. Back in 1988, a manufacturer (Canberra) announced the first 100% efficiency counter. This germanium counter would have the same count rate as a 3" x 3" NaI crystal when exposed to the same radiation source. As of 2004, germanium counters with 200% relative efficiency were commercially available.

The primary disadvantage of germanium detectors (beyond cost, of course) is that they must be used at liquid nitrogen temperature. The detector assembly contains a vacuum dewar to hold the liquid nitrogen. In addition, the Ge(Li) lithium drifted type counters must always be cooled, even during storage or the lithium ions drift out and the detector is useless. (Over time, counter manufacturers were able to reduce the cost of the HPGe type. As of the late 1990s, Ge(Li) counters are no longer made.) The high purity Ge counters can be stored at room temperature. Figure 58 is a cut away view of a Ge(Li) detector showing the germanium ingot and the "cold finger" copper rod (cut off) that was immersed in liquid nitrogen. (This $5,000 display resulted from a technician going on vacation and not arranging for a substitute technician to refill the liquid nitrogen supply.) A complete HPGe detector and liquid nitrogen cryostat are shown in Figure 59 along with an efficiency curve.

The last of the semiconductor detectors commonly used in radiation protection is the cadmium telluride detector alloyed with a small amount of zinc (CZT). It is a small gamma and X-ray detector which operates at room temperature and does not need a photomultiplier tube. Because of a high atomic number (Z_{eff} = 50.2), the sensitivity is much higher than a germanium counter (Z = 32). CZT also exhibits high

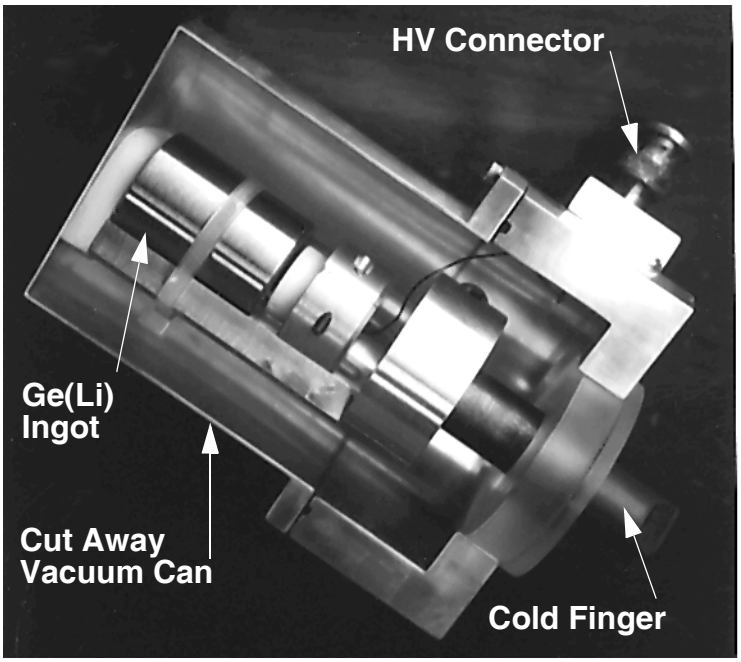
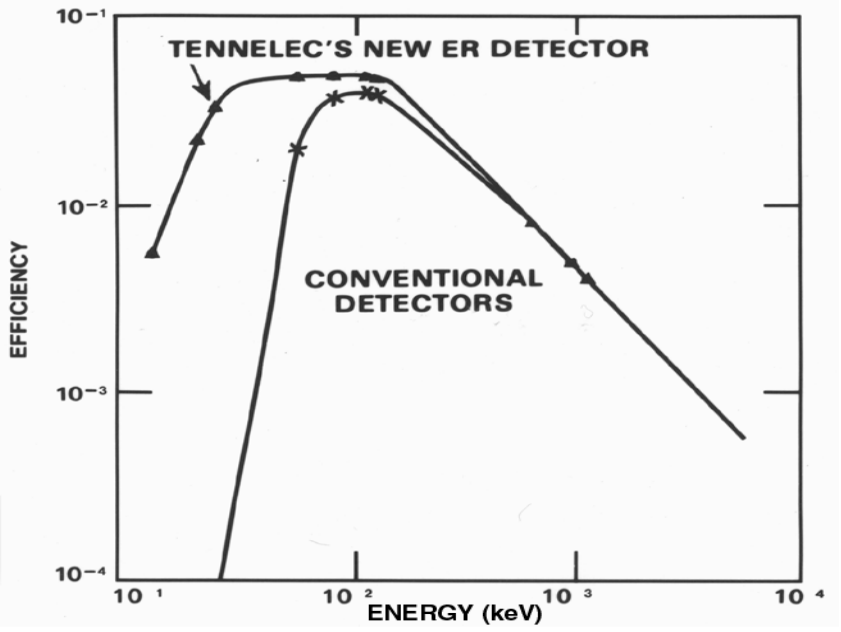


Fig. 58 - Cut away photo of a Ge(Li) detector

electron mobility and has a large bandgap. These properties lead to a high efficiency for electric charge collection and that produces good energy resolution. The detectors



Courtesy of Tennelec, Inc.

Fig. 59 - A commercial HPGe detector and cryostat

Courtesy of Radiation Monitoring Devices

Fig. 60 - Examples of CZT radiation detectors

are available in sizes ranging from 1 mm to 13 mm (1/2 inch) in diameter and from 1 to 2 mm thick. Figure 60 shows a variety of complete detectors manufactured by Radiation Monitoring Devices, Inc., including one mounted inside a BNC coaxial cable connector (top left corner of photo). Detectors have been made small enough to fit inside hypodermic needles for use in brain research.

The energy resolution of a cadmium telluride detector is better than NaI(Tl) scintillators but not as good as germanium counters. For low energy gamma ray spectroscopy, they are a competitor with NaI. However, at present, manufacturers have been unable to produce CZT counters with large volume. Thus, it takes much longer to obtain a suitable gamma spectrum at the higher energies as not enough gammas are interacting in the small volume, compared to a NaI detector. Figure 61 shows well resolved peaks at 20 keV and 140 keV. Because of the good sensitivity at low energies, the detectors have found application as wound monitors in plutonium workers. Because of the small size, vendors have recently been installing CZT detectors in hand held multichannel analyzers for homeland security applications.

CZT detectors have an operating temperature range of from -20° C to +30° C and require a bias voltage of only 50 V. The semiconductor material itself is not hygroscopic, so packaging into a detector assembly is easier as hermetic seals are not needed. Chemically, cadmium telluride is considered toxic, particularly in dust form.

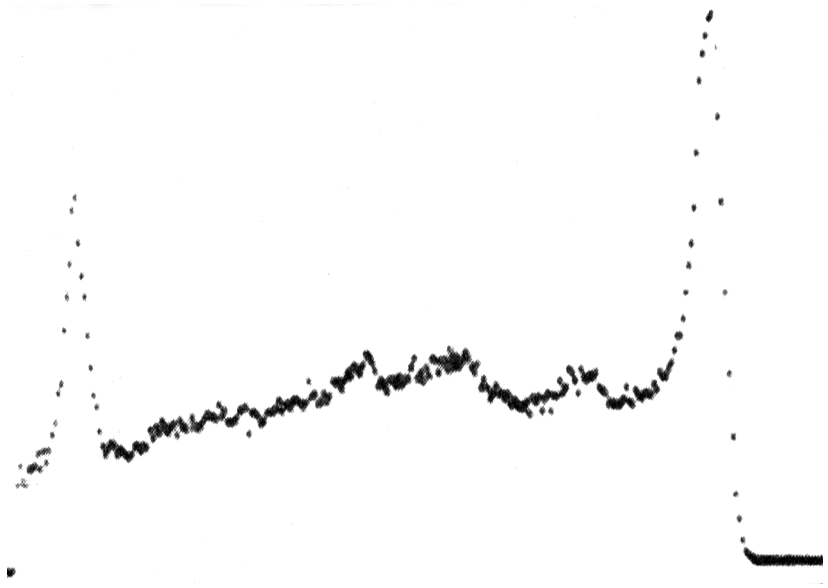


Fig. 61 - A CZT energy spectrum with peaks at 20 and 140 keV

Commercially, the substance has two other applications. In crystalline form, it is transparent to infrared light, so it has some use in making optical windows and lenses. It is also being used more often in the manufacture of thin film solar cells.

Problem Set

1. List an advantage and disadvantage of each radiation detection mechanism shown in Figure 1.

2. Name an application in which a chemical dosimeter would be useful.

3. Why is an ion chamber not referred to as a “counter?”

4. Calculate the number of electrons that would be released and the number that would be collected (per particle) in an ionization chamber exposed to alphas of 4.7 MeV. How many coulombs of charge, of one sign, would be collected per alpha particle?

5. What causes the region of continuous discharge in the gas-filled detector characteristic curve? Why is the counter unusable for radiation detection in this region?

6. Briefly describe the two methods used by ion chamber designers to prevent gas multiplication. Why are TWO methods needed?

7. Why are some pocket dosimeters lined with plastic? With Boron?

8. Why does a proportional counter usually have a very thin wire for the collecting electrode rather than the rod characteristic of ion chambers?

9. If the 4.7 MeV alpha of problem 4 were stopped in a proportional counter with a gas multiplication of 10^6 calculate the number of electrons collected and the total charge of one sign collected per alpha particle.

10. Define the term "dead time." Give typical values for an ion chamber, proportional counter and GM counter. Why does having a dead time limit the maximum count rate that can be measured by a counter?

11. Show in a calculation that if a proportional counter has output pulses of the order of millivolts a Geiger counter would generate pulses with amplitudes in the range of volts.

12. What is the instrument of choice for counting a mixed alpha-beta smear sample? Describe how it is possible to electronically distinguish between alpha-produced pulses and beta-produced pulses in this counter.

13. What is a multichannel pulse height analyzer instrument used for in radiation protection technology?

14. How can a proportional counter be made sensitive to thermal neutrons? to fast neutrons?

15. How is the pulse formation in a GM counter different from that in a proportional counter? How is the pulse terminated in the GM counter?

16. Describe how the problem of "saturation" can occur in a Geiger tube. What practical consequences might this pose for a radiation protection technologist?

17. Explain why a steel wall GM tube overresponds to low energy photons even though it was properly calibrated against a Cs-137 source. How would the tube respond if the steel wall were replaced by beryllium?

18. What is the chief application of liquid scintillation counters? Why is it necessary that the detector be in the liquid state for this application?

19. List one advantage and one disadvantage of a liquid scintillation counter used for counting alpha emitters on wipe samples.

20. Describe the principle of operation of a SDD neutron detector. What is the chief advantage over other neutron detectors?

21. Describe the various components used to fabricate a complete NaI(Tl) scintillation counter and the operating principle of each.

22. Why is it important to prevent a NaI crystal from contacting room air?

23. Calculate the % energy resolution of a NaI scintillator if the peak width at half amplitude is 90 keV for a 1.25 MeV gamma ray.

24. Would a NaI(Tl) counter be likely to distinguish the peaks from two gamma rays with energies of 900 keV and 940 keV? Why or why not?

25. Why must a scintillation counter be enclosed in a light-tight housing?

26. How can a scintillation counter be designed to record neutrons?

27. Name the three types of semiconductor counters discussed and give one practical application for each.

28. How is a "depletion region" formed? What role does it play in semiconductor counter operation?

29. Name an advantage and a disadvantage of a germanium counter compared to a NaI(Tl) scintillation counter.

30. What is the advantage of the HPGe detector compared to the lithium-drifted variety?

31. List one advantage and one disadvantage of a CZT detector compared to A) NaI(Tl) and B) HPGe.

S-1. What is meant by a "tissue equivalent" ion chamber? What is its advantage over ordinary ion chambers?

S-2. Is a detector resolving time the same, longer or shorter than the detector dead time?

S-3. What is the usual use of an extrapolation chamber in radiation protection?

S-4. An RO-20 ion chamber survey instrument has been calibrated at standard conditions. If it is used for a survey on a mountain where the barometer reads 28.4" and the temperature is 45° F, the meter readings should be multiplied by what factor for these conditions?

S-5. Describe how the construction of a Phoswich scintillator differs from a traditional scintillation counter.

S-6. Of what practical significance is the fact that plastic scintillation phosphors have such short pulse widths?

S-7. Why are lanthanum bromide crystals growing in popularity for scintillation spectroscopy?

S-8. How is it possible to have an HPGe detector with "110%" efficiency?

Other Resources

1. G. F. Knoll, "Radiation Detection and Measurement," 3rd Edition, John Wiley & Sons, New York, 2000.

2. W. Price, "Nuclear Radiation Detection," 2nd Edition, McGraw-Hill, New York, 1964. Available used on Amazon.com.

3. D. Gollnick, "Experimental Radiological Health Physics," Pergamon Press, New York, 1978. Available on Amazon.com.

4. Radiochromic Dosimetry - visit the following website for lots of information and for commercial availability of SIRAD dosimeters: www.jplabs.com.

8

External Personnel Dosimetry

Outline of this Chapter

Chapter Summary

This chapter is devoted to a study of radiation badge systems used to measure the dose equivalent received by a radiation worker from external fields. Chapter 9 will discuss the measurement of dose from internally deposited radionuclides. After a short introduction discussing characteristics of an "ideal" personnel dosimeter, the photographic badge is discussed in detail. Film blackening by radiation follows an "S" shaped curve with increasing dose. With proper calibration, and corrections applied for the huge overresponse of film at low energy, the film badge can report doses from about 0.1 mSv to about 10 Sv. Track film can be used for neutron dosimetry.

TLD badges use phosphors that respond linearly with increasing radiation dose. Heating the phosphor releases a light pulse with an intensity that is proportional to the absorbed dose. Commercially used phosphors usually have less energy dependence than film because they are made of materials with a Z number closer to 7.5 than film. By incorporating Li-6 and/or B-10 in the phosphors, TLDs can be made for neutron dosimetry.

The new kid on the block is OSL, Optically Stimulated Luminescence. A light beam excites the luminescence rather than heat as in the TLD case. Aluminum oxide has proven to be very radiation sensitive, and the beta ray and gamma ray dose information is retained in the reading process, in contrast to TLDs.

In the USA, all radiation badge processors must now be accredited by the NVLAP, a quality assurance program under the National Institute of Standards and Technology. Badges are tested in several different types of radiation fields. The program has greatly increased user confidence in published badge results.

Dosimetry following accidental criticality is complicated by the uncertainties in the neutron component of the dose. A criticality badge makes use of activation detectors to separate the neutrons into energy intervals to allow correct interpretation. Such a badge typically contains sulphur, indium, gold, cadmium and copper.

An emerging technology is represented by electronic dosimeters. These units may someday replace both alarming dosimeters and personnel badges. The dosimeter is plugged into a reader installed in a personal computer and the beta-gamma-x-ray dose information stored and added to the previously accumulated dose information for a worker. Alarm set points on dose and rate are programmed by the PC. They are yet to be approved in the USA as a replacement to a film or TLD badge for legal worker dosimetry.

Special badges are available for specific applications. Badges for radon dosimetry, and fast and intermediate neutrons are briefly covered.

The Ideal Personnel Dosimeter

Before beginning with the details of current badge systems, the characteristics of an "ideal radiation badge" will first be covered. If you were totally free to choose the operational characteristics desired in a badge system, what would you ask for? Some characteristics of an ideal badge are presented in Figure 1.

Small size	**Resists light, heat, humidity**
Rugged construction	**Rapid, simple readout**
Energy independent	**Cheap**
Angular independent	**Range 10 mrem to 1000 rem**
Radiation type indicated	**Re-readable**
Negligible fading	**Accurate**
Linear radiation response	**Professional appearance**
Direct Reading	**Alarming option**

Fig. 1 - Desired characteristics in a personnel badge

An energy independent response means that the radiation badge gives the correct dose equivalent no matter what the energy is of the detected radiation. Thus, no energy correction factor need be applied. Angular independence means that the badge reads correctly if exposed from the side as well as directly through the front. Fading means the loss of recorded information with the passage of time. The time span of particular interest is the exposure period for the badge, usually one calendar month. Rapid readout means the answer is obtained quickly. Simple readout means that the procedures for extracting the dose information are technically simple, thus, reducing the possibility of errors. The desired range of 10 mrem to 1,000 rem bears comment. Current radiation control regulations do not require reporting a worker dose of less than 10% of the annual limits. This translates to about 10 mrem/week. The upper limit of 1,000 rem is determined from the maximum survivable dose for humans under acute exposure conditions and with medical care (see Chapter 4). Doses over this limit are of academic interest but of no concern to the former(?) worker! Having the badge re-readable means the actual sensitive element in the badge is available at any time to be read over again if some question of the dose arises. It should be noted that this is not a legal requirement in the United States. A dosimetry report form signed by the technologist actually reading the badge is considered of equal weight in the "eyes of the law" as the actual sensitive element of the badge.

Obviously, the purpose of a radiation badge is to measure "worker dose." How the badge does this is, of course, subject to interpretation. Since 1988, any badges issued to workers under an NRC or agreement state license must be supplied by an "accredited dosimetry processor." This is an organization with current accreditation from the National Voluntary Laboratory Accreditation Program (NVLAP) of the National Institute of Standards and Technology. In the government sector, the corresponding organization is the Department of Energy Laboratory Accreditation Program or DOELAP. The accreditation must be for the types of radiations which the badge is designed to report. For each approved radiation, the processor reports the "Shallow

Dose Equivalent, $H_p(0.07)$" or "Deep-dose Equivalent, $H_p(10)$" or both, as appropriate. These are defined as the dose equivalent, measured in rem, at the respective depths of 0.07 mm and 10 mm in a rectangular slab of soft tissue of a density of 1 g/cm^3 and a size of 30 by 30 by 15 cm. These definitions are equivalent to what were loosely called the "skin dose" and "whole body dose" in the good old days!

The remainder of this chapter will cover several commonly used personnel dosimetry badge systems. These include film badges, TLD badges, OSL badges, electronic dosimeters and the activation type criticality badges. In each case, the principle of operation will be discussed followed by operating characteristics and some advantages and disadvantages of the use of that type of badge.

Photographic Badge Systems

Basic Principles

Photographic film used for personnel dosimetry is constructed similar to the sketch shown in Figure 2. The foundation for the film is a transparent polyester plas-

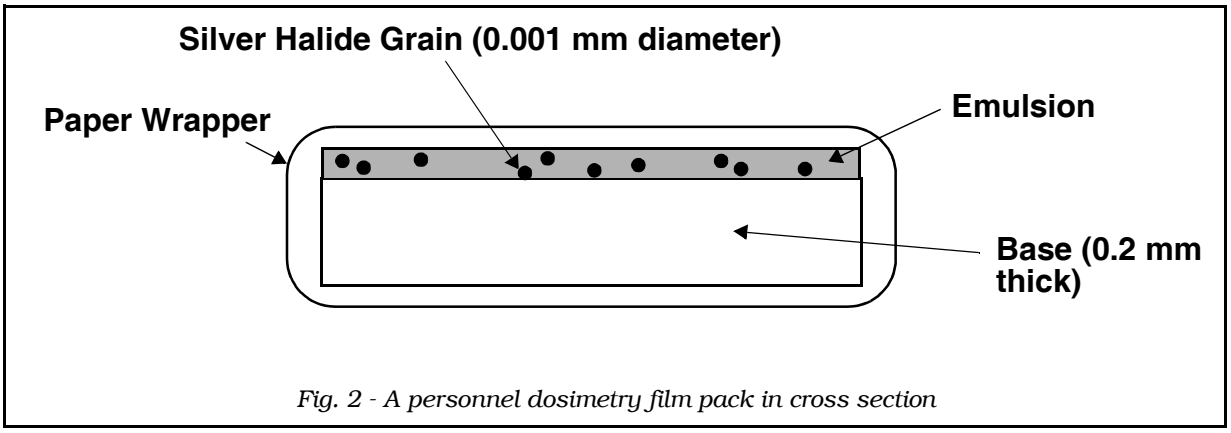


Fig. 2 - A personnel dosimetry film pack in cross section

tic sheet. It provides the rigidity needed to easily handle the processed film. The emulsion is spread on top of either one or both sides of the plastic base, in a coating that is 25 to 30 microns thick (recall that 1,000 microns = 1 mm). It is a gelatin matrix which supports the suspended silver halide grains. Most of the grain is silver bromide (AgBr) but a small percentage of the molecules are silver iodide (AgI). Each grain is from 1 to 2 microns in diameter and contains about 10 million Ag^+ ions covalently bonded to the bromine or iodine ions. After being manufactured, the film is encased in several layers of paper to form a light-tight housing. A photograph of a commercial personnel dosimetry film pack is given in Figure 3.

The theory of film exposure and development was described by Gurney and Mott back in the 1890s. Energy from the radiation which interacts in the film will release electrons in the AgBr grains. When it slows down sufficiently, the electron will be captured by an Ag^+ ion which will then be reduced to a neutral Ag atom. The neutral Ag now acts as an "electron trap" since it is a different species than the surrounding silver and bromine ions. The next electron to pass in the vicinity will be attracted to the neutral Ag and be captured to form an Ag^- ion. This ion will experience an

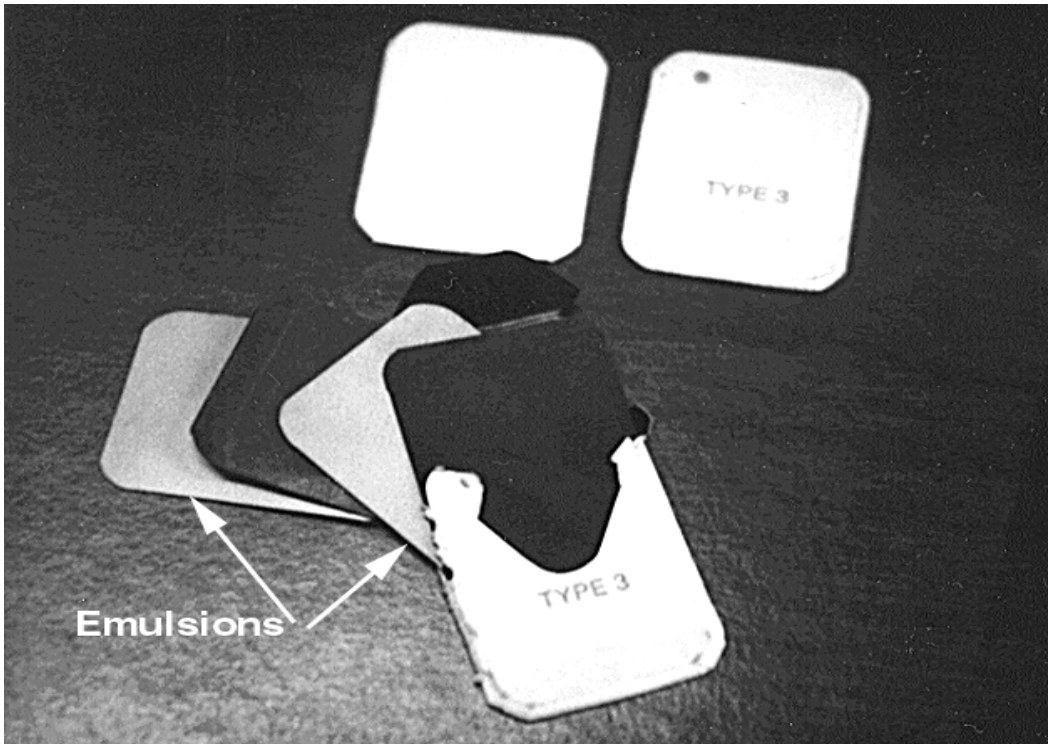


Fig. 3 - A commercial film pack

attractive Coulomb force from a nearby Ag^+ ion and the two will combine to form an even larger electron trap, now 2 neutral silver atoms large.

As shown in Figure 4, this process continues repeating until the trap gains at least 4 to 6 Ag atoms. At this point, the trap is now designated a "latent image center," and it plays a crucial role in the development process.

In a practical film badge, the film is exposed to radiation over some period such as a month. Different grains are, thus, at different places along the chain of events leading to latent image center formation, depending on the total radiation dose received by the badge. At the end of the badge period, the film pack is removed from the badge holder and processed.

$$\text{Radiation} + \text{Film} \rightarrow e^-$$
$$e^- + Ag^+ \rightarrow Ag$$
$$Ag + e^- \rightarrow Ag^-$$
$$Ag^- + Ag^+ \rightarrow 2\,Ag$$
$$2\,Ag + e^- \rightarrow AgAg^-$$
$$AgAg^- + Ag^+ \rightarrow 3\,Ag$$
$$\text{Etc. .....}$$
$$\text{4 to 6 Ag} = \text{Latent Image Center}$$

Fig. 4 - The formation of a latent image center

The first step is development. The bare film is placed in a light-tight tank and flooded with a solution rich in free electrons. The electrons permeate the emulsion and begin attacking the silver ions in the grains with the objective of reducing them to metallic silver. At this point, the latent image centers carry out their role as a catalyst in the grains. The silver reduction reaction, $Ag^+ + e^- \rightarrow Ag$, is carried to completion only in grains having a latent image center. In grains without such a center, the reaction rate is too slow to proceed. If development time, agitation and solution temperature are carefully controlled, it is possible to remove the film at this point in development before any of the non-image center containing grains start developing. This preserves the dose information on the film in the form of the number of grains which have been reduced to metallic silver.

The next step, after washing out the developer, is called "fixing." The film is immersed in a sodium thiosulphate solution which dissolves out any remaining AgBr grains. The deposited Ag metal grains are untouched. The fixing is followed by a long wash to remove all the fixer and AgBr. If this is not done, the film will gradually yellow due to sulphide formation. Then, the final step is to dry the film which hardens the emulsion sufficiently for handling purposes. The film appears to be blackened as a result of the metallic silver grains blocking incident light. The amount of "blackness" is, of course, related to the radiation dose for the badge.

The processed film is now ready for interpretation. The film is "read" by quantitatively measuring the blackness with an optical densitometer. This instrument determines the film optical density which is defined in Figure 5. The reason for the logarithmic dependence is that the human eye perceives changes in light intensity logarithmically (just as the ear responds logarithmically to sound intensity). The meaning of the O.D. is clearer from the table of values in Figure 6. The human eye can still distinguish changes in optical density up to about O.D. = 4 (only one light photon in 10,000 getting through). See Sample Problem 1.

$$\text{Film Optical Density} = \text{O.D.} = \log_{10} I_0/I$$

where I_0 = Incoming light intensity

and I = Transmitted light intensity

Fig. 5 - Definition of film optical density

Optical Density	I_0/I	% Transmission
0	1	100
1	10	10
2	100	1
3	1000	0.1

Fig. 6 - Optical density vs. transmission

Sample Problem 1

GIVEN:
A film has a measured O.D. of 1.18.
FIND:
What % transmission does this film have?
SOLUTION:
O.D. = $\log_{10} I_0/I$ = 1.18 so, by definition, $I_0/I = 10^{1.18}$ = 15.1. Now, % transmission = I/I_0 x 100% = 1/15.1 x 100% = 6.60%

The densitometer is illustrated in the sketch of Figure 7. A light source projects a beam through a hole in the top. The film is placed over the hole and a light detector is positioned on top of the film. The fraction of light transmitted is converted to O.D. units and displayed on a meter scale. Photocells or photomultiplier tubes are commonly used for the light detector. If the processed films are stored they can be re-read again if a future question arises over some worker's dose record.

Film Response To Radiation

If a series of film packs, all from the same lot number, are exposed to different doses from the same radiation source and then the films all developed identically, a graph of the optical density versus the radiation exposure in roentgens has the shape shown in Figure 8.

This film "characteristic curve" exhibits a "toe", "linear region" and "shoulder" similar to a letter "S." Clearly, film shows a nonlinear response with exposure. For proper dose assessment, a calibration curve like Figure 8 must be constructed for

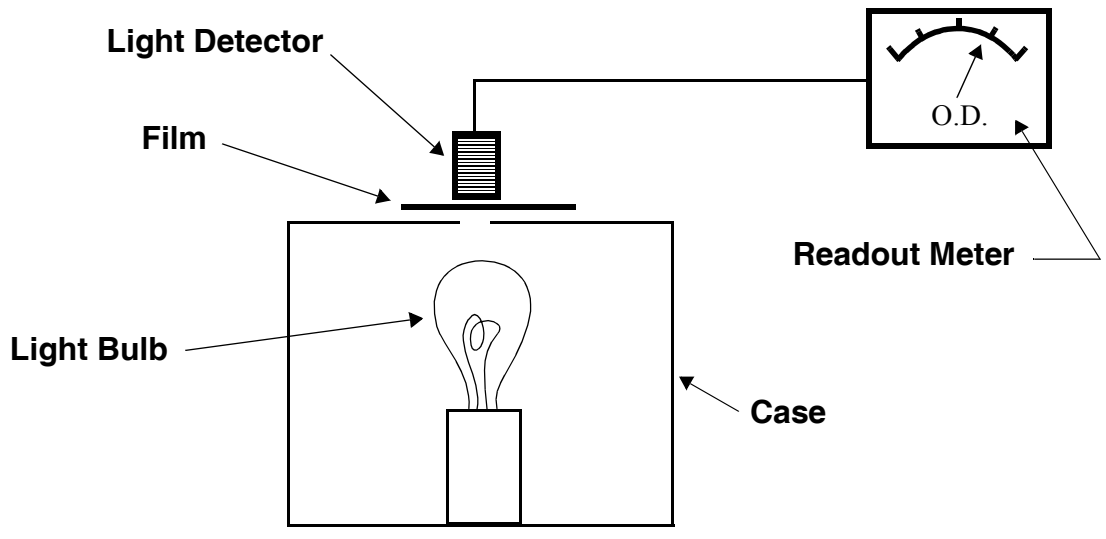


Fig. 7 - Construction of an optical densitometer

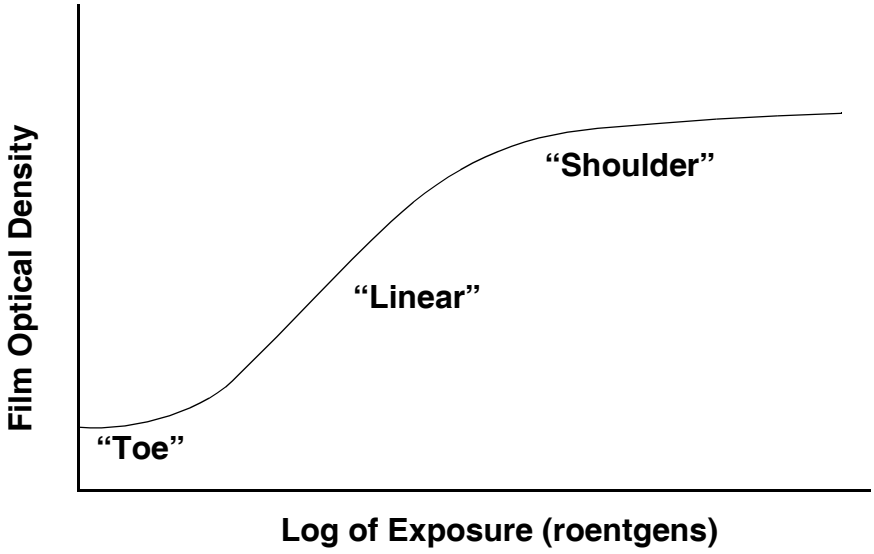


Fig. 8 - Film response to radiation exposure

each different lot number ("batch") of film used.

It was mentioned above that the film emulsion is spread either on one or both sides of the plastic support base. A single piece of photographic film is capable of covering a range of exposures of about a factor of 1,000. That is, the lowest and highest readings possible (from just starting into the toe of the characteristic curve to just entering the flat region above the shoulder) differ by 1,000 times. If the film sensitivity is adjusted to read 10 mR in the toe, the highest reading before that film is completely black will be 10 R (i.e., 1,000 x 10 mR = 10 R). This is satisfactory for normal routine dosimetry, but it is not high enough for accident conditions. The personnel dosimetry film pack thus includes two separate pieces of film differing in sensitivity by about a factor of 100. The "low range film" (highest sensitivity) covers 10 mR to 10 R and the "high range film" covers 1 R to 1,000 R for accident readings. The high sensitivity low range film covers the mR range by using a double emulsion film. This puts more silver bromide in the path of the gamma rays so there is a greater chance of an interaction. The low sensitivity, high range film is made with the emulsion spread only on one side of the plastic base.

The other major characteristic of interest in a dosimeter is the energy response. This can be measured in film by taking a set of film packs, irradiating them all to an exposure of one roentgen from photon sources of different energies, and then processing them identically. A graph of O.D. vs. energy would be similar to Figure 9.

At photon energies above about 200 keV the film shows energy independent response. But at low photon energies, the film overresponds by as much as 2,000% to 4,000%. This is due to the Z^3 dependence of the photoelectric photon capture probability at low gamma ray energies. The film is being used to measure an exposure (defined in the medium AIR with an effective Z = 7.5) by placing silver (Z = 47) and bromine (Z = 35) in the path of the gamma rays.

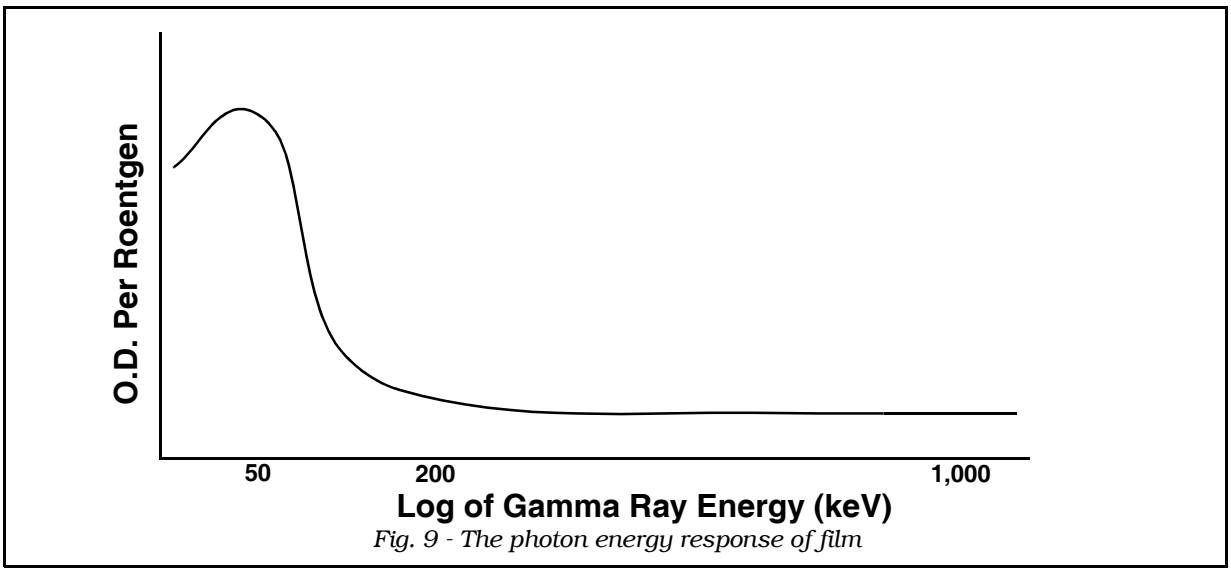


Fig. 9 - The photon energy response of film

At energies above 200 keV, the Compton scattering interaction is dominant and its cross section is independent of Z so silver and air absorb gamma rays equally per gram. Below 200 keV, the photoelectric effect is dominant and then the silver is a much better absorber of gamma rays than air so the film reads very high. At extremely low energies, the paper wrapper around the film and the badge case attenuate the photons so the response again decreases. It then becomes essential to know the energy of the photons that expose a badge before the personnel dose can be assigned. This information is available as a result of placing the film pack in a badge holder.

Film Badge Holder Design

The holder employs various materials in the form of filters to allow the energy of the radiation to be measured. Once the energy is determined, an energy correction factor is applied to the optical density to obtain the exposure value for that badge. The principle of operation is perhaps best explained by reference to Figure 10.

This particular holder uses an open window and three filter regions: aluminum, lead and plastic. Other badges designed to provide even more information make use of additional iron, cadmium, tantalum, tin and teflon filters. At medium photon energies, most of the interactions will be Compton scattering. Since the cross section per gram is energy independent, the Al, Pb and plastic will all attenuate equally so the film will be uniformly blackened under all the filters. At low photon energies, the attenuation of the filters will be proportional to Z^3 due to the photoelectric effect. Now, the Pb will highly attenuate the field, and the Al filter will stop many more photons than the plastic so the film will be lightest under the lead and darkest under the plastic filter. By measuring the ratios of the film optical densities under the filters, the film badge processor can determine the average photon energy and thus, not report a low energy photon exposure as 20 to 40 mSv when the badge only received 1 mSv.

As a practical example, Co-60 photons (1.17 and 1.33 MeV) produce equal

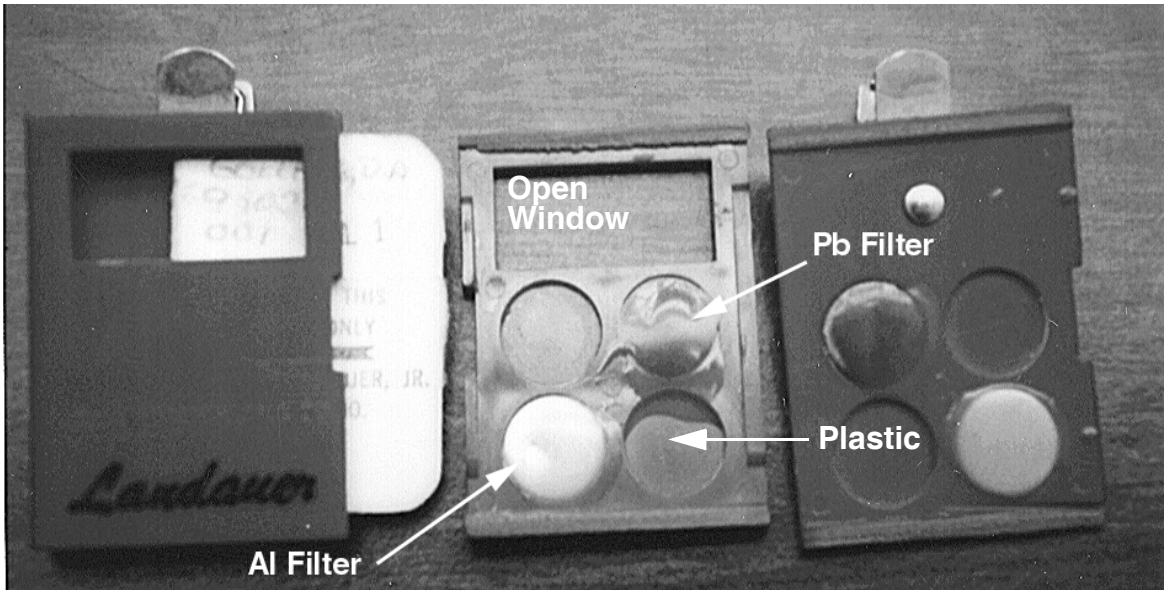


Fig. 10 - A commercial film badge holder

optical density for the entire film while low energy X-rays will produce distinct images of the filters on the badge. This is demonstrated in Figure 11 where identical film packs have been placed in the holder of Figure 10 and given the same exposures to

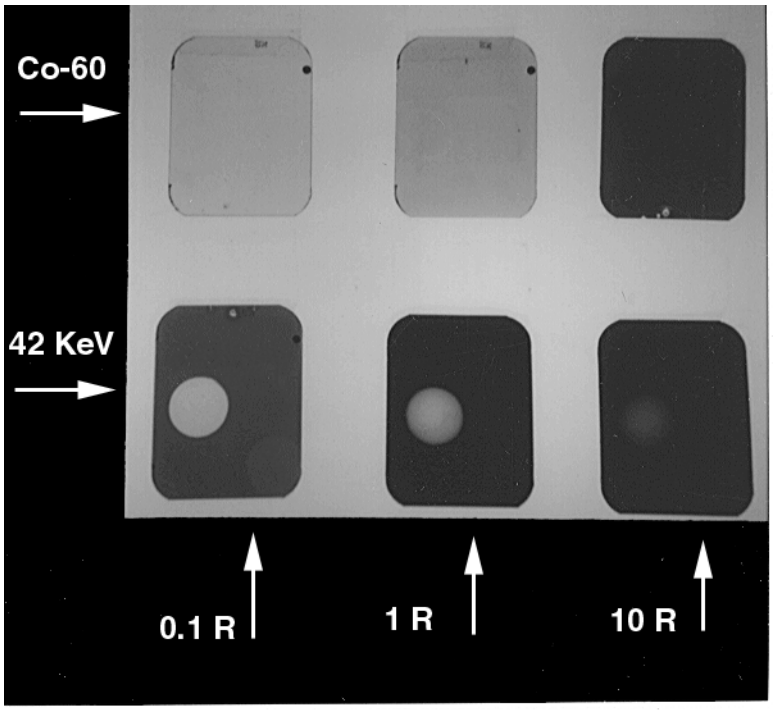


Fig. 11 - Film response to low and medium energy photons

Fig. 12 - Additional film badge holders

either Co-60 or 42 keV X-rays followed by identical processing.

Although it takes additional work to make the energy determination with a film badge system, the energy is known. This could be important in ALARA programs where knowledge of the energy might indicate which job location was responsible for unwarranted exposures of workers. Some TLD badge systems (to be discussed below) do not provide photon energy information.

Another advantage of the film badge system is that multiple significant exposures of the badge under operational conditions will show up as multiple filter images. These films can be distinguished from films with a single filter image that would indicate a single acute exposure. Finally, through proper design of the filters, the angle of incidence of the exposure can also be determined by the film badge.

The previously mentioned teflon filter gives maximum discrimination between beta rays and photons. The difference in optical density between the open window region and the teflon filter region on the processed film gives the dose due to the betas. Unfortunately, film badges respond poorly to low and medium energy betas. The layers of paper wrapper surrounding the film emulsion will absorb beta rays up to about 1 MeV to such an extent that only a qualitative indication of the “presence” of high energy betas can be obtained. Film also shows a strong energy dependence for betas so that unacceptably large “correction factors” must be applied to medium energy beta sources, and the beta energy must be known beforehand. Commercial film badge processors generally do not report any doses from betas below 1 to 1.5 MeV in energy. (If in doubt, check the “fine print” on the back of your monthly report form). Figure 12 shows some other types of commercially available film badge holders.

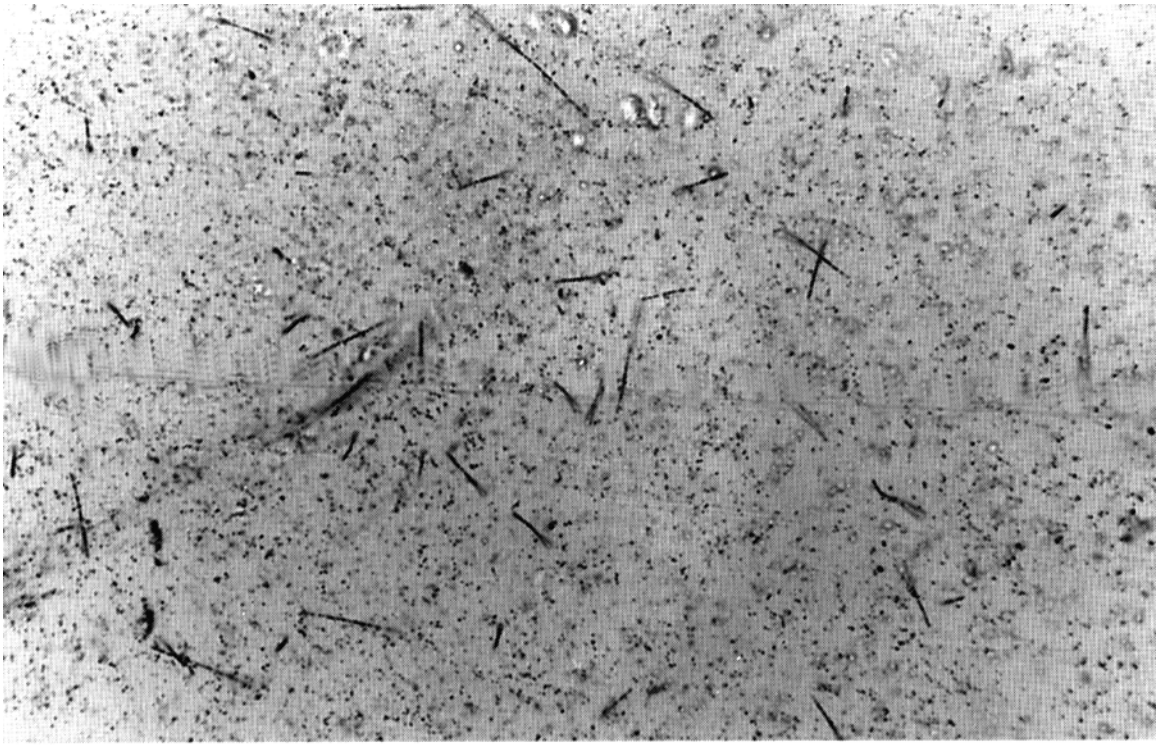


Fig. 13 - Microscopic tracks produced by recoil protons in fast neutron track film

Film badges can also be used to record neutron exposures. Thermal neutrons can be detected in a couple of ways. In the special neutron track film, the emulsion is made about three times thicker than the beta gamma X-ray film discussed above. Also, the grain size is reduced to about 0.3 micron. When exposed to thermal neutrons, nitrogen nuclei in the emulsion can capture a neutron which produces a proton and a C-14 nucleus. The ejected proton has high stopping power and so travels only a short distance in the emulsion. Because it deposits large amounts of energy, the proton is capable of directly producing latent image centers along its short path. This series of grains in a row appears after development and is termed a "track." Under microscopic examination, the tracks in the emulsion look like short lines. By counting the number of tracks in a unit area, the thermal neutron dose is determined. In regular, non-track film, thermal neutrons can be measured by incorporating a cadmium filter. When a thermal neutron is caught by a Cd nucleus, capture gamma rays are emitted which cause higher optical density in that region than in the adjacent film regions.

Fast neutrons are detected in a film badge by using track film. Neutrons with energies over 0.5 MeV will elastically scatter from hydrogen nuclei in the emulsion. The recoiling hydrogen then is capable of producing a track which can be seen microscopically after film processing (see Figure 13). Commonly used fast neutron track films have a useful range of about 200 mrem to 250 rem. A dose equivalent of 1 rem of fast neutrons produces a track density of the order of 26,000 tracks per cm^2. Figure 14 shows a commercial track film packet.

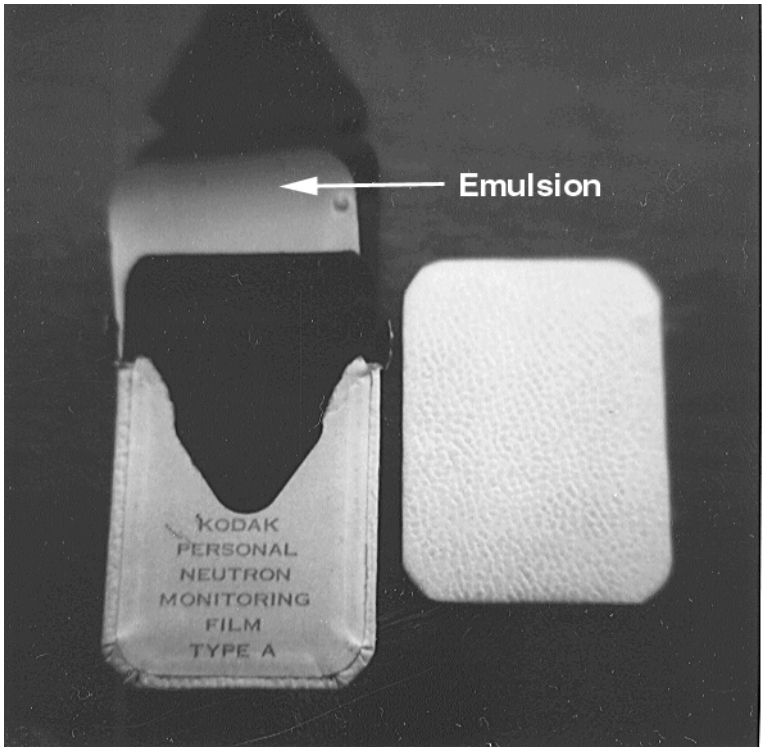


Fig. 14 - Kodak NTA neutron track film pack

Fading of the tracks with time is a major problem with track film. It is greatly accelerated by high humidity. At normal humidity, about one-half of the tracks produced in a properly sealed film pack will have faded below the detection limit within two weeks after formation. An unsealed track emulsion will have a fading half-life of only 2 days.

Most commercial film badge processors report fast neutron doses only for neutrons between 1 and 14 MeV. Therefore, neutron track film is virtually useless in reactor health physics programs. The average neutron energy inside containment of PWR nuclear plants is between 50 and 100 keV and in BWR plants the average varies from 150 to 250 keV. None of these neutrons would be recorded by track film.

The etched track dosimeter has found growing acceptance in neutron dosimetry. Neutrons above about 1 MeV incident on certain plastic foils will transfer enough energy to protons (hydrogen nuclei) in an elastic collision to enable the proton to produce a latent track. By later etching of the foil in a hot caustic solution, these latent tracks appear as puncture holes in the plastic foil. They can then be counted to determine the fast neutron dose equivalent.

Figure 15 shows a commercial badge using the CR-39 plastic foil combined with a TLD albedo dosimeter. The CR-39 covers the fast neutrons and the albedo section works in the lower energy ranges. The badge shown in Figure 15 has a minimum detectable dose of about 10 millirem. The CR-39 plastic is not sensitive to X-rays or gamma rays. The badge shows no fading in a one month exposure period at 22° C

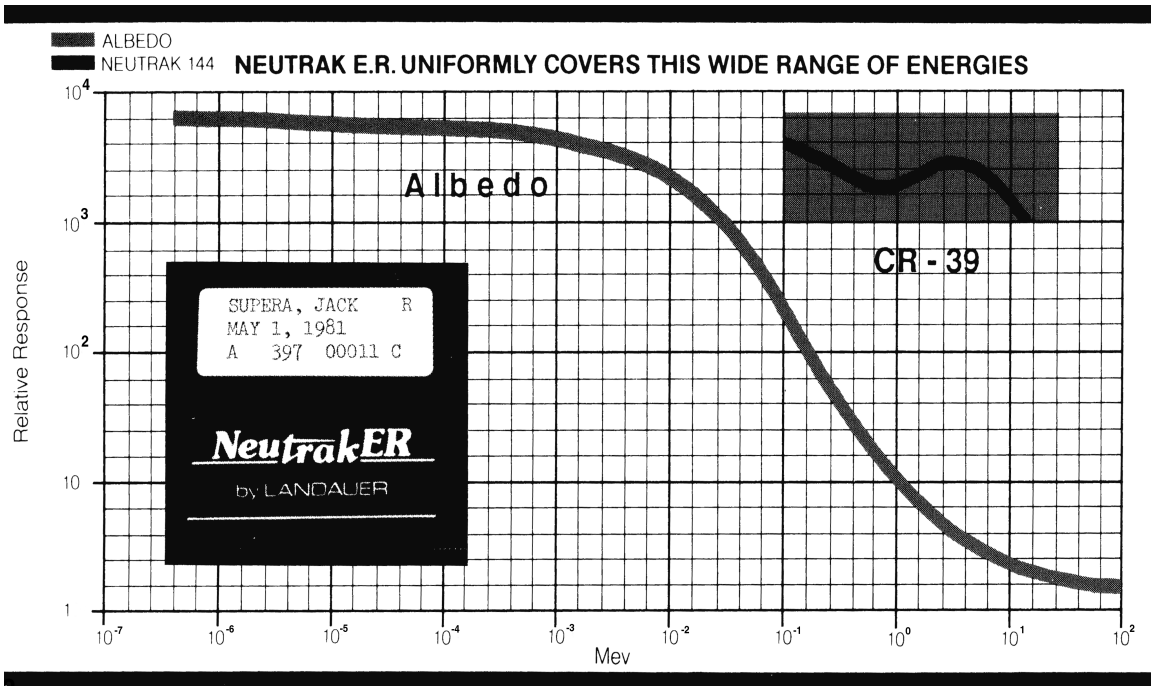


Fig. 15 - A commercial etched track and albedo combination neutron badge

Courtesy, Landauer, Inc., Glenwood, IL

temperature. The accuracy of the results is improved if the energies of the neutrons in the workplace are known. Figure 16 shows a magnified view of holes etched in a piece of CR-39 plastic foil.

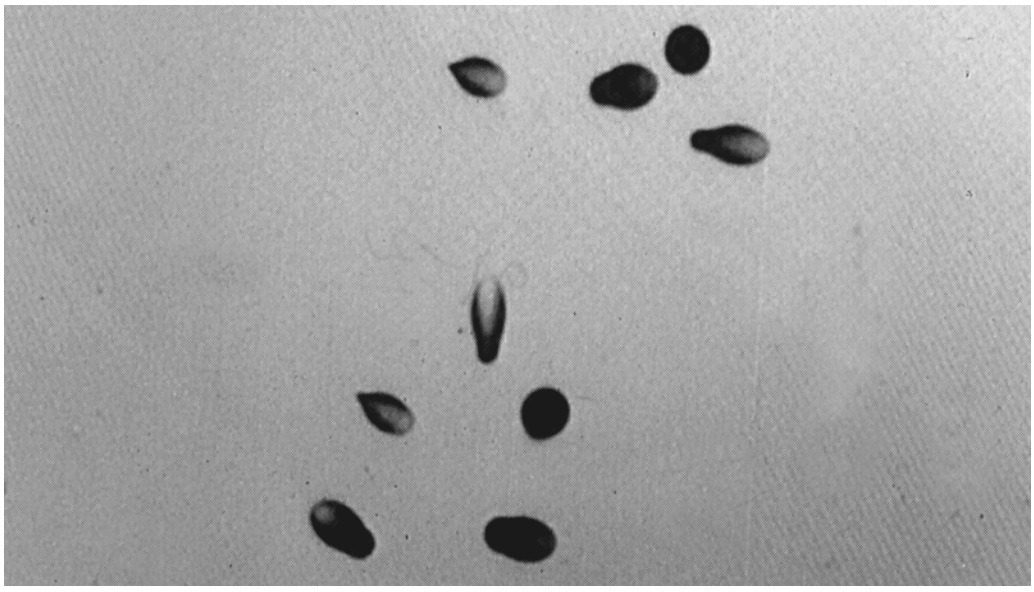


Fig. 16 - Microscopic holes produced in CR-39 after etching

Courtesy, Landauer, Inc., Glenwood, IL

Thermoluminescence Badge Systems

Principle Of Thermoluminescence

The phenomenon of thermoluminescence (or TL) was rediscovered in the 1960s and was perfected as a dosimetry technique in the 1970s through introduction of a higher level of quality assurance in the chemical makeup of the phosphors. As a result of interactions, the ionizing radiation transfers energy to electrons of the phosphor atoms. Through a series of steps, the thermoluminescence process releases light as illustrated in Figure 17. The energized electrons detach from the atoms (Step 1 in Fig. 17) and move somewhat freely (Step 2) around inside the phosphor crystal. Many of them eventually become trapped at a luminescence center (Step 3). These are usually impurity atoms added to the phosphor during manufacture. The impurities are carefully chosen to produce relatively stable electron traps of the desired energy. When the phosphor is heated, the thermal energy causes the electrons to escape from the traps (Step 4) and return to their ground state, i.e., they become reattached to the phosphor atoms. In dropping from a higher energy to a lower one, the energy difference is given off in the form of a light photon. When the electron is bound to a phosphor atom, it is said to be in the "valence band." When it becomes free, it goes into the "conduction band". The energy gap between the valence and conduction bands is called the forbidden energy gap. The trapping centers are located within the forbidden energy gap.

The energy actually received by the phosphor, the absorbed dose, directly determines the number of electrons released to the conduction band. This in turn determines the number of electrons to be trapped. When heated, all the traps release their electrons so that ultimately the intensity of the light flash is directly proportional to the phosphor dose. This offers the possibility of a dosimeter with a linear radiation response.

Examining the light flash in more detail, it becomes evident that the light is

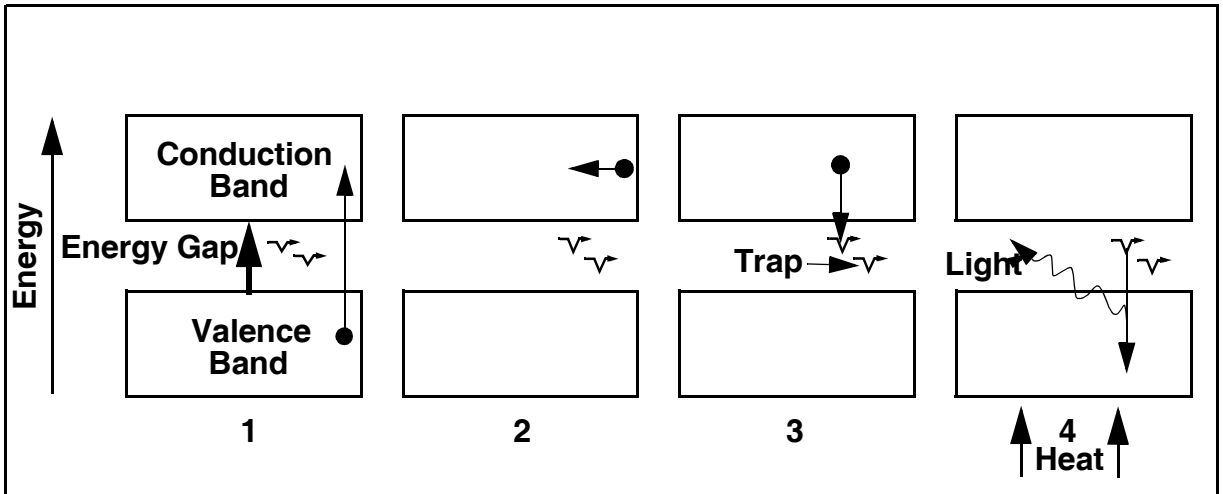


Fig. 17 - The thermoluminescence process

actually given off in separate bursts as the phosphor crystal reaches different temperatures. This is best shown by a "glow curve" which is a graph of the thermoluminescence light intensity versus the phosphor temperature. Glow curves for the most popular thermoluminescent dosimeter (TLD) materials, lithium fluoride (LiF), lithium tetraborate, ($Li_2B_4O_7$:Cu) and calcium sulphate, ($CaSO_4$:Tm), are given in Figures 18, 19 and 20.

Each of these different peaks is caused by a different set of electron trapping

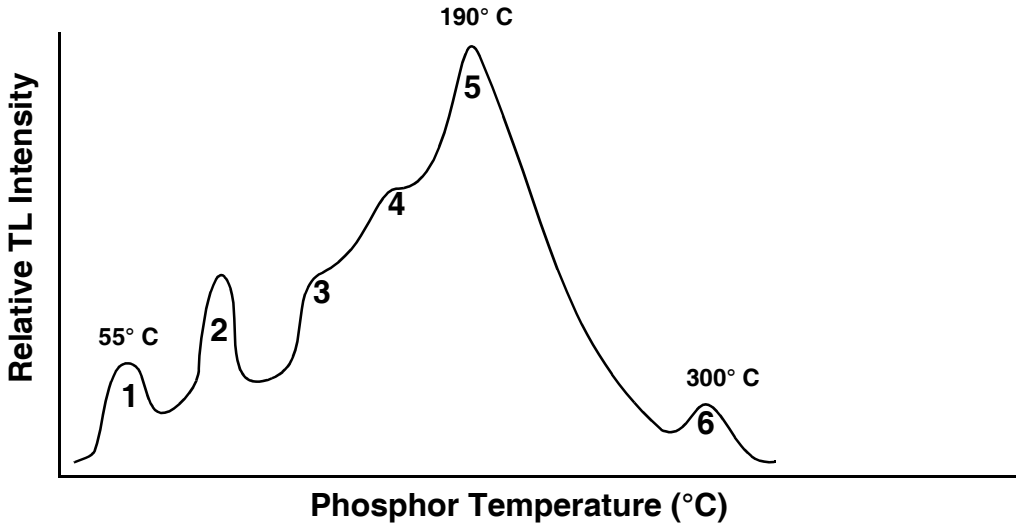


Fig. 18 - The TL glow curve for lithium fluoride

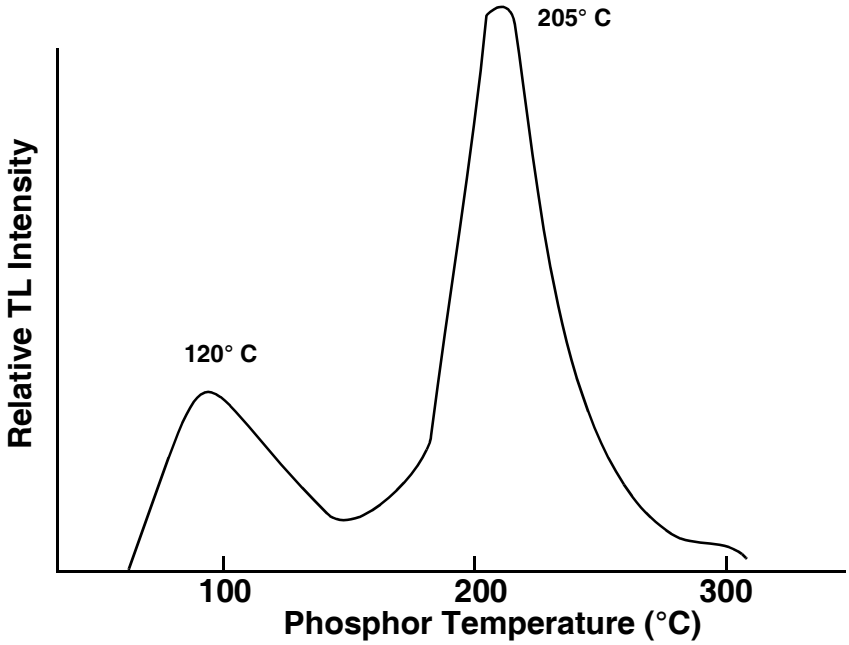


Fig. 19 - The TL glow curve for lithium tetraborate

centers. Each set is located at a different energy within the forbidden gap and so different amounts of thermal energy must be supplied to the phosphor to release the electrons in the respective traps. Each trap is thus characterized by a particular temperature at which it releases electrons. For example, LiF has traps at six different

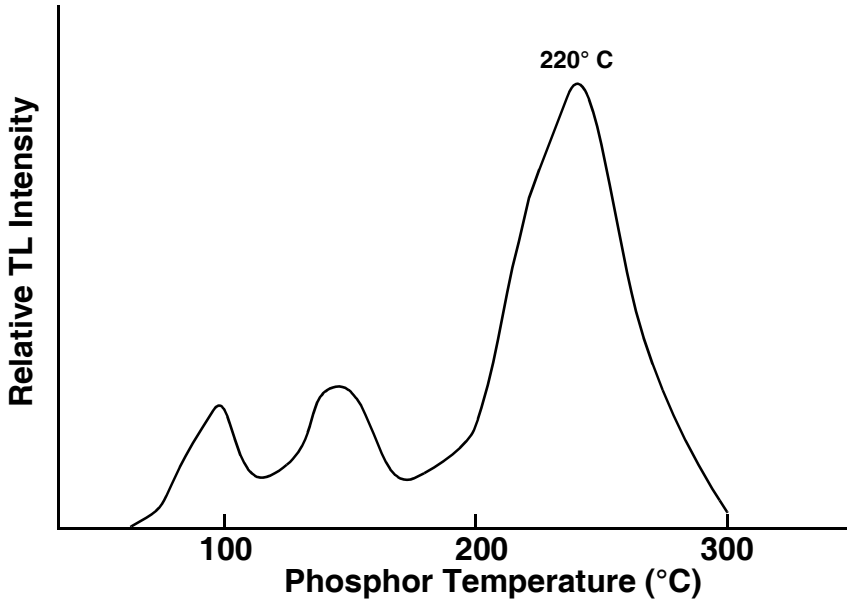


Fig. 20 - The TL glow curve for calcium sulphate

energies, while calcium sulphate has traps at three different energies.

As discussed earlier, fading (signal loss with the passage of time) is always an important factor in the choice of a radiation badge. For TLD badges, the fading is directly related to the electron trap energies. The higher the temperature required to dump a trap, the less the fading. Fading is often measured by the fading half-life, the time it takes to lose half of the stored dose information. A one Sv dose equivalent will read out as 0.5 Sv after one fading half-life. In the case of LiF, the fading half-lives of the various peaks are listed in the table of Figure 21 along with the temperature at which each peak releases its electrons.

The glow curve for each different phosphor used in dosimetry is different since the basic chemical compound determines the energies of the valence and conduction bands and the particular impurity atoms "doped" into the phosphor during manufacturing determine the trap energies in the forbidden gap. To be useful for personnel

Peak Number:	1	2	3	4	5	6
Temperature, °C:	55	105	150	170	190	300
Fading Half-life:	5 min	10 hr	6 mo	7 yr	80 yr	10^6 yr

Fig. 21 - LiF glow curve peak characteristics

dosimetry, the traps need to have a reasonably long fading half-life (compared to the badge period length) and the frequency (or wavelength) of the emitted light should reasonably match the photomultiplier tube response. Looking back at the glow curves of Figures 18 – 20, it is evident that the first two peaks in LiF and $CaSO_4$:Tm and the first peak in $Li_2B_4O_7$:Cu fade much too rapidly to be useful. On the other hand, it can be observed that the low temperature peaks are well separated from the remaining peaks and furthermore are quite a bit smaller than the remaining peaks. If these low temperature peaks could be eliminated, the fading problem would be insignificant. In practice, this is done by using a "preheating" cycle on the reader. For LiF, the phosphor is heated to about 135° C and any light emitted is ignored. Then, the temperature is raised to about 255° C and the emitted light is recorded for peaks 3 through 5. Peak 6 is used for neutron dosimetry and will be discussed later. $CaSO_4$:Tm and $Li_2B_4O_7$:Cu are preheated to about 120°. Preheating produces dosimeters with acceptable fading over a one-month period.

LiF emits light at about 4000 angstroms which puts it in the visible optical spectrum with a bright blue color. This is well matched to commercial photomultiplier tubes. Lithium tetraborate emits at 3680 angstroms (violet) and $CaSO_4$:Tm emits at 4520 in the blue-green portion of the visible spectrum.

In principle, a TLD reader is a simple instrument. It consists of a support for the phosphor, a heater to raise the temperature, a photomultiplier tube to measure the light output and some type of meter or digital readout to display the information. Figure 22 is a schematic drawing of the components.

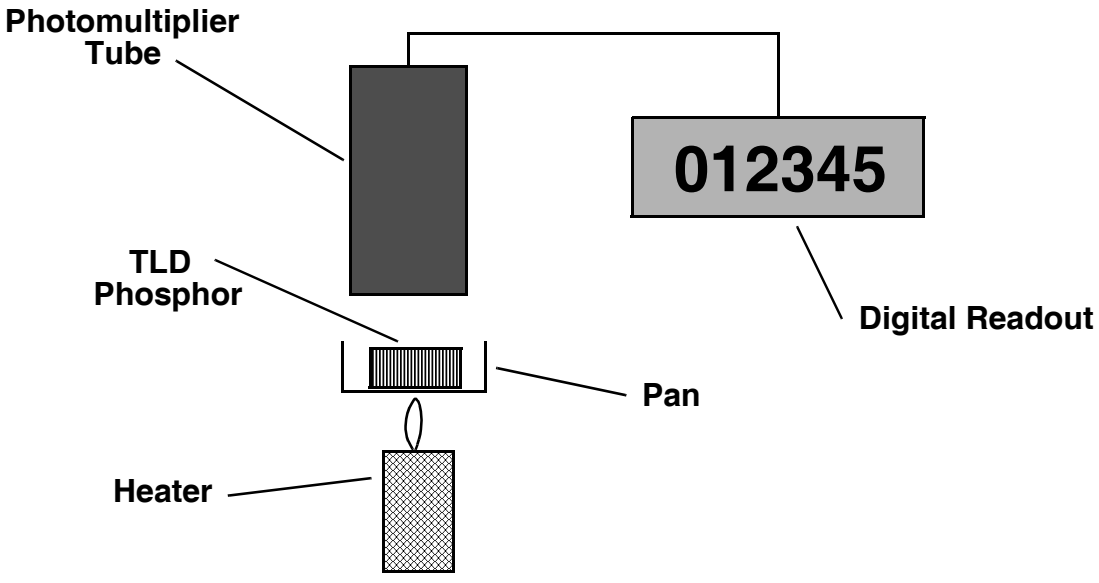


Fig. 22 - The basic components of a TLD reader

In practice, the commercial reader is complex and costly. The entire assembly must allow ready access to quickly load and unload badges yet be completely light-tight for the sensitive photomultiplier tube. The tube sensitivity depends somewhat on its temperature so some readers employ cooling units for the photocathode. Since a quantitative readout is desirable, most TLD readers use a digital voltmeter which gives

a digital display that can be calibrated directly in mR or R or the corresponding SI units. Since the rate of heating and the maximum temperature reached in the read cycle affect the final reading, these are carefully controlled by electronic circuits. To measure at the lowest end of the dose range, infrared light from the heater assembly (pan glow) must be filtered out and then corrected electronically. Finally, since oxygen is itself a thermoluminescent gas, room air must be excluded from the vicinity of the heater and pan during reading. Otherwise, a spurious reading (usually of the order of a few 10s of mR equivalent gamma rays) is added to the readout result. This is accomplished for LiF readers by installation of a plumbing system which bathes the phosphor in pure nitrogen or argon purge gas during the read cycle. Additional information on commercial readers will be presented after some individual phosphor characteristics are first explored.

Characteristics Of Lithium Fluoride

The dose response and energy response will now be examined for the LiF phosphor and the results compared with film. Figure 23 shows the basic response of LiF as a function of increasing radiation exposure.

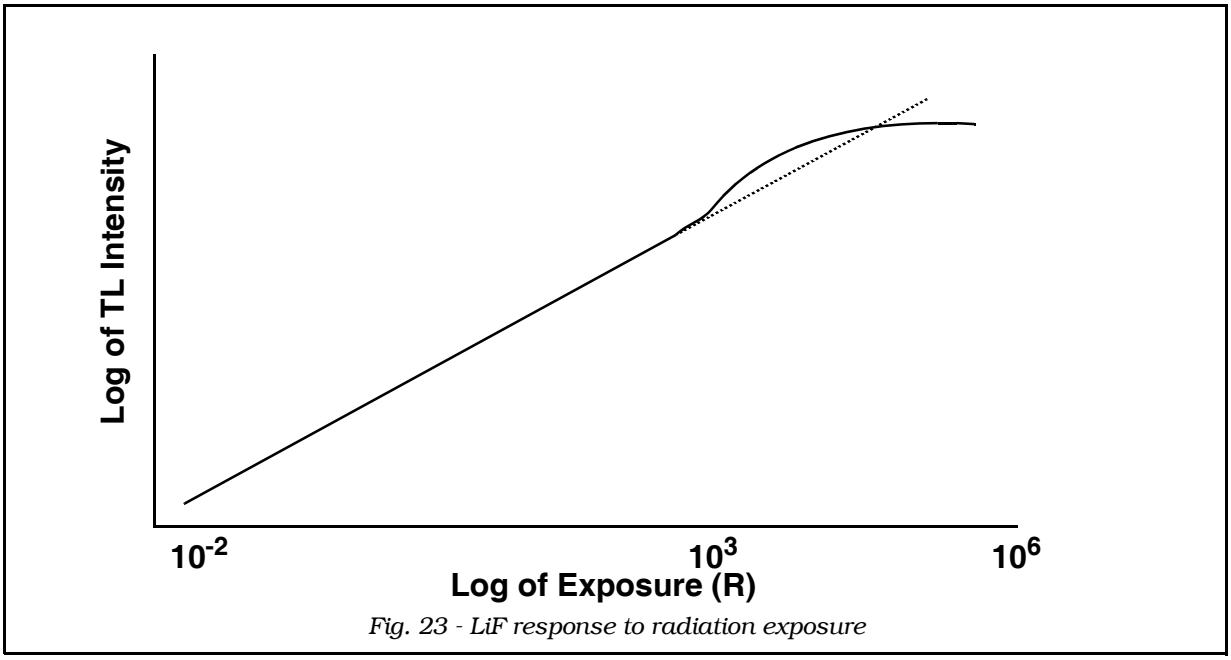
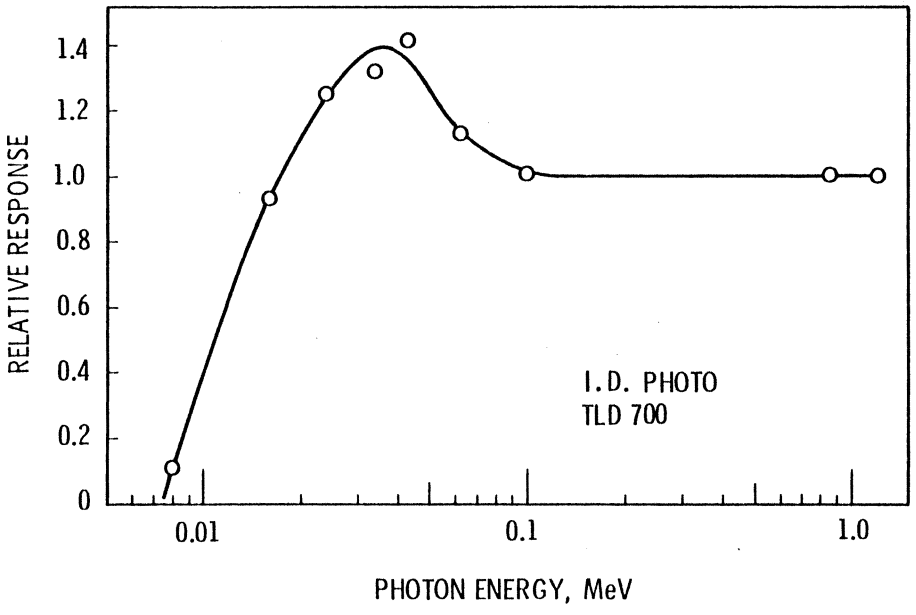


Fig. 23 - LiF response to radiation exposure

From the scale it is evident that LiF gives a linear dose response over the useful radiation protection range of 10 mR (10^{-2} R) to 1,000 R. This makes it easier to calibrate a TLD reader using LiF than a film badge system because "two points determine a line." Note that above 1,000 R the response goes "supralinear." This means the phosphor overresponds, with the light intensity output increasing faster than the exposure in this region. Practically, this means that the LiF phosphor can be used to measure exposures up to about a million R but a nonlinear calibration correction factor must be applied for exposures over 1,000 R. The single TLD phosphor thus covers

a much wider exposure range than either piece of film in a film pack, or even both films taken together.

As was done with film, the next major characteristic to be examined is the energy response. In the TLD case, this is given by a graph of the TL light intensity per unit exposure versus the energy of the photons. Such a plot is given in Figure 24 for LiF. At low energies, due to the photoelectric effect, the LiF phosphor overresponds since the effective atomic number is 8.2 for LiF. However, the overresponse is only about 30% compared to the 2,000 to 4,000% overresponse of film. At some facilities, the 30% overresponse at 10 keV is considered small enough to ignore for personnel dosimetry purposes. Other facilities employ filters in the holder to flatten the overresponse. Examples of both types of holders are shown later in this chapter.



Courtesy Hanford Laboratory

Fig. 24 - The photon response vs. energy (MeV) of LiF

The TLD phosphor is reusable upon heating to a high temperature. This process is called annealing. It is similar to the heating cycle used in reading the phosphor except that the temperature is raised higher and sustained for a longer period of time. For example, LiF is normally read while heating to 255° Celsius over a time span of about 20 seconds. It is often annealed at 400° C for one hour. This assures that all of the trapped electrons are released from the trap sites and have returned to the low energy valence band. After proper annealing, the phosphor has the same sensitivity as previously, so it can be reused as a dosimeter. Tests have shown that a given phosphor sample of LiF can be used and annealed over 1,000 times without loss of radiation sensitivity. This is an advantage over a film dosimetry system since a new film pack must be used each badge period.

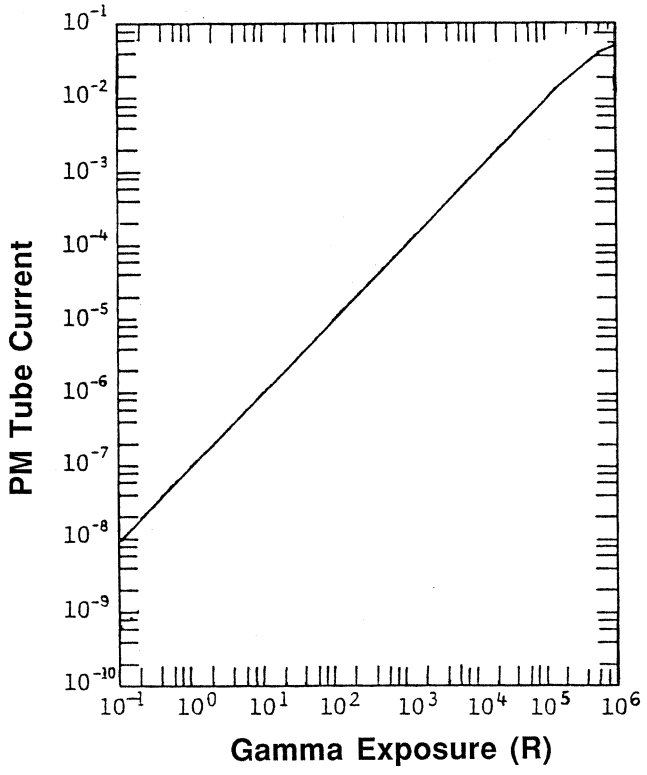
At the same time, the annealing ability also represents a disadvantage for the TLD badge. The act of heating the phosphor to read the dose information destroys that dose information permanently. Once the traps release their electrons, the information is lost. This can evolve into a major crisis when the badge has valuable dose

information. It might be the only easily recoverable badge following a major radiation accident. Loss of information occurs only upon failure of the reader due to component failure or loss of applied power during the read cycle. Some badge systems contain a backup phosphor to prevent this unlikely occurrence.

Even though the TLD phosphor itself is "reset" by the reading cycle and therefore, is not available as a permanent record like a piece of developed film, the glow curve from the TLD reader can be run out on a strip chart recorder to provide a permanent record. In the eyes of the law, a signed TLD personnel record sheet of readings from a TLD reader is considered to be a legal, permanent record of dose. The sheet carries equal weight to a piece of personnel film if there ever is a question of dose to a worker and the battle ends up in the judicial system.

Characteristics of Lithium Tetraborate

$Li_2B_4O_7$:Cu is one of a "second generation" of newer TLD phosphors pioneered by the Japanese. They set out to design a tissue equivalent phosphor with good TLD characteristics and they succeeded admirably. The phosphor was based on a similar material, $Li_2B_4O_7$:Mn, the manganese activated form introduced in 1965. However, the older form suffered from low sensitivity, rapid fading and absorption of moisture from room air. In 1981, the copper activated form was discovered and developed.



Courtesy of Panasonic

Fig. 25 - The radiation exposure response of lithium tetraborate

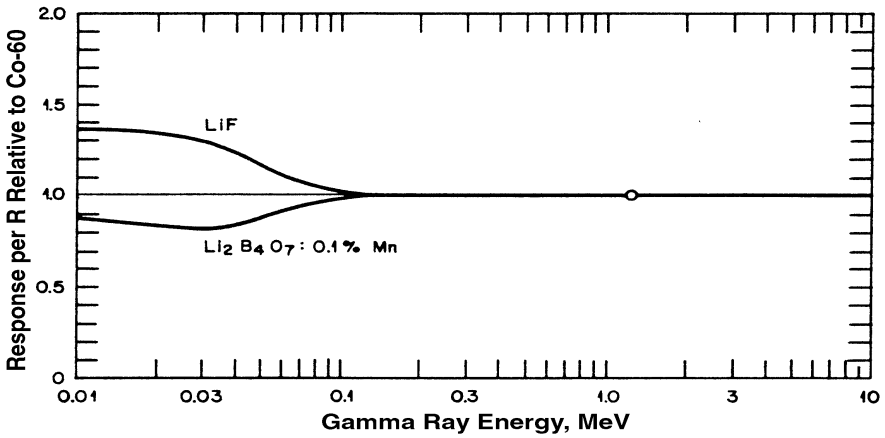


Fig. 26 - The energy response of lithium tetraborate

Courtesy of Frank Attix

The $Li_2B_4O_7$:Cu phosphor begins with a mixture of lithium carbonate and boric acid. It is melted at 950° C and then cooled and ground to a fine powder. The copper activator, dissolved in acetone, is then added and sintered to produce the final phosphor with 0.03% by weight copper. It has an effective atomic number of 7.3. The glow curve was presented earlier (Figure 19). Figure 25 shows the radiation response of $Li_2B_4O_7$:Cu. It has a linear response from 1 mR to about 100 kR. For comparison, the manganese activated form is linear only to 150 R.

The energy response of $Li_2B_4O_7$:Cu follows prediction – the Z_{eff} of 7.3 is slightly below the 7.5 of air and tissue so the phosphor under-responds at low photon energies. The energy response is flat to within ±10% from 40 keV to 7 MeV. This is

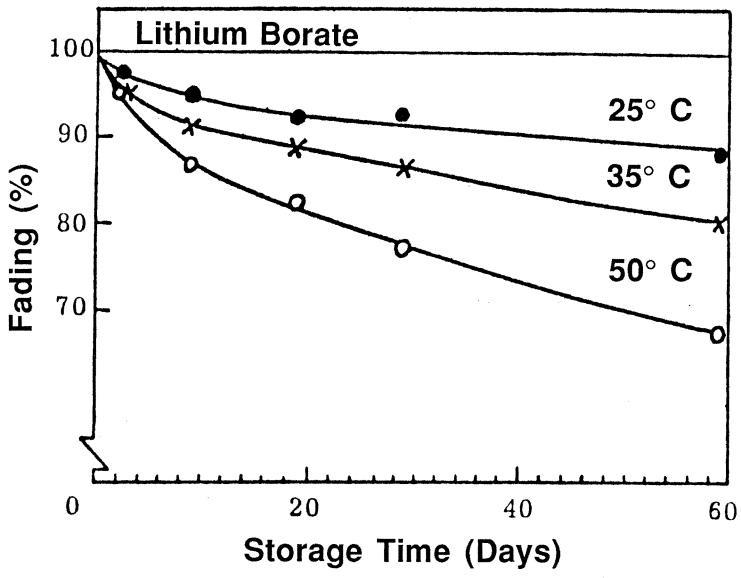


Fig. 27 - Fading characteristics of lithium tetraborate

Courtesy of Panasonic

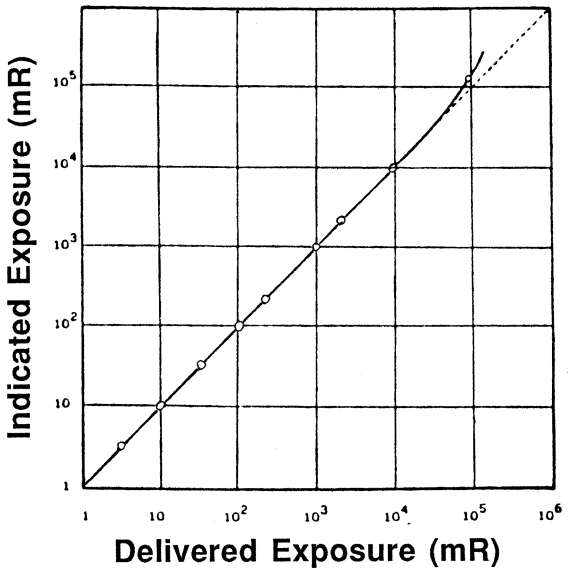
illustrated by Figure 26. (See Sample Problem 2). The response of LiF is shown for comparison. Finally, Figure 27 depicts the fading response of the lithium tetraborate phosphor. Results for three different storage temperatures are given.

Sample Problem 2

GIVEN:
A technician is exposed to 1 mSv of x-rays with an average energy of 30 keV. He is wearing a TLD badge with both LiF and lithium tetraborate phosphors.
FIND:
What dose will each phosphor read?
SOLUTION:
From Fig. 26, at 30 keV, LiF overresponds by 1.3 times so it would read 1.3 mSv. Lithium tetraborate underresponds by 0.82 times so that phosphor would read 0.82 mSv.

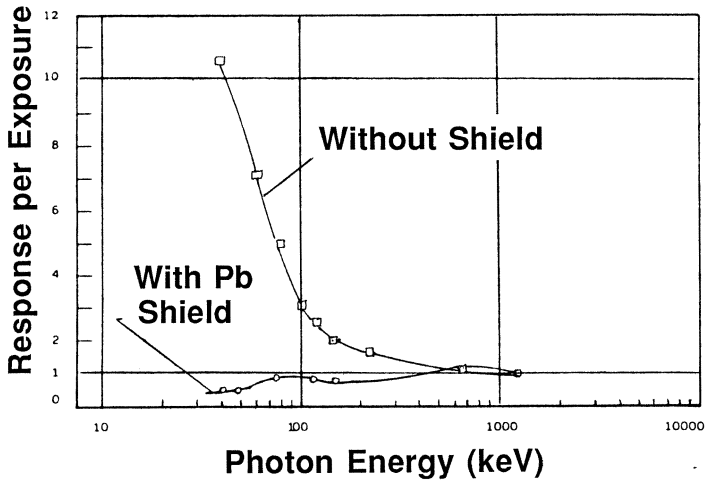
Characteristics Of Calcium Sulphate

Calcium sulphate with thulium activator, $CaSO_4$:Tm, is another of the second generation TLD Japanese-developed phosphors. The material has very high sensitivity and exhibits excellent fading characteristics. Due to the Z_{eff} of 15, it will overrespond at low energies. This was considered desirable since it could be paired up with $Li_2B_4O_7$:Cu phosphor to indicate the energy of the incoming radiation from the ratio of the readings. (This will be discussed shortly). The radiation response is displayed in Figure 28. Note that although the material is linear only to 50 R, it extends the low range of measurable dose to 0.1 mR. This dose can be read with a



Courtesy of Panasonic

Fig. 28 - The radiation exposure response of calcium sulphate



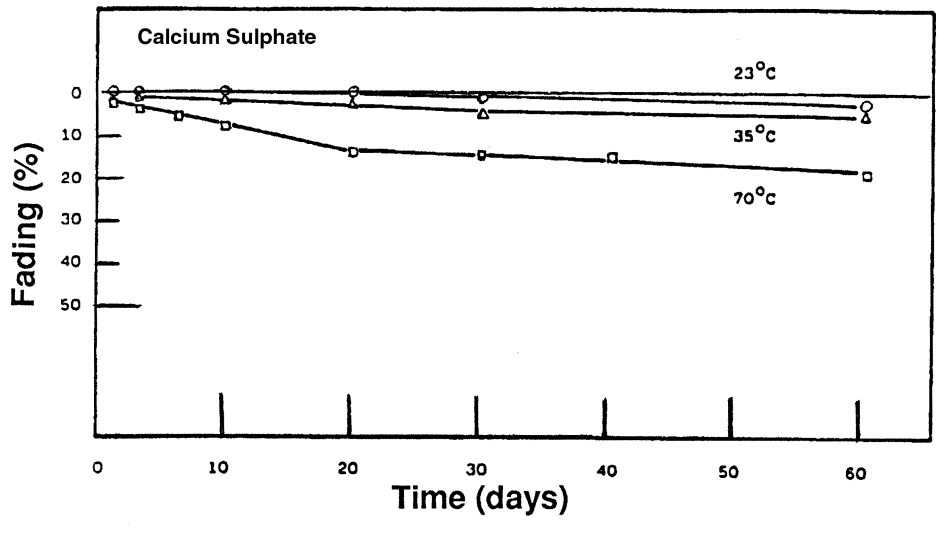
Courtesy of Panasonic

Fig. 29 - The photon energy response of calcium sulphate

standard deviation of ±20%.

With a high effective atomic number of 15, $CaSO_4$:Tm would be expected to overrespond at low photon energies. Such is the case, as shown by Figure 29. The overresponse can be corrected by shielding (filtering) the incoming gamma radiation with a high Z material before it reaches the phosphor. The response of $CaSO_4$:Tm under a lead shield is shown for comparison.

The fading characteristics are displayed in Figure 30. At normal room temperatures, the signal loss is very minor over the course of the normal badge period of one month.



Courtesy of Panasonic

Fig. 30 - The fading characteristics of calcium sulphate

	LiF	CaF_2	$CaSO_4$:Tm	$Li_2B_4O_7$:Cu
Density (g/cm^3)	2.6	3.18	2.7	1.6
Effective Z	8.2	16	15	7.3
Glow Peak (°C)	190	180	220	205
Rel. Sensitivity	1.000	30	70	?
Overresponse @30 keV	25%	1250%	1000%	-15%
Fading	5%/yr	16%/2 wk	1%/mo	10%/mo

Fig. 31 - A summary of the characteristics of popular TLD phosphors

The table in Figure 31 summarizes the characteristics of some of the more common phosphors. The higher sensitivity phosphors have application in environmental monitoring where sub-mrem dose equivalents are sometimes measured. Although they are more sensitive than LiF, they have a much poorer energy response, which is sometimes corrected by a specially designed filter holder.

In addition to being commercially produced in powder form, many TLD phosphors are available in several other physical forms. One of the most common forms is the extruded chip. Powdered microcrystals are pressed at high temperature and pressure into a solid chip. Lithium fluoride chips are 3 mm square and about 0.9 mm thick. They can be conveniently handled by tweezers (A special vacuum tweezer prevents surface scratches and chipping. These reduce sensitivity and are caused by using regular forceps on the sides of the chip). Other common geometric shapes include cylindrical rods and disks of teflon incorporating the phosphor.

Neutron Response Of TLD Phosphors

The characteristics given just above for the phosphors which contain lithium as part of their composition apply, strictly speaking, only to material in which the lithium atoms are present in their natural isotopic abundance: 7.5% Li-6 and 92.5% Li-7. It has been found that Li-6 has a large cross section (probability) for capturing thermal neutrons. The reaction is given in Figure 32. The large energy release is locally

$$^{6}_{3}Li + ^{1}_{0}n \rightarrow ^{3}_{1}H + ^{4}_{2}He + 4.8\ MeV$$

Fig. 32 - The lithium-6 thermal neutron capture reaction

deposited as both of the reaction products have very high stopping power. As an example, if LiF containing Li-6 isotope is exposed to a thermal neutron field, the phosphor will indicate a radiation dose due to the neutron capture reaction. On the other hand, if the LiF contains no Li-6, the phosphor will not show a dose upon exposure to thermal neutrons. Fortunately, such special phosphors with different isotopically separated lithium isotopes are commercially available. The phosphor containing

mostly Li-6 is called TLD-600 by Harshaw, and the one containing mostly Li-7 is referred to by the trade name TLD-700. Lithium fluoride with the natural abundance of lithium isotopes is known by Harshaw's trade name TLD-100. The isotopic ratios for these phosphors are listed in the chart of Figure 33.

Phosphor	% Li-6	% Li-7
TLD-100	7.5	92.5
TLD-600	95.6	4.4
TLD-700	0.07	99.93

Fig. 33 - Isotopic ratios for Harshaw phosphors from Bicron NE

To measure thermal neutron fields, combinations of TLD-600 and TLD-700 are used together. The TLD-600 reading will be due to thermal neutrons + any gamma rays (which interact identically in Li-6 and Li-7 fluoride) while the reading in the TLD-700 will be due exclusively to gamma rays. By subtracting the TLD-700 reading in mrem from the TLD-600 reading, the dose difference is due only to the thermal neutrons. Note that TLD-600 does not read directly in millisieverts for thermal neutrons. It has been shown by investigators that the TLD-600 – TLD-700 difference (expressed in mSv of gamma equivalent exposure) must be multiplied by an energy dependent conversion factor (expressed as mSv of thermal neutrons per mSv of gamma equivalent) to give the correct dose equivalent (in mSv) of thermal neutrons. The manufacturer suggests a factor of 0.006. See Sample Problem 3.

$Li_2B_4O_7$:Cu is also commercially available with altered Li-6 and Li-7 concentrations. In addition, boron enriched in ^{10}B is used in the neutron sensitive phosphor while the corresponding neutron insensitive phosphor contains only ^{11}B. These special phosphors are available from Panasonic Industrial Co., U.S. representatives for the Japanese developers Matsushita Electric Corp. The response of the Panasonic phosphors in their neutron badge is depicted in Figure 34.

Sample Problem 3

GIVEN:
The phosphors from a neutron badge exposed to a mixed field give the following readings in mSv of gamma ray equivalent dose: TLD 600 = 15.1 mSv and TLD 700 = 2.2 mSv. Assume a conversion factor of 0.006.
FIND:
What thermal neutron and gamma doses did the badge receive?
SOLUTION:
The gamma dose is the TLD 700 reading, i.e., = 2.2 mSv (220 mrem). The difference between the phosphor readings = 15.1 – 2.2 = 12.9 gamma mSv. This difference x 0.006 thermal neutron mSv/gamma equivalent mSv gives the thermal component, i.e., 12.9 x 0.006 = 0.077 mSv (7.7 mrem) thermal neutron dose.

Radiation Type	$^{7}Li_2{}^{11}B_4O_7$	$^{6}Li_2{}^{10}B_4O_7$
Thermal neutrons	Insensitive	Very Sensitive
Epithermal neut.	Insensitive	Sensitive
Fission neutrons	Insensitive	Insensitive
Gammas	Sensitive (Equivalent)	

Courtesy of Panasonic

Fig. 34 - Neutron response of Panasonic phosphors

TLD Badge Systems

Figure 35 shows some badge holders designed for LiF extruded chips. None of these provides an energy correction for photons. The large student badge (in the lower center of the Figure) was designed for university students working with radiation. It provides a simple twist-open configuration for chip removal and was purposely

Fig. 35 - Some examples of LiF TLD badge holders

designed to be large in size to facilitate recovery at the end of the class period. The ring badge (in the upper right) is used for "extremity monitoring," i.e., measuring the dose to the hands. It holds a TLD chip and is adjustable to most finger sizes. It has little negative effect on finger dexterity. As an example, the ring badge can be worn under surgical gloves to measure doses to a radiologist implanting radium needles.

The other three-chip badge holder is a commercial design. It has two chips shielded by 285 mg/cm^2 and a third behind a 7 mg/cm^2 filter. This latter chip reads the shallow dose equivalent (skin dose) while the former give the Deep-dose Equivalent (whole body dose).

Without doubt, the most popular personnel dosimetry system among U.S. nuclear power reactor sites is the Panasonic system consisting of UD-800 Series badges and a UD-710 Series automated reader. Both the badge and the reader incorporate several radical design innovations.

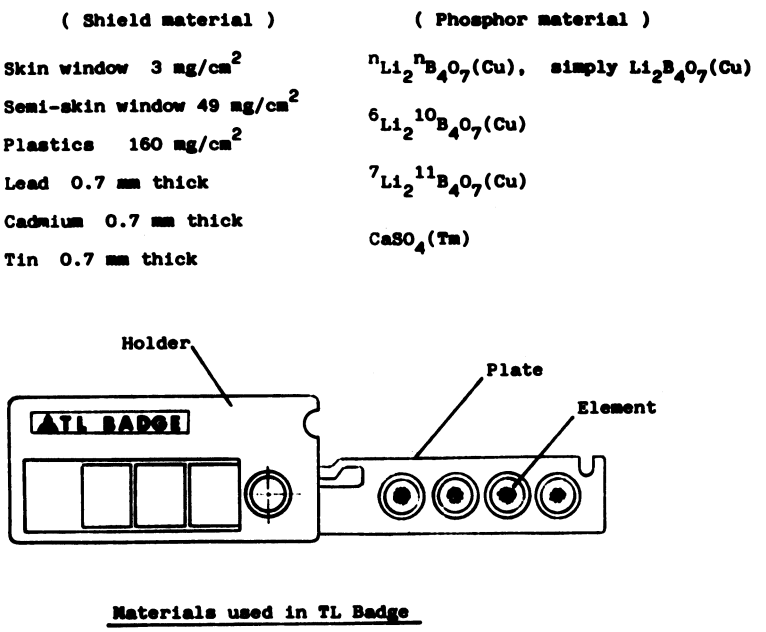
A typical UD-800 series badge is shown in the photo of Figure 36. It consists of a plastic case and clip for attachment to clothing, a hollow plastic block containing a series of punched holes for identification coding and filters, and an insert plate holding the four TLD phosphors. A detailed drawing of the hollow holder block and plate is presented in Figure 37.

The complete dosimeter is relatively small. It is designed for high accuracy, wide dose range, and ruggedness. The insert plate holds four phosphor elements. The first element is for measurement of shallow dose equivalent, H_S ("skin dose"). The phosphor has a density thickness of only 15 mg/cm^2, is 3 mm in diameter and has a mass of only 0.8 milligrams. It is mounted on a carbon-coated plastic substrate of 11 mg/cm^2 density thickness. It is covered with a teflon and polyethylene transparent window 28 mg/cm^2 thick (see sketch in Figure 38). The window effectively seals the phosphor against intrusion of dust, moisture and sweaty fingers!

Ideally, the ICRP recommends skin measurement at a depth of 7 mg/cm^2. The

Courtesy of Panasonic

Fig. 36 - The Panasonic UD-800 series TLD badge



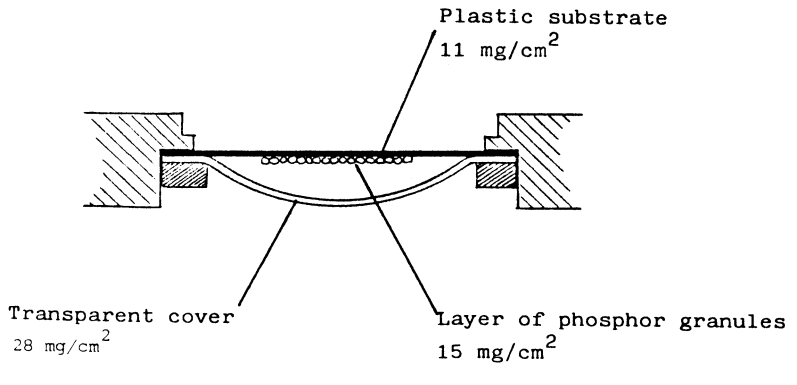
Courtesy of Panasonic

Fig. 37 - Construction details of Panasonic badge

UD-800 badge has an overlying thickness of 14 mg/cm^2. However, the lithium tetraborate is almost tissue equivalent. Panasonic claims that beta doses to the skin can be accurately measured for beta energies over 190 keV.

The second TLD element in the UD-800 badge is also $Li_2B_4O_7$:Cu. It is shielded by 300 mg/cm^2 and is used for measuring the Deep-dose Equivalent, H_d. Recall that H_d is defined at 1,000 mg/cm^2 depth. This is accomplished in the UD-800 badge by Panasonic adding 700 mg/cm^2 of plastic projecting out the front of the hanger case overlying element number 2.

The third and fourth elements use $CaSO_4$:Tm phosphor. The third element is shielded only by the thin plastic wall of the hollow block and hanger case. The fourth



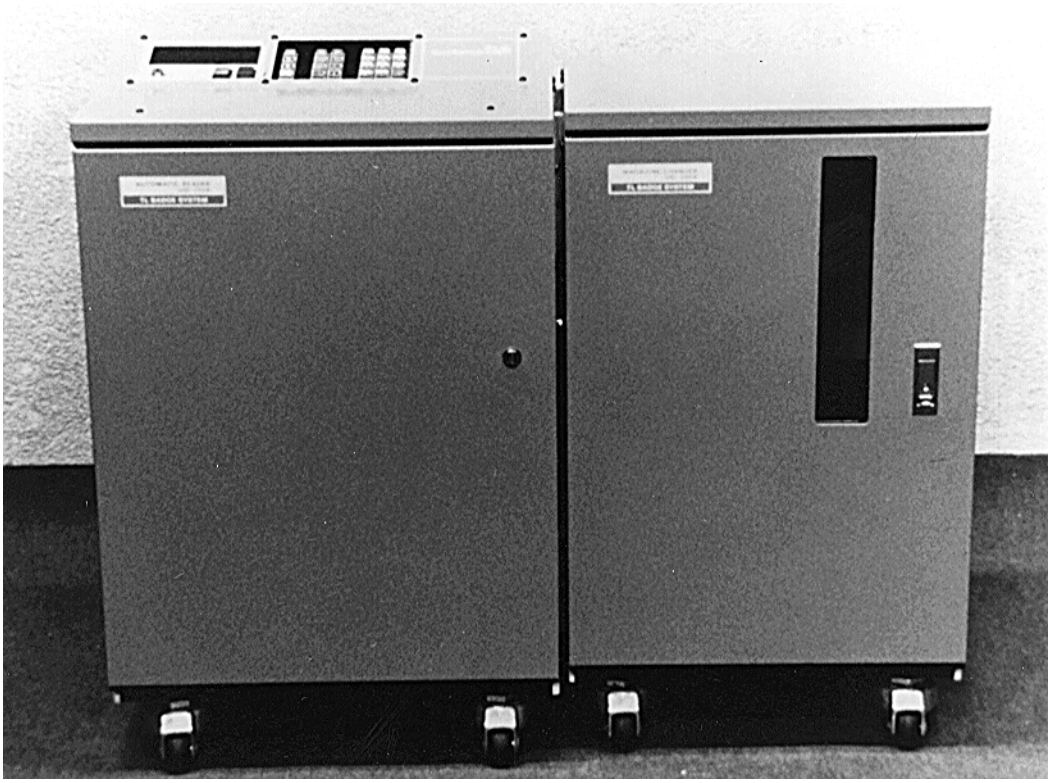
Courtesy of Panasonic

Fig. 38 - Detail of a Panasonic badge element

is shielded by 700 mg/cm^2 of lead. The lead flattens the energy response. The high sensitivity calcium sulphate is used in two ways. Since the unshielded third element overresponds at low photon energies, it can be used in conjunction with the shielded fourth element to estimate the photon energy. By calculating the ratio of the recorded doses under the 3rd and 4th elements, the average energy of the photons exposing the badge is determined in exactly the same way as use of filter ratios in film badge processing. Finally, the high sensitivity $CaSO_4$:Tm is used to extend the dose range of the dosimeter down to 0.1 mrem (1 μSv). This low dose can be read with a standard deviation of only ±20%.

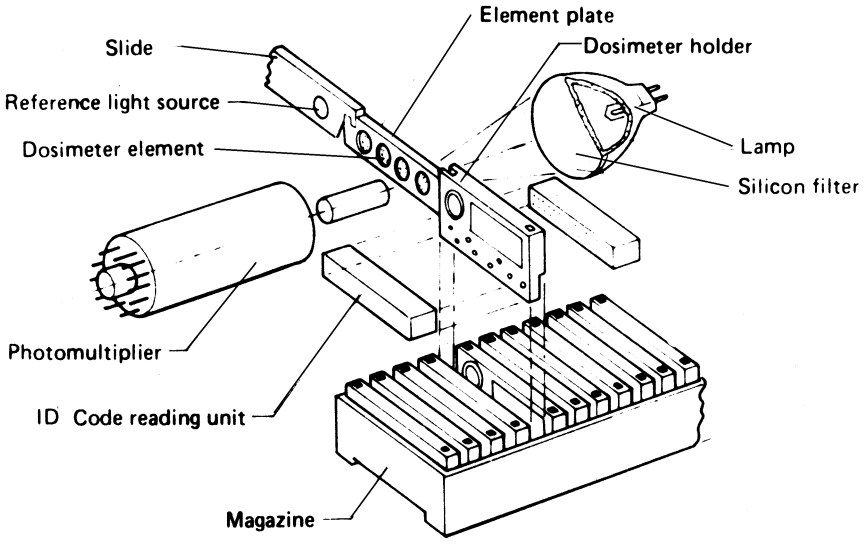
Figure 39 shows the Panasonic UD-710 series automated reader for the UD-800 family of badges with its accompanying magazine changer. Each magazine holds 50 dosimeters and the changer has a capacity of 500 dosimeters. The reader requires 20 seconds per badge for preheating, reading and annealing! The 500 badge capacity can be processed in 3 hours.

One of the more unique features of the reader is the optical heating system. As a direct result of the very low mass per phosphor element (0.8 mg), the temperature can be raised rapidly with relatively low heat input. Panasonic employs a tungsten flash lamp as the heater. The infrared light pulse is directed on the black carbon-coated substrate behind the phosphor. A complete read cycle consists of 3 light pulses. The 1st pulse preheats the phosphor to 120° C in about 0.2 second. The 2nd pulse raises the temperature to 300° in 0.8 second for reading the dose information.



Courtesy of Panasonic

Fig. 39 - The automatic Panasonic reader



Courtesy of Panasonic

Fig. 40 - Panasonic optical reader details

Finally, the 3rd pulse pushes the temperature to 350° to anneal the badge for reuse. A sketch of the optics and mechanical components in the reader is provided in Figure 40.

Additionally, the reader contains a dedicated microcomputer. This component performs self-tests on the reader, controls the read/anneal cycle, interprets the ID coding on each badge and automatically adjusts the sensitivity calibration factor by comparison with an internal reference light source. Numerical data can be displayed on the control panel and transmitted to an external computer or printer.

A few other commercially available badge systems will now be covered. Figure 41 shows a special packaging technique. It is designed to prevent one of the most common reasons for TLD sensitivity change – handling of the chip which introduces foreign substances onto the chip surface. If any stray substance is on the chip when it is heated for reading or annealing, it will be baked into the phosphor and will alter the sensitivity of that chip. Badge processors use ultrasonic cleaning baths between uses to reduce this effect. By sealing the chip inside an evacuated glass tube, the handling problems are virtually eliminated. The phosphor will not come in contact with either tweezers or sweaty hands. The disadvantage, besides cost, is that the chip can't be heated through the vacuum. Therefore, the glass bulb dosimeter has a built-in heater ribbon. The TLD reader supplies an electric current through the heater ribbon which causes the chip to read out. The annealing cycle is also accomplished by use of the internal heater.

The next photo, in Figure 42, shows the holders designed for use with the two bulb dosimeters of the previous figure. The dosimeter on the left was designed for environmental monitoring. It uses a CaF_2 phosphor in chip form. Since this is one of the phosphors that overresponds to low

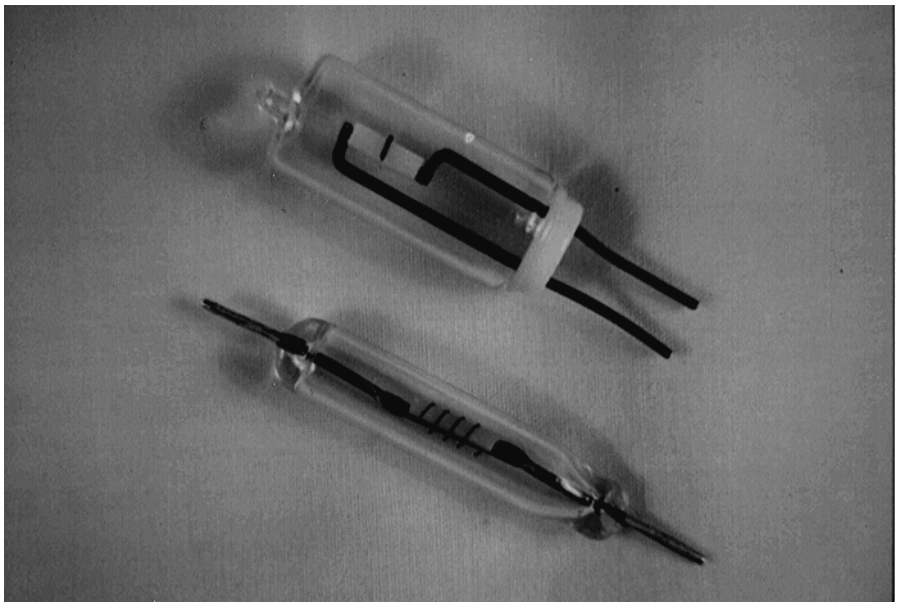


Fig. 41 - Some TLD bulb dosimeters

photon energies, an "energy compensating shield" is provided which uses different thicknesses of three different metal filters to flatten the energy response of the encapsulated dosimeter. The dosimeter on the right was used in the U.S. Navy nuclear submarine program. It, too, uses the calcium fluoride phosphor. The overresponse is again corrected by use of internal metal filters. In addition, as shown in the inset in the photo, the holder cap functions as a criticality dosimeter.

Figure 43 shows a TLD badge system using the phosphor incorporated in a teflon matrix. This particular phosphor was designed to be the same physical size as a Kodak film pack. Thus, the customers could still retain the old film badge holder as a cost savings device. Note that the

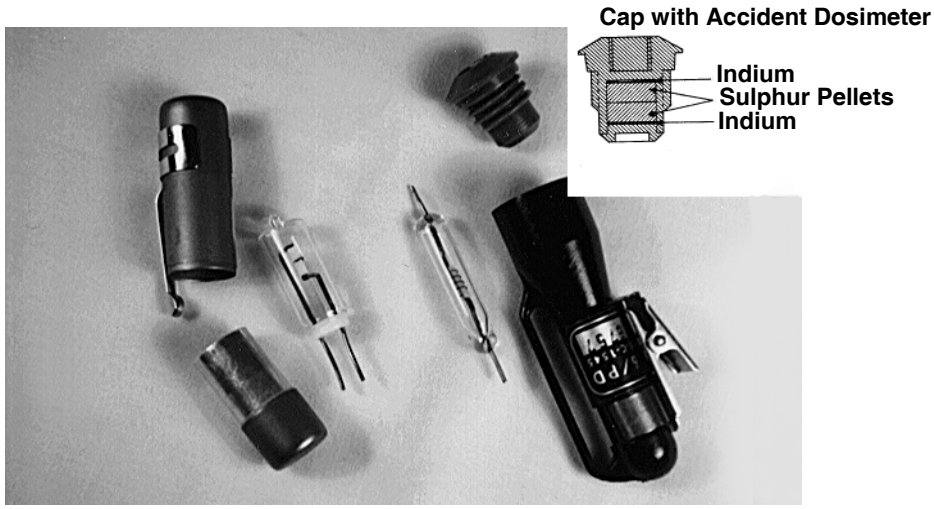


Fig. 42 - Badge holders for bulb dosimeters

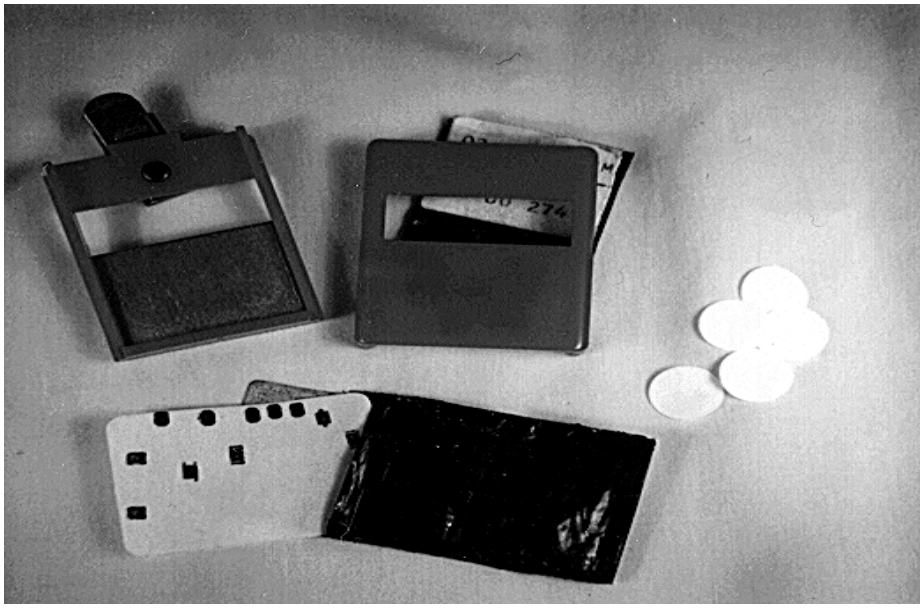


Fig. 43 - A teflon matrix TLD system

dosimeter element has coded magnetic numbers to allow for automatic processing of the badges. Also shown are some teflon disk dosimeters with embedded phosphor.

Another solution to the chip handling problem was the design by Harshaw in which two TLD chips were sealed between layers of teflon plastic and then attached to a small metal plate. An employee number could be encoded on the plate to allow for automatic processing with computer interfaced data storage. The chips were heated THROUGH THE PLASTIC COVER and so the costly internal heater is eliminated yet the package prevented foreign matter from contacting the chip and eliminated tweezer handling fatalities., i.e., major loss of sensitivity due to breaking off chunks of phosphor by grabbing the chips too forcefully. See Figure 44.

Before leaving TLD systems, some quality control problems unique to TLD will be discussed. As mentioned earlier, the TL phosphor properties result from VERY CAREFULLY CONTROLLED ADDITION OF ACTIVATOR ATOMS to the batch during manufacturing. These doped atoms are in the parts per million range. Thus, small changes in concentration from batch to batch will produce phosphors with different radiation sensitivity. It is necessary to either purchase new phosphor with a sensitivity matched to the old or to physically keep separate different batch number phosphors. Commercial suppliers measure and record each batch sensitivity so matched dosimeters can be supplied. Also, handling can cause corners to break off from chips. This reduces the chip sensitivity due to the loss of that part of the total mass. Prudent TLD badge processors will periodically sort the dosimeters by exposing them all to an identical test exposure and then reading them. By discarding dosimeters that read too high or too low, a set with uniform sensitivity is maintained. Another potential problem with LiF is that exposure of the phosphor to doses approaching several hundred rem total with multiple read/anneal cycles seems to reduce the chip sensitivity. Thus, some TLD processors discard chips exposed to those levels.

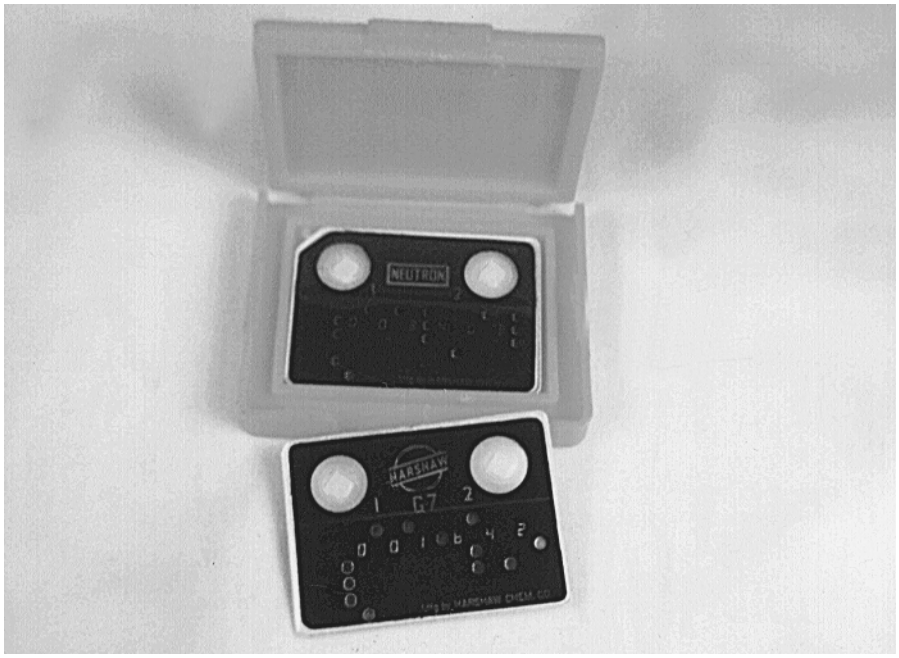


Fig. 44 - A different TLD packaging technique

Hybrid Badges

There are some badge systems which incorporate both film and TLD. Some facilities prefer to make use of the advantages of both dosimeter types. Others are in the process of converting from one system to the other and want to document the relative response of each system. Figure 45 shows a badge designed by the Comision Nacional de Energia Atomica of Argentina. The top section contains filters and two film packs (neutron track emulsion + regular emulsion) and the bottom section has compartments for different TLD chips to read thermal neutron and gamma/X-ray dose equivalents. An additional hybrid badge system will be shown under the criticality badge section later in this Chapter.

Optically Stimulated Luminescence Badge Systems

Principle of OSL

Optically Stimulated Luminescence, discovered in the 1950s, refers to the release of light of a certain wavelength during the exposure of a phosphor to a different wavelength of light. It is caused by trapped electrons being excited and then

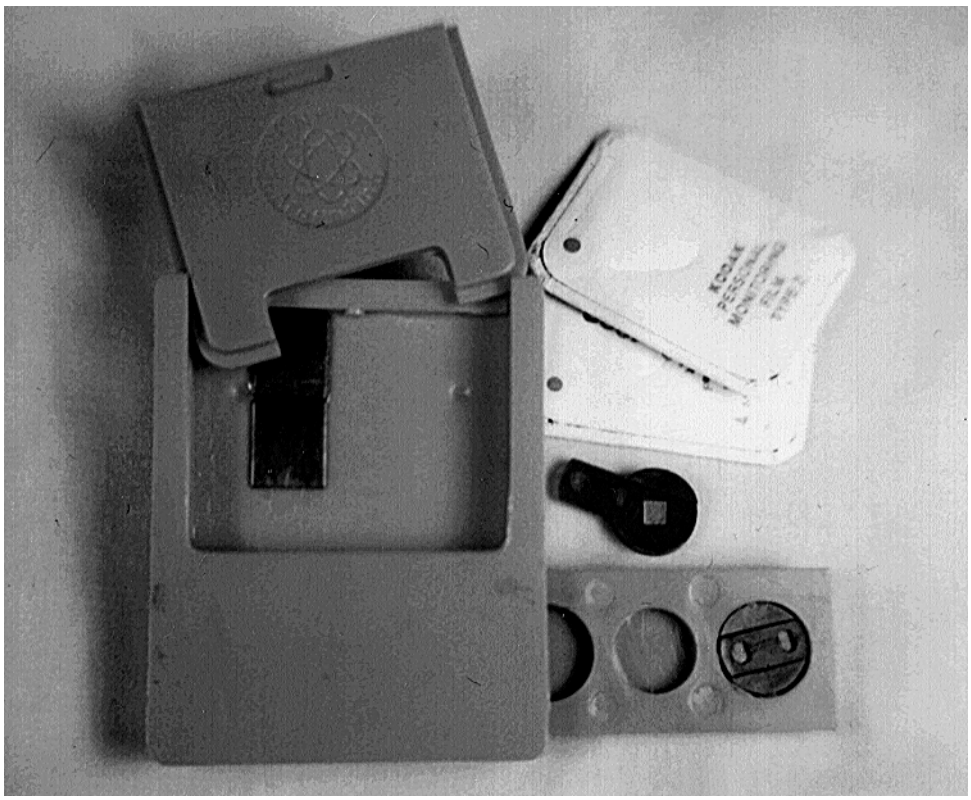


Fig. 45 - A hybrid badge using film and TLD

releasing their excitation energy as the luminescence signal. Many TL phosphors will exhibit OSL under the right conditions. The principle is quite similar to thermoluminescence except that in the case of OSL, the luminescence is produced by a light beam rather than through heating. Also, in contrast to TLD, with a proper choice of phosphor, the electrons released from the traps by the stimulating light only represent a tiny fraction of the total in the traps. These two differences can lead to some big advantages over TLDs. On the downside, early work with OSL did not discover a phosphor with good fading characteristics and with reasonable sensitivity for radiation protection purposes.

The elimination of the heating requirement for OSL vs. TLD means that it is much easier to package a phosphor that doesn't have to withstand repeated high temperature cycles. Also, there is no need to bathe the phosphor in an inert gas to eliminate room oxygen during the read cycle as is necessary with a TLD reader. The fact that stimulated electrons return to the traps means that the DOSE INFORMATION IS NOT LOST BY READING. To re-read an OSL dosimeter, it is merely necessary to place it in the reader and process it normally. As a practical matter, it is also possible to anneal the OSL phosphor so that it is reusable.

The phosphor used by Landauer, Inc. of Glenwood, IL is specifically manufactured internally for OSL use. The aluminum oxide is doped with a carbon impurity to produce stable electron traps that appear in the energy gap inside the crystalline

structure. The reading process involves exposing the phosphor to a short series of ten pulses of green laser light. After each pulse, the blue luminescence is recorded by a photomultiplier tube. The total intensity recorded for the pulse series represents the reading. This technique is called pulsed OSL or POSL.

Characteristics of Aluminum Oxide

In 1998, Landauer introduced their Luxel® OSL based dosimetry system using aluminum oxide, Al_2O_3, for the phosphor. This material provides dose information for both beta particles and gamma rays. The luminescence is stimulated using a laser light source inside the reader.

The radiation response of Landauer's dosimeter is shown in Figure 46. Note that the phosphor is highly linear. This is due to the use of laser light to excite the phosphor rather than heating to a high temperature as in the TLD case.

Since the effective atomic number of aluminum oxide is about 11, the photon energy response would be expected to show a small over-response for low photon energies. Figure 47 shows the pertinent graph of the gamma ray response of the Luxel® phosphor. It over-responds by about 2.8 times at 60 keV. This phosphor also shows extremely good fading characteristics. Measurements by Landauer show no measurable drop in signal over a six month period.

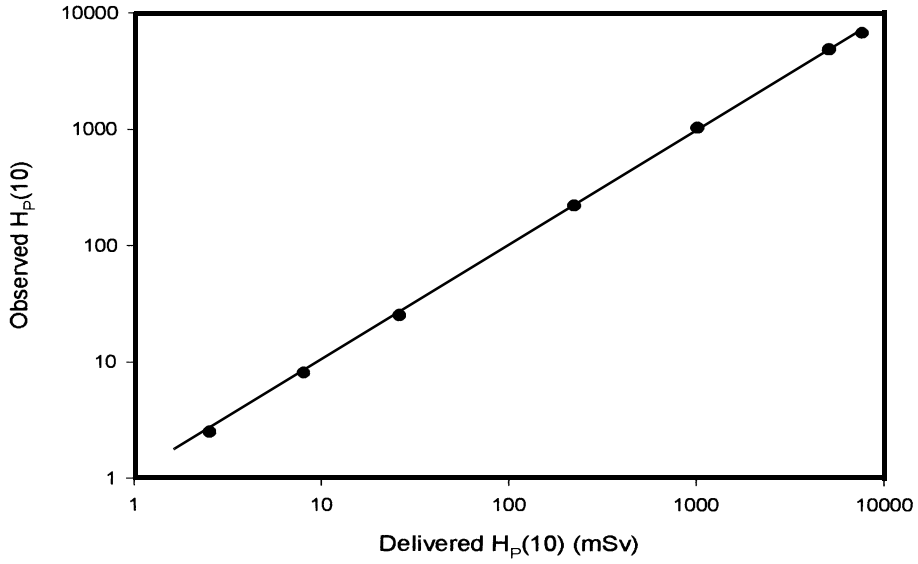
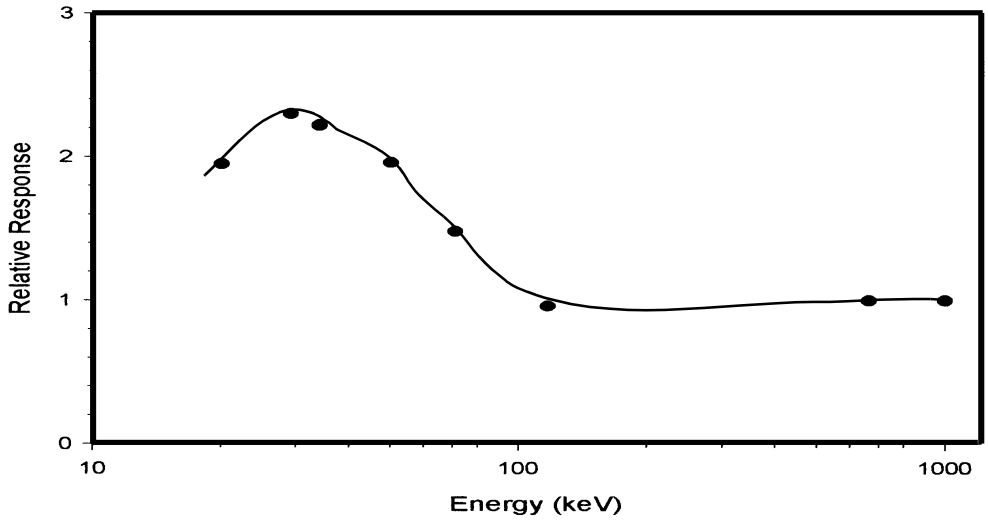


Fig. 46 - The radiation response of aluminum oxide OSL phosphor

OSL Badge Systems

World-wide, OSL badges account for about a quarter of the total badges in use. A photo of the Landauer Luxel® badge, without the badge holder, is shown in Figure 48. The polyester sealed powder package at the left of the photo is the aluminum



Courtesy of Landauer, Inc.

Fig. 47 - The photon energy response of aluminum oxide OSL phosphor

oxide phosphor spread over an active area of ½ square inch. In the badge, the phosphor is sandwiched inside the hexagonal filter pack with four filtered regions available for analysis. The open window provides filtration of 20 mg/sq cm and is sensitive to beta particles above 150 keV as well as low energy photons. The solid copper and tin filters provide 182 and 372 mg/sq cm of density thickness respectively. The deep dose equivalent, lens of the eye dose and the shallow dose equivalent are calculated using the luminescence intensities under the three filters. Since the phosphor is sensitive to room light, the phosphor and filter package are heat sealed inside a rugged light-tight packet and then sealed inside the hexagonal plastic case shown at the right side.

Courtesy of Landauer, Inc.

Fig. 48 - The Luxel® aluminum oxide OSL personnel dosimeter

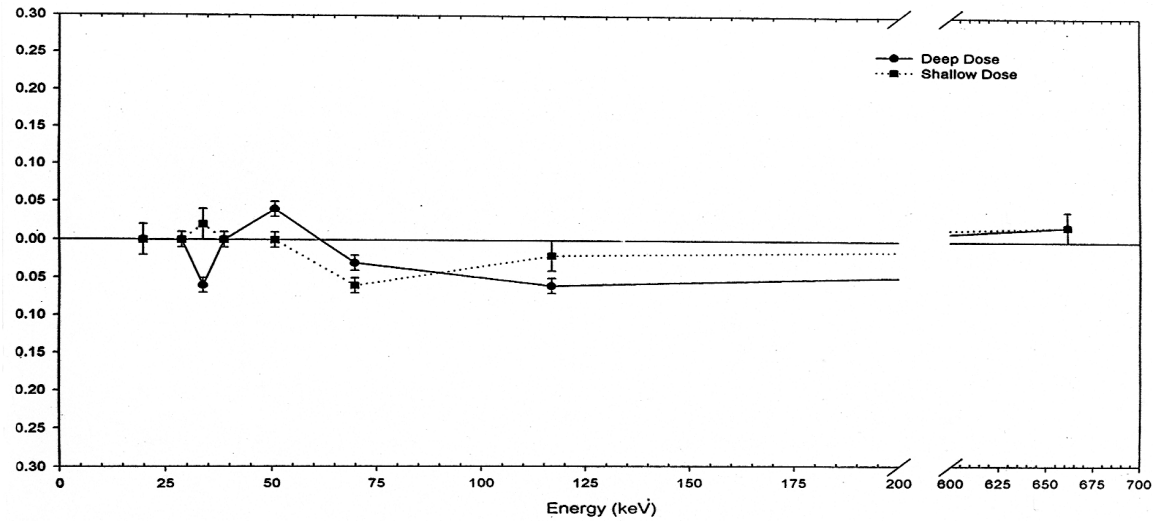


Fig. 49 - The energy response of the complete Luxel® dosimeter badge

Figure 49 shows a gamma energy response curve for the complete dosimeter, including the filters. The overresponse noted in Figure 47 for the bare phosphor has been eliminated by the inclusion of the filter package.

The Landauer Luxel® badge has one additional filter, a square copper grid with 1 mm holes. By mapping an image of the OSL signal passing through the grid holes, abnormal exposure conditions can be detected. During normal occupational exposure, the badge is exposed from many different angles as the worker moves around in the radiation workplace. Thus, the image of the tiny grid holes is smeared out. In contrast, if the badge is maliciously exposed by placing it near a strong radiation source, the grid holes will be imaged as sharp peaks. This very useful accident investigative technique has been used for years with film badges but is not available to TLD users. So, the OSL badge has another advantage over TLD.

In 2003, Landauer, Inc. and Panasonic joined forces to produce the InLight™ System for large dosimetry processors, e.g., nuclear power plants or government labs. The system uses the Panasonic case, holder and plate shown in Figure 36, but the TLD phosphors are replaced by Landauer's aluminum oxide OSL phosphor. The InLight™ System has several advantages. Badges don't have to be annealed, the sensitivity is linear from 1 mrad to 1,000 rad, badges are re-readable and no purge gas is needed to remove thermoluminescent atmospheric oxygen during the read cycle. The automatic read cycle is reduced from 20 seconds per TLD badge to only 13 seconds per OSL badge using continuous wave OSL (CW-OSL) so throughput is raised to 280 badge readings per hour. Figure 50 shows the InLight™ System 200 badge capacity automatic reader. Typically, the InLight™ badge is issued for a 12 month period. The employee turns it in twice a month for reading and then gets a new badge for the next year. The differences in dose during the semi-monthly readings gives the increment of dose for that 2 weeks. The final year end reading gives the annual cumulative dose. Lab measurements have shown that the 24 readings of the badge deplete the OSL signal by only 10% over the year.

But what about the smaller licensee's needs? Landauer has this covered too. In 2005, they introduced the microStar, to "Go where no reader has gone before." This unit is small and lightweight. It provides dosimeter readings in 13 seconds, making it especially useful for nuclear emergency operations. As with the larger capacity InLight™ System, there is no annealing, no purge gas, and badges are re-readable. Figure 51 shows the microStar unit which can be attached to a PDA for readout of dose information. The badge plate is inserted manually and the four phosphors are read sequentially by rotating the knob.

Courtesy of Landauer, Inc.

Fig. 50 - The InLight™ automatic OSL badge reader

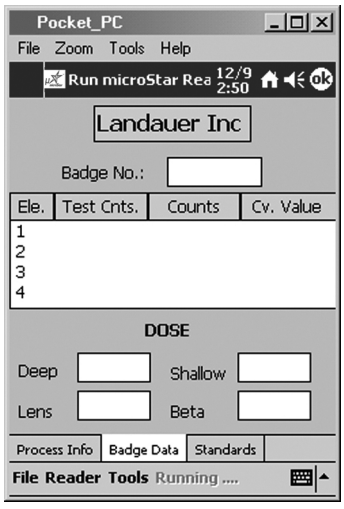
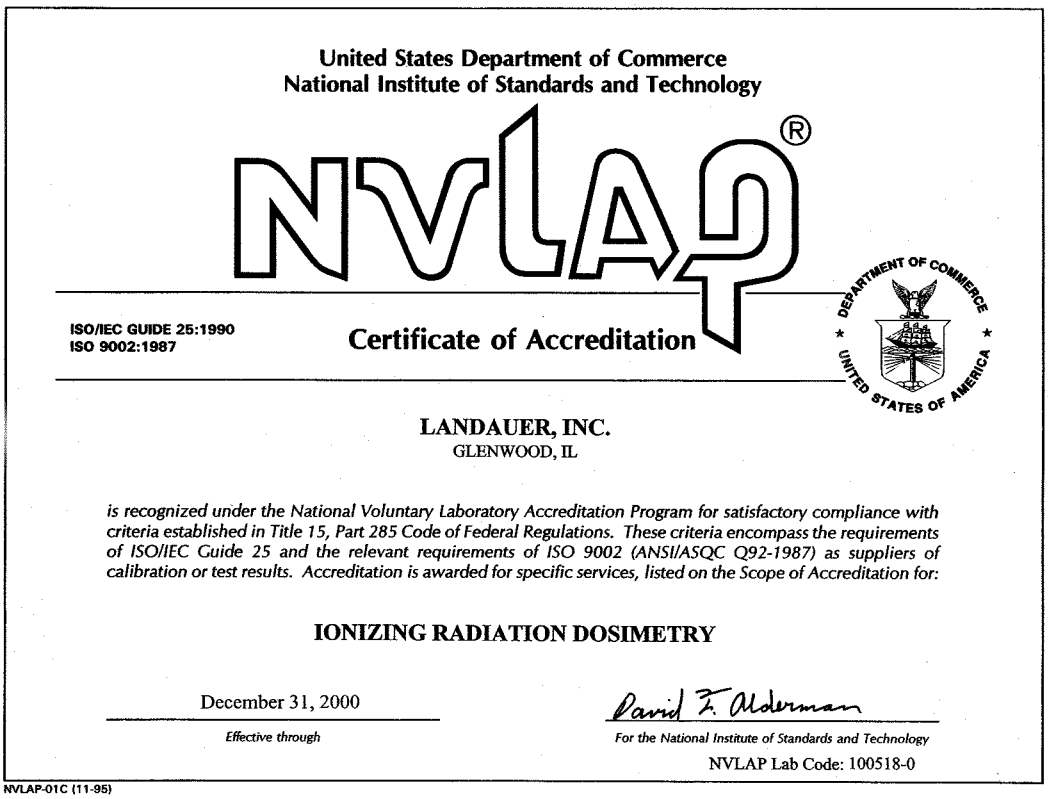


Fig. 51 - The microStar portable OSL badge reader

Radiation Badge Performance Testing

World-wide, there are about 5 million radiation badges in use. In years past, there were many complaints from users about the accuracy (or, rather, the lack of it) of the badge processors. In the early 1980s there were well over 100 organizations in the U.S. which were processing personnel radiation badges. In order to compete profitably, costs must be reduced to the bare bones. This means that quality assurance programs are often cut or eliminated.

In 1973, the Conference of Radiation Control Program Directors (a group of Federal and State regulatory personnel) set up a task force to establish a testing program of U.S. processors. In 1975, the ball was passed to the Health Physics Society which set about writing a draft standard in cooperation with users, processors and regulators. In 1978 this work resulted in publication of Draft American National Standard N13.11 (ANSI N13.11). As a result of two different pilot tests, numerous revisions were made in the draft and a final manuscript for ANSI N13.11 was submitted to the printer in1982. An NRC-sponsored pilot study was conducted in 1981 and 1982 to allow processors to voluntarily participate in a trial run. In 1987, the NRC amended the Code of Federal Regulations to require all personnel badge processors offering commercial services to be accredited by the National Voluntary Laboratory Accreditation Program, NVLAP. A sample NVLAP certificate is exhibited in Figure 52. The National Institute of Standards and Technology administers the NVLAP program.



United States Department of Commerce
National Institute of Standards and Technology

NVLAP®

ISO/IEC GUIDE 25:1990
ISO 9002:1987

Certificate of Accreditation

DEPARTMENT OF COMMERCE ★ UNITED STATES OF AMERICA ★

LANDAUER, INC.
GLENWOOD, IL

is recognized under the National Voluntary Laboratory Accreditation Program for satisfactory compliance with criteria established in Title 15, Part 285 Code of Federal Regulations. These criteria encompass the requirements of ISO/IEC Guide 25 and the relevant requirements of ISO 9002 (ANSI/ASQC Q92-1987) as suppliers of calibration or test results. Accreditation is awarded for specific services, listed on the Scope of Accreditation for:

IONIZING RADIATION DOSIMETRY

December 31, 2000
Effective through

David F. Alderman
For the National Institute of Standards and Technology

NVLAP Lab Code: 100518-0

NVLAP-01C (11-95)

Courtesy of Landauer, Inc.

Fig. 52 - A sample NVLAP certificate

Category	Percent of Services Passing	
	FILM	TLD
Gamma Rays	43%	63%
High Energy X-ray	58%	68%
Low Energy X-ray	49%	57%
Beta Rays	44%	67%
Neutrons	33%	46%
Gamma + X-ray	70%	81%
Gamma + Beta	63%	69%
Gamma + Neutron	48%	48%
OVERALL AVERAGES	51%	62%

Fig. 53 - Percentage of badge processors passing, by badge type, BEFORE final ANSI Standard

A similar program, for the U.S. Department of Energy and for DOE contractors, is called DOELAP.

The results of the early pilot test runs provided some insight into the state of the art in personnel dosimetry. The second pilot study tested performance in eight different radiation categories. A total of 59 different U.S. processors participated. The table in Figure 53 shows the percentage of film and TLD processors passing in each category. As is evident from the figure, only about half the film badge processors could meet the standard. The TLD processors did only marginally better.

Under the current NVLAP program, testing is done to American National Standard, ANSI N13.11-2009. Since 1986, Pacific Northwest National Laboratory in Richland, Washington has been the official Proficiency Testing Laboratory for the NVLAP program. Since the 2009 revision of the standard, there are five "Test Categories," one for the "accident range" which is defined as "high levels of deep absorbed dose resulting from uncontrolled conditions" and four categories for the "protection range" defined as "shallow and deep dose equivalent for the purpose of ascertaining the effectiveness of radiation protection measures." Test doses for the accident range are between 5 and 500 rads while test conditions for the protection range are below 30 rem. The table in Figure 54 lists the current categories, the test radiations and the passing criteria. A category is passed if the square root of the sum of the standard deviation squared (accuracy) plus the bias squared (precision) of the test badge results is less than the table value. (Sample Problem 4.) This change in the standard was made to be consistent with current statistical theory.

The other recent change is the addition of an angular dependence category for badge testing. This was effective with the 2001 version of the ANSI standard. Testing of angular dependence is only done on x-ray and gamma badges.

To become accredited, the processor must supply either 15 or 21 dosimeters, depending on the test category, to the Proficiency Testing Laboratory. They are sent in

Test Category	Dose Range	Passing Level:	
		Deep	Shallow
I. Accidents, photons (x-rays and Cs-137)	5 - 500 rad	24%	no test
II. Photons/Photon Mixtures (x-rays, Cs-137, Am-241, Co-60)	0.05 - 5 rem	30%	30%
III. Beta particles	0.25 - 25 rem	no test	30%
IV. Photon/beta Mixtures, Shallow	0.30 - 30 rem		30%
Photon/beta Mixtures, Deep	0.05 - 5 rem	30%	
V. Neutron/Photon Mixtures (Cf-252 + Category II)	0.15 - 5 rem	30%	no test

Fig. 54 - NVLAP testing criteria as of 2001

separate groups, over a period of 3 to 6 months. After each group is returned, the processor must report his results to the test lab within 60 days. High energy photon performance is tested with a Cs-137 gamma irradiator in air. Low energy photons for the test are obtained from a filtered X-ray beam. The beta testing is done with a sealed ^{90}Sr-^{90}Y source that has a 100 mg/cm^2 iron filter. The filter takes out most of the low energy betas from the strontium parent but transmits the 2.27 MeV betas of the yttrium daughter. Betas from Kr-85 and from a uranium slab are also permitted under the 2009 standard. Neutrons used for testing are supposed to approximate fission neutrons. The specified source for testing is either bare or heavy water moderated ^{252}Cf, a spontaneous fission emitter. There is no test category for thermal neutrons. At the test lab, badges are irradiated while mounted on a rectangular solid acrylic plastic phantom, 30 cm by 30 cm by 15 cm thick. NVLAP accreditation must be renewed every two years.

With the changes in ANSI N13.11 one final requirement was added to the accreditation procedure. This is called the 10% rule. In addition to the average reading meeting both the bias and accuracy limits for the 15 badge set of submitted dosimeters, no more than 10% of the badges can fall outside the bias/accuracy range.

The portion of the radiation work force in the USA that is employed by the

Sample Problem 4

GIVEN:
A NVLAP accredited processor reports a deep dose equivalent of 3.2 mSv of high energy photons for a worker.
FIND:
What range is possible for the reported dose?
SOLUTION:
From Fig. 54, the possible error is ± 30% so the range is from 2.2 mSv (3.2 - 30%) to 4.2 mSv (3.2 + 30%).

Department of Energy, DOE, have their own badge testing program. The DOELAP or DOE Laboratory Accreditation Program is similar to the NVLAP program. Testing is conducted at the Radiological and Environmental Sciences Lab at the Idaho National Engineering and Environmental Lab, INEE. The DOELAP program used to conform to the DOE/EH-0027 Standard. Since release of the 2009 ANSI standard, both programs now use the ANSI Standard exclusively. It added some options specifically to meet DOE needs. Both DOELAP and NVLAP processors must pass a rigorous on-site assessment that evaluates lab personnel, equipment, facilities, quality assurance programs, dosimeters, calibration procedures, and processing procedures.

A preview as to how processors would perform under the test requirements was given by the voluntary third pilot study of 1981-82. Results obtained from the pilot testing lab are summarized as Figure 55. No distinction between badge types is made in this summary table.

The positive effect of implementing the NVLAP program is clearly evident. During the first quarter of operations under official NVLAP procedures, the badge processors attempting accreditation showed a 93% passing rate, much better than in any of the three pilot studies. One of the results of the program has been a continual drop in the number of badge processors in the United States. Back in 1993, there were 75 processors with NVLAP accreditation. By 2000, the number had fallen to 40 U.S. processors. This number remained at 40 in 2011. The number of DOELAP accredited facilities stands at 22 as of the same year.

An interesting trend is the gradual drop in film badge use. In 2010, The 40 NVLAP processors submitted 96 different badge types for approval. Only 4 of the submitted badge types were film!

In 1987, the NRC commissioners recommended that extremity monitors (ring badges) be added to NVLAP accreditation. The criteria in ANSI Standard N13.32-2008 are being used to test ring badge performance. This standard was first released in 1995 but participation has been voluntary. The NRC, to date, has not made this a national requirement in the U.S.

Radiation Category	Percentage Passing
Accident Range, Low Energy Gamma	55%
Accident Range, High Energy Gamma	82%
Low Energy Photons	55%
High Energy Photons	94%
Beta Particles	86%
Photon Mixtures	59%
Photon + Beta	84%
Neutron + Photon	72%
OVERALL AVERAGE	75%

Fig. 55 - Processor performance on 1981-82 voluntary pilot test of ANSI Standard

Criticality Badges

Criticality Accidents

A criticality accident results from an uncontrolled release of energy caused by nuclear fission. As was shown in Chapter 6, for a mass of uranium to achieve criticality, the number of neutrons in one generation must equal or exceed the number of neutrons in the previous generation, i.e., k_{eff} is > or = to 1. In the practical case, k_{eff} is the product of five factors. These are listed in Figure 56. Thus, the multiplication factor depends on the materials present in and near a uranium assembly, the size of the assembly and the actual physical arrangement of the parts.

e = Ratio of # of neutrons slowing below U-238 threshold to # of neutrons per fission

p = Probability of resonance capture escape from 1 keV to 5 eV

f = Fraction of thermal neutrons absorbed in uranium atoms

η = # of fission neutrons per neutron captured in uranium

L = Probability of neutron <u>not</u> leaking outside the assembly

Fig. 56 - The five factors influencing k_{eff}

There are several methods of preventing criticality accidents. Prohibiting the mass of individual pieces of uranium from exceeding a specified maximum size is one very effective method. The table in Figure 57 lists the smallest masses of some fissionable nuclei which can reach criticality under optimized conditions. By controlling the shape of a piece of fissionable material, the leakage can be increased thus reducing k_{eff}. The worst shape, from a criticality point of view, is a sphere because it has the lowest surface area per unit volume. If a critical spherical mass is rolled out into a sheet, the effective multiplication factor will be well below 1.0 because of leakage from the large surface area. Finally, criticality is reduced by administrative control of the immediate surrounding area with regard to materials that can reflect or moderate neutrons. These include low Z materials such as hydrogen-containing substances (water, wax, plastic, radiation protection technologists), carbon (graphite), beryllium

Radioactive	Critical Masses Measured in Kilograms		
Nuclide	Bare Solution	Bare Metal	Water Reflected
U-233	1.2	16.5	7.3
U-235	1.5	49	22.8
Pu-239	0.9	10	5.4
Am-241	??	113	105

Fig. 57 - Minimum critical masses of selected nuclides

and concrete. If these materials are close by, they will increase the flux of thermal neutrons in the fissionable assembly.

Criticality Badge Principles

The information needed from an accident dosimetry point of view is the dose due to gamma rays and the dose due to neutrons. The first is quite simple. Most criticality badges use either a film pack or TLD element to record the photon component in the conventional manner.

The neutron dosimetry is much harder. Recall that the flux to dose equivalent rate conversion factor (Chapter 5) is strongly dependent on knowledge of the neutron energy. The badge must then provide two pieces of information – the total number of neutrons passing through and the fraction of that total in each of several energy intervals. Then, the badge processor can use the conversion factors appropriate for each energy. Finally, the neutron dose is just the sum of all the dose equivalents for each of the energy intervals.

The criticality badge uses the principle of neutron activation to obtain the needed information. Various materials are selected which have the property of capturing neutrons to form a reasonably long-lived daughter activity under one of two conditions. A resonance activation detector will capture neutrons only within a narrow energy range. The cross section is minimal for other energies. A threshold activation detector will capture neutrons only if they exceed some minimum energy. A schematic plot of these two types of neutron capture probabilities is given in Figure 58.

Some materials commonly found in criticality badges are tabulated along with their characteristics in Figure 59. The first two entries have a double resonance – thermal neutrons plus slow neutrons. These two are separated out, practically, by use of cadmium filters. Recall that cadmium has a very high cross section for thermal neutron capture. By combining TWO foils in a badge holder, ONE SHIELDED BY CADMIUM, the two different neutron energies can be distinguished by subtraction of the daughter activities. By subtracting the daughter activities of adjacent threshold detectors, the neutrons in some energy interval are recorded. For example, the difference between the sulphur and copper activities will give the neutrons in the interval

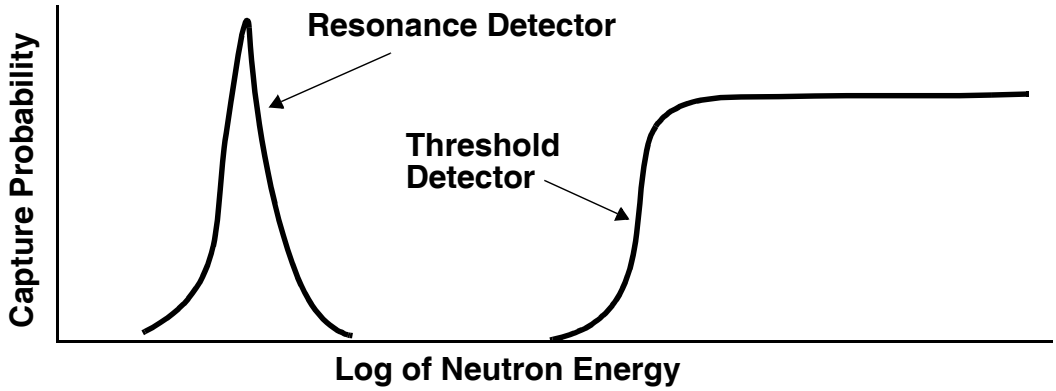


Fig. 58 - The two types of neutron capture response

Foil Detector	Resonance Energy	Threshold E.	Daughter Product
In-115	thermal, 1.5 eV	- - - - -	54 min In-116
Au-197	thermal, 5 eV	- - - - -	2.7 day Au-198
In-115	- - - - -	1 MeV	4.5 hr In-115
S-32	- - - - -	2.9 MeV	14 day P-32
Cu-63	- - - - -	11.4 MeV	9.7 min Cu-62

Fig. 59 - Characteristics of activation detectors

from 2.9 MeV to 11.4 MeV. (Cu = >11.4 while S = >2.9, so S – Cu = neutrons >2.9 but <11.4.) The actual number of neutrons in an energy interval is computed from the measured activity of the radioactive daughter. See Sample Problem 5.

Criticality Badge Holders

The holder for a criticality badge system provides spaces for the various activation detectors, cadmium covers for selected detectors and space for a gamma TLD or film detector. Figure 60 shows a commercially available criticality badge. The inside construction for this badge is shown in Figure 61. Two more examples are shown in Figure 62. These badges are from two different DOE government laboratories. The unit on the right of Figure 62 is a hybrid badge in that the gamma dosimeter space holds either a Harshaw TLD plate (as shown) or a standard film pack.

Use of a criticality badge does not guarantee that the appropriate dose information will be available in the event of a criticality accident. The mere act of recovery of the badge following an accident may necessitate heroic efforts. Criticality accidents are usually associated with large energy release, explosion and spread of large amounts of contamination. If recovery of the badge is delayed, some of the short-lived

Sample Problem 5

GIVEN:
A criticality badge is retrieved after an accident. It only shows a P-32 activity that would be produced by a neutron flux of 1.29 x 10^{14} n/cm^2-sec and a Cu-62 activity that would be produced by a flux of 2.1 x 10^{13} n/cm^2-sec.
FIND:
What information can be obtained regarding the neutron component?
SOLUTION:
From Fig. 59, all neutrons were above 2.9 MeV as the indium and gold showed no radioactivity. The flux of 2.1 x 10^{13} n/cm^2-sec is the correct value for the neutrons above 11.4 MeV, the copper threshold. The flux difference, 1.29 x 10^{14} n/cm^2-sec – 2.1 x 10^{13} n/cm^2-sec = 1.08 x 10^{14} n/cm^2-sec is the flux value for neutrons in the energy interval between 2.9 MeV and 11.4 MeV.

Courtesy of Reactor Experiments, Inc.

Fig. 60 - A commercial criticality dosimeter

activities will be lost. As shown in Figure 59, the copper daughter has a half-life of LESS THAN 10 MINUTES. In addition, the calculation of the number of neutrons passing through the badge is not a simple task. First of all, the detector must have a

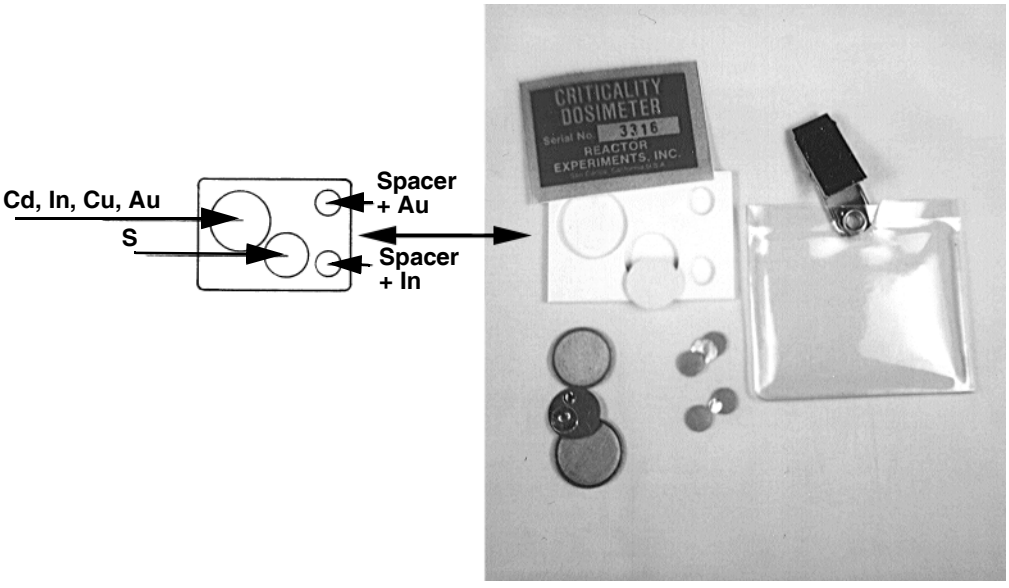


Fig. 61 - Inside construction of criticality dosimeter

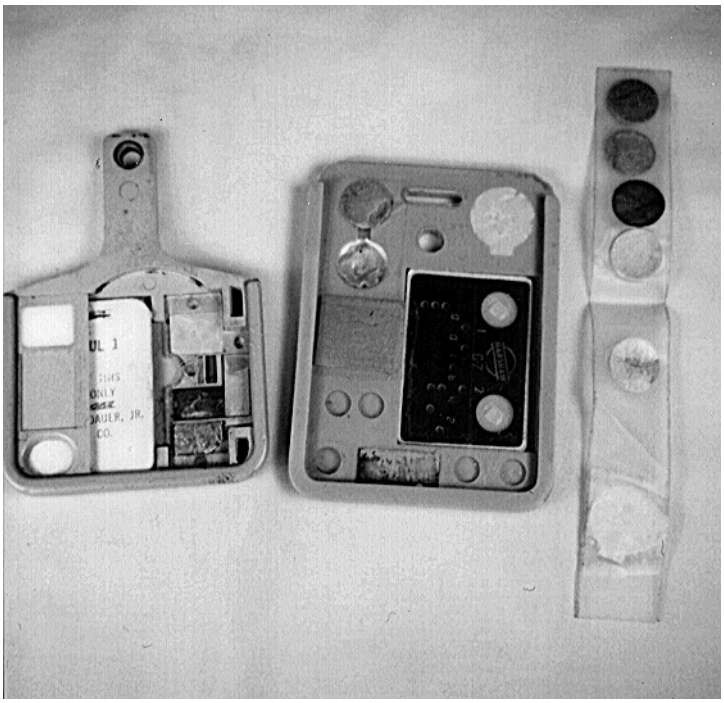


Fig. 62 - Two DOE government weapons lab criticality badges

known efficiency for the daughter activity. Then, corrections have to be applied to account for absorption of some of the radiations by the mass of the detector (self-shielding) and for the absorption of neutrons by the other detectors in the badge. Finally, the cross sections for the reactions leading to the daughter activity must be known and interfering activities must be taken into account.

In addition to the above problems, the badges described in this section are not suitable when the criticality burst is shielded by, for example, several feet of water. This would be the case in an accident in a fuel handling pool at a nuclear reactor site, or the actual case of the 1999 Japanese criticality accident where the uranium was in water solution inside a tank. The presence of large, bulky moderators will shift the neutron energies down into the intermediate range, from about 1 keV to 500 keV. Unfortunately, this is below the first threshold energy for the badge and also well above the resonances for the detectors, so the badge misses most of the neutron component.

Electronic Personnel Dosimeters

Introduction

The electronic personnel dosimeter (EPD) definitely has a role in personnel dosimetry. As one example, the radiation worker wears a small electronic unit. At the end of the work shift, the EPD is waved past an infrared reader at the access control

point. The deep and shallow dose equivalents for the day are recorded in a computer and the dosimeter is reset for it's next use. The daily doses are added to the worker's annual and lifetime totals so they are current at all times.

The evolution of the electronic dosimeter dates back a few decades. With the availability of transistors and then integrated circuits and microprocessors, it inevitably followed that smaller portable instruments with dosimetry capability would become commercially available. Small pen-sized "chirpers" were an early development. They would emit a sound when a preset dose rate was exceeded. Then, small integrating electronic units that clipped on a belt became common place. They recorded total dose and could be reset for reuse. The latest units store dose and dose rate information and can transmit this information to a computer remotely.

EPD suppliers keep working toward the goal of a complete electronic replacement for film, TLD or OSL badges. The problem continues to be reliability. In the past, a passive dosimeter (such as TLD) was clearly more rugged under extreme working conditions than an EPD. But progress is continually being made. This development is crucial if the EPD is ever to become "the dosimeter of record." This is a legal term which means the appropriate regulatory authorities have accepted the dosimeter as accurately recording a person's dose history for legal purposes. To date in the United States, only film, TLD and OSL badges have been accepted. EPDs have been able to pass the NVLAP testing criteria. They have been accepted as a dosimeter of record in the United Kingdom since 1994. They probably won't be accepted in the USA.

Michael Lantz, Senior Health Physicist at the Palo Verde Nuclear Generating Station has published a number of shortcomings of the EPD as a dosimeter of record. He points out that several trials took place in the late 1990s where EPDs were substituted for TLD programs. None of the trial sites retained the EPDs as their primary dosimeter after the testing. He attributes this to a variety of weaknesses of the various commercial EPDs:

- the silicon pin diode detectors have variable energy response at low energy
- the mechanical failure rates are up to 30% per year
- the data loss rate for EPDs exceeds that of TLDs
- one type of EPD misreads dose rate at rates below 200 mR/hr
- EPDs are larger and heavier than TLDs
- the angular response is worse than TLDs (because of the case, and battery)
- some EPDs turn off in magnetic fields near electric motors
- the EPDs can't measure beta or neutron exposures
- high dose rates may be displayed near radio frequency or microwave sources
- there is no certification program in place to judge the performance of EPDs

In spite of the numerous weaknesses noted, Mr. Lantz supports the continued use of EPDs - just not as a dosimeter of record. He notes, "The benefits of EPDs as incremental dosimeters are numerous and their advanced features add convenience and safety to all work within radiologically controlled areas." EPDs can display and alarm at preset doses and rates, the range is so wide that they replace several pocket ion chambers with a single EPD, and some models now have a "histogram feature" which means they store the dose accumulated sequentially over, for example, one minute intervals. This makes ALARA reviews following high dose rate jobs so much more meaningful. Finally, the American Nuclear Insurers take the position that personnel dosimetry always requires two dosimeters, preferably with separate, independent failure modes. This effectively dashes the hopes of having a single EPD dosimeter

of record. But the combination of a passive badge and an EPD will provide exceptional dosimetry for those licensees that elect that option.

Commercial Applications

Shown in Figure 63 is the Siemens "New EPD" electronic dosimeter. At 95 grams, including battery, the unit is heavier and larger than a passive badge. The radiological performance is illustrated by the energy response curve of Figure 64. Clearly, the EPD has a more than acceptable deep and shallow dose response in the radiation protection range. In contrast to most film and TLD badges, the device is accurate within a ± 30% tolerance range clear out to 6 MeV.

The New EPD is sensitive to x-rays, gammas and to beta rays above 250 keV. It provides a direct, digital readout of both deep and shallow dose equivalents. It is also able to display the current dose rate and has user adjustable alarm set points on both dose and on dose rate. (It can be used as a chirper and an alarming dosimeter). Finally, the data is retained for up to 10 years if the battery is accidently removed, so power failure is not a problem. Figure 65 shows the readout unit configured as an access control terminal. Plugging the EPD into the terminal downloads the data and allows the user to reset doses or reset alarm points. An optional infrared reader allows a one second issue and return time for a dosimeter, bypassing the need to physically insert the EPD into a reader socket.

Fig. 63 - A commercial electronic dosimeter

Courtesy, Siemens Environmental Systems Ltd.

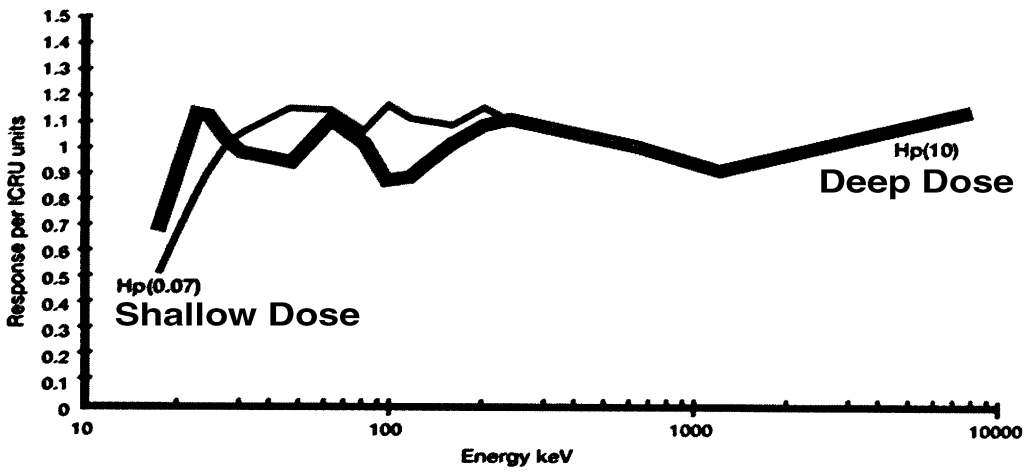


Fig. 64 - Energy response of Siemens EPD

Courtesy, Siemens Environmental Systems Ltd.

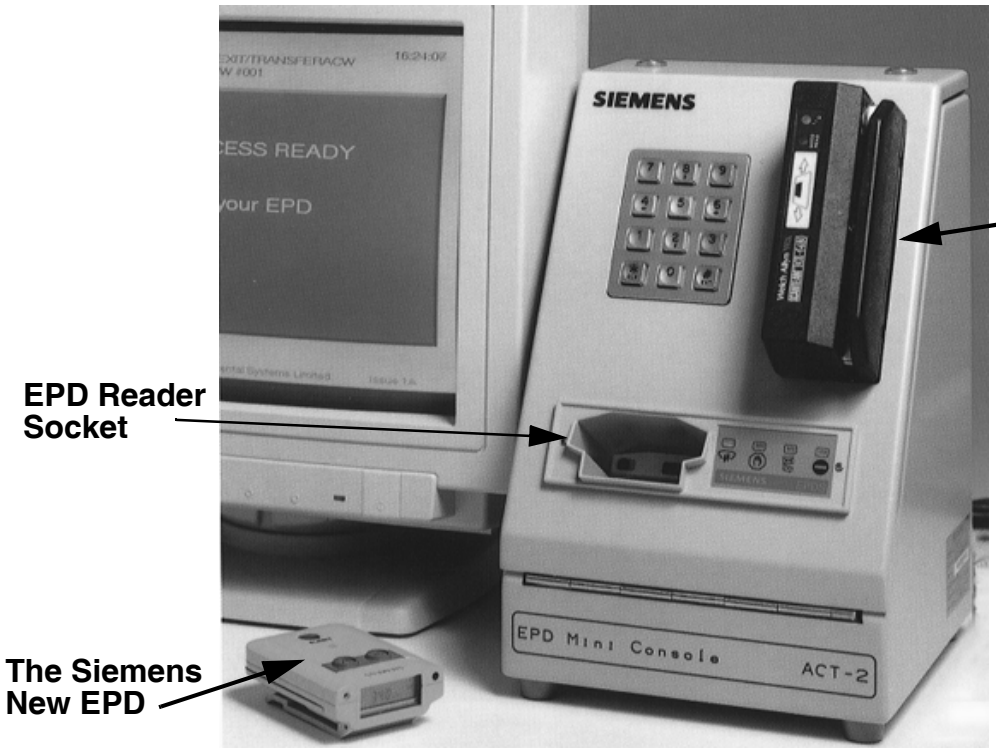


Fig. 65 - A commercial access control point terminal for EPDs

Courtesy, Siemens Environmental Systems Ltd.

Special Badge Applications

Special personnel radiation badges have been designed for a variety of unusual applications. Three such badge systems will be briefly discussed to illustrate. They include badges designed to measure radon, fast

Fig. 66 - A commercial radon personnel dosimeter

Courtesy, Terradex Corp.

neutrons and intermediate neutrons.

Radon gas poses some special problems from a radiation protection standpoint. It is widely present in low concentrations as a result of decay of ever present radium in soil and building materials. It decays by alpha emission and so can produce relatively high lung doses (the quality factor is 20). The conventional dosimeters discussed in this chapter all have the sensitive element covered by sufficient thickness of material that NO ALPHA PARTICLE CAN PENETRATE. In addition, the decay of the Rn-222 (half-life = 3.8 days) leads to four additional daughters, in series, which are radioactive with half-lives less than 10 minutes each. Thus, air containing Rn-222 will also contain approximately equal activity of Po-218, Pb-214, Bi-214 and Po-214. Badges have been developed which make use of the "etched track principle" discussed earlier in this chapter in regard to fast neutron detection. The etched holes are counted so the number of alphas passing through, and hence the dose equivalent, is determined. Figure 66 shows a commercially available radon badge with a wide enough range to cover from background environmental levels up to the radon concentrations found in uranium mine atmospheres.

Some relatively new techniques have found commercial application in neutron dosimetry. One idea, used commercially by Eberline, uses the common Harshaw TLD-100 chips as a fast neutron dosimeter. Peak number 6 (see Figure 18) of the glow curve is produced by thermal neutrons which, in the case of an occupational exposure, have been moderated by the body of the person wearing the badge. Through use of a double read cycle in which the phosphor is first heated to 250° and then to 325° the fast neutron dose information is obtained.

In the case of intermediate neutron energies such as would be found in the vicinity of a nuclear reactor or a linear accelerator operating above about 15 MeV, the albedo badge is often used. The term albedo refers to neutrons which have been reflected or scattered. Intermediate energy neutrons strike the human body that is wearing the badge. They are moderated in energy and are scattered in different directions. A few become moderated to thermal energy and are reflected from the body through the back of the badge into combinations of TLD-600 and TLD-700 dosimeters which measure the dose. By shielding the badge on the front side with cadmium, external thermal neutrons in the workplace are rejected. Two major problems remain with this system. The badge must be kept in close contact with the body to retain its sensitivity. Secondly, the badge response per mrem changes by a factor of about 2,000 over a neutron energy range from 1 keV to 10 MeV. This means that the average neutron energy must be known in advance in order that the badge processor can assign the correct calibration factor to the readings. Figure 67 shows a commercial albedo badge for neutron dosimetry.

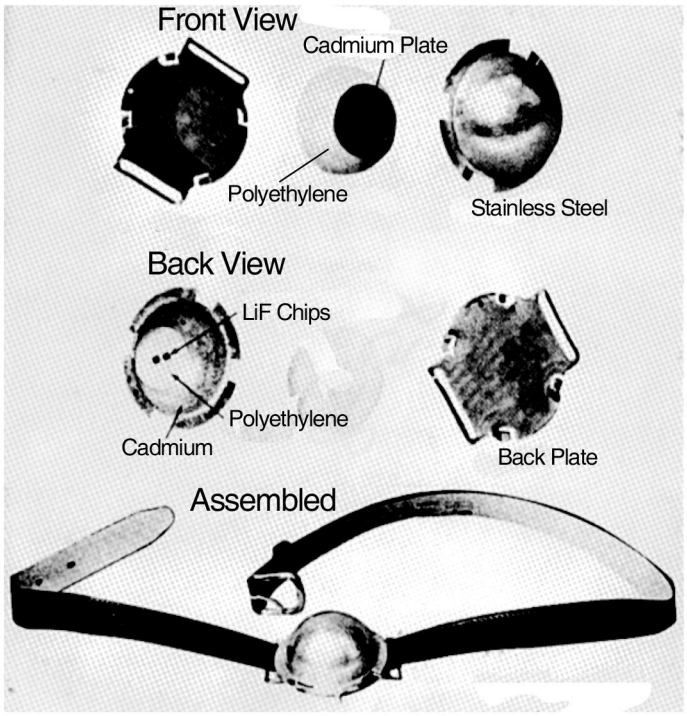


Fig. 67 - A commercial albedo type neutron badge

U.S. Regulatory Requirements

With the implementation of the "new" 10 CFR 20 regulations in the USA in 1994, the rules for personnel monitoring changed drastically. First, requirements for

internal dosimetry, (covered in the next chapter), were added. Then the external monitoring rules were updated. Four different types of external dose limits must be met. First, the deep dose equivalent to the whole body must be determined at a tissue depth of 1 cm, i.e., under an absorber equal to 1,000 mg/cm^2 of density thickness. Secondly, the dose equivalent to the lens of the eye must be measured under a density thickness of 300 mg/cm^2. Note that no weighting factor is used in either of these determinations. (It might be helpful to review the early sections of Chapter 5.) Next, the shallow dose equivalent to the skin must be monitored. Finally, the shallow dose equivalent to the extremities (officially defined as "hand, elbow, arm below the elbow, foot, knee and leg below the knee."). Both of these shallow dose equivalents are measured under 7 mg/cm^2 and are averaged over an area of 1 cm^2.

Attention in the new law is also given to badge placement. The badge must measure the maximum dose to the part of the body being monitored. For relatively uniform fields, the NRC recommends placement on the front of the trunk between the head and the waist. For non-uniform fields, place the badge near the part of the body receiving the highest dose. For example, the badge is placed on the head if dose rates are higher at head level than for any other part of the body. If it is not possible to predict ahead of time the body section that will receive the largest dose, the NRC recommends using multiple badges and recording only the badge with the highest dose.

Finally, there is a requirement [10 CFR 20.1201(a)(1)(ii)] for measuring the external and internal dose sum to individual body organs under certain conditions. This will be covered in more detail in the next chapter. However, for this calculation, the external component of the sum is merely the deep dose equivalent as measured by the normal personnel badge.

NRC Forms 4 and 5 are used for reporting the multitude of different doses now required for certain workers. These will be treated near the end of Chapter 9 after internal dose assessment has been covered.

Problem Set

1. What is meant by the term "flat energy response" as applied to a personnel dosimeter?

2. Do you agree with the text position that the necessary range for a personnel badge is from 10 mrem to 1,000 rem? Why or why not? Do commercially available badge systems cover this range for all desired radiations?

3. Describe the formation of a latent image center in photographic emulsions.

4. Describe the steps taken to "process" photographic film. Which of these steps is the most critical in terms of quality assurance in the dose information?

5. Calculate the optical density of the combination of two films, one with an O.D. = 1 and the other with O.D. = 2. What fraction of the incident light photons would be transmitted by the combination?

6. In the film characteristic curve of Figure 8, why doesn't the curve start out at "0" on the Optical Density axis?

7. In a film pack containing two pieces of film, which contains more silver bromide, the "low sensitivity" or the "high sensitivity" film?

8. Discuss the reasons why a film exposed to 1 R of Co-60 would be much less optically dense than one exposed to 1 R of 50 keV photons. What is your prediction about the O.D. of the same type of film exposed to 1 R of N-16 gamma rays?

9. What does the term "filter ratios" mean with respect to a film badge? How are they used to measure the energy of the exposing gamma rays?

10. Describe a practical film badge design feature which would indicate whether the badge was exposed from the front or the rear. Would this technique work with a TLD badge?

11. Describe how both fast and thermal neutrons can be detected by a film badge. What is meant by "fading" in the case of neutron track film?

12. Discuss the mechanism by which electron traps become "populated" in a TLD phosphor. Describe how the TL light is produced during heating.

13. Of what practical use is a glow curve for a phosphor? In what sense are the phosphor fading characteristics dependent on the glow curve?

14. Sketch the basic components of a TLD reader and describe their function. Describe some of the improvements made in commercial TLD readers to increase sensitivity and reliability.

15. Reconcile the statement that "TLD-100 has a linear dose response over the radiation protection range" with the non-linear curve in Figure 23.

16. What is the chief reason for the better energy response of common TLD phosphors than that of film?

17. How is the reusability of a TLD chip both an advantage and a disadvantage?

18. Describe how LiF phosphors can be used to record thermal neutron dose equivalents. Why won't this same technique work for fast neutrons?

19. Name some advantages and disadvantages of other TLD phosphors compared to lithium fluoride.

20. Name two differences in the physical mechanisms taking place during the readout cycle between TLD and OSL phosphors.

21. What is meant by the terms "shallow dose equivalent" and "deep dose

equivalent?" How is a personnel badge designed to read each of them?

22. Briefly discuss some of the practical handling problems with TLD dosimeters in the extruded chip form. How are these problems solved by a TLD processor?

23. What are some conditions under which a criticality accident would likely occur? What changes could be made in these conditions to reduce the chance of such an accident?

24. How does a criticality dosimeter allow the neutron energy spectrum to be estimated? What are some practical problems in obtaining this information?

25. Name some of the usual metal elements likely to be found inside a gamma-neutron film badge holder.

26. Briefly discuss the principle of operation of an albedo dosimeter.

27. What are the four types of external measurements required under the current 10 CFR 20 regulations for occupational workers?

S-1. $CaSO_4$ overresponds to low energy photons by a large factor. Why, then, is it used in the Panasonic TLD badge?

S-2. How does the heater in the Panasonic system differ significantly from most conventional TLD reader heaters?

S-3. What is the operating principle of an etched track dosimeter? For what two radiations is this principle useful?

S-4. What is the chief drawback to use of electronic dosimeters in the USA?

Other Resources

1. "Use of Personal Monitors to Estimate Effective Dose Equivalent and Effective Dose to Workers for External Exposure to Low-LET Radiation," NCRP Report Number 122, Washington, 1995.

2. "American National Standard for Dosimetry - Personnel Dosimetry Performance - Criteria for Testing," ANSI N13.11, Amer. National Standards Inst., New York, 2009.

3. "American National Standard - Performance Testing of Extremity Dosimeters," ANSI N13.32, Amer. National Standards Inst., New York, 2008.

9

Internal Dosimetry Techniques

Outline of this Chapter

Chapter Summary

This chapter covers the other half of the personnel dosimetry problem – internal dose assessment. We begin by covering two methods of finding the radioisotopes and activities deposited in a worker at some given time. Next, calculational methods are presented to estimate the activity of the intake. Finally, methods are discussed for estimating the internal committed dose equivalent following the intake. At the end of the chapter, methods for adding together external and internal dose are covered as a way of concluding both Chapters 8 and 9.

Bioassay techniques are one way to determine the content of radioactivity in the total body (body burden) or in a body organ (organ burden). The technique involves sampling a body elimination product such as urine and calculating the present burden from the concentration of radioactivity in the sample. Bioassays are limited to radioactive contaminants with sufficient solubility in body fluids to show up with a measurable concentration in the sample.

In vivo counting treats the body, or a portion of it, as a radioactive sample. By counting the emitted radiations, an estimate of body or organ burden at the time of measurement can be made. This technique clearly is limited to radioactive contaminants which emit radiations that can penetrate overlying body tissue and be detected externally.

Once the body or organ burden is determined, the original intake activity must be calculated. This is often done by measuring the burden over an extended time period and then calculating the initial intake using published intake retention fractions. The intake activity is used to calculate the internal dose to the body or an organ.

Finally, the dose is calculated. Usually, methods developed by the International Commission on Radiological Protection are used. The 50 year committed effective dose equivalent, CEDE, is found by comparing the intake activity with the annual limit on intake, the ALI. One ALI results in a CEDE of 50 mSv (5 rem) in the United States or 20 mSv (2 rem) in the rest of the world. The special case of an embryo/fetus exposed through the occupational dose to the mother is also examined.

This chapter concludes with a discussion on the proper way to add internal and external doses (Chapters 8 and 9) to satisfy regulations. NRC Forms 4 and 5 are covered as one example of proper dosimetry record-keeping.

Introduction

As a result of dosimeter badge advances over the years and because of the inherent simplicity of the problem, it has been common practice for years to assign the reading of an external personnel dosimeter as the "dose of record" for a worker. Of course, this was in the old days when the external dose was "the dose." Now, in a more enlightened millennium, "the dose" includes both an external component from field exposures AND an internal component from radionuclides deposited in the body.

There is no direct reading internal dosimeter badge available (nor is there ever likely to be one). Instead, the internal component of a worker's legal dose history is arrived at through a series of steps - see Figure 1.

1) Determine the body (or organ) burden
[from bioassay or whole body (or partial body) count]
2) Compute the initial intake at time = 0 or intake history
3) Choose a dose model
4) Calculate the internal committed effective dose equivalent

Fig. 1 - The steps in calculating an internal dose

The usual starting point is to characterize the deposited radioactivity in terms of radionuclide identification, activity and body location, i.e., determine the "burden." These tasks fall to the bioassay or whole body counter laboratory. Both will be discussed below. Next, for the case of an acute, accidental intake, an estimate must be made of the total activity taken in for each radionuclide. If the intakes are more or less continuous over time (the usual occupational scenario) then the intake during the time period being evaluated is calculated. Choice of a dose model is perhaps the most difficult step. For many radionuclides, their behavior in the body is not known very well. This is particularly true for individuals that don't fit the "Reference Person" characteristics very well, e.g., infants, or unusual body size adults. Finally, all the parameters are fed into the dose model and a result obtained. The result is the committed effective dose equivalent or CEDE discussed in Chapter 5.

Bioassay Techniques

Basic Principles

As used in radiation protection, the term bioassay refers to some analysis procedure for determining the nature and activity of the internal contamination present in a person by making measurements on a body excretion product. It is assumed that the concentration of radioactivity in the body elimination product is proportional to the activity deposited in the body. The activity concentration in the sample is measured by conventional techniques (e.g., calibrated gas flow proportional counter). Next, a guess is made as to the value of the proportionality constant based on the

Exhaled Air	**Sweat**
Nail Clippings	**Saliva**
Nasal Mucous	**Hair**
Urine	**Feces**

Fig. 2 - Some body products useful for bioassay

previously measured behavior of these radioisotopes under similar conditions. The answer obtained from this calculation is called the body burden, the total activity content of a particular radioisotope in a person. If the specific distribution is known, the organ burden might be specified (e.g., the thyroid burden or lung burden is 2.3 MBq). Note that bioassay measurements give the burden at the time of measurement, not intake, and so the actual intake of radioactivity that resulted in the measured burden must be estimated by calculations which are the subject of a later section in this chapter.

There are a number of body elimination products that have been used from time to time for bioassay procedures. Figure 2 is a listing of the more common samples. Routine, large-scale bioassay programs almost exclusively use urine as the sample. The procedure is termed "urinalysis." This is due to the ease of collection and for aesthetic reasons. Nasal swab samples and exhaled air samples (obtained by inflating a balloon) also are common in certain facets of the nuclear industry where there is a reasonable potential for actual inhalation of radioactivity. Sampling by nasal swab is common during a radiation accident or decontamination activities.

Radioactive contaminants that can enter the body are often classified into the somewhat loosely defined categories of "soluble" and "insoluble" where the solubility is with respect to body fluids. In addition to solubility, the route of intake to the body must be specified, in many cases, before a complete analysis can be made. The various entry routes to the body are listed in Figure 3.

The first two listed are by far the most common. Percutaneous uptake refers to absorption of radionuclides directly through the skin membrane. This is a common route for tritium exposure (H-3) as the molecule is so small.

The insoluble contaminants are usually the more difficult problem in measuring body burdens. In the case of ingestion, since the radioisotope is insoluble, it passes relatively unscathed directly through the gastrointestinal tract. If the nuclide does not emit radiation which can be detected external to the body, then a fecal analysis is performed to measure the body burden. In the case of inhalation of insoluble radionuclides, the body clearance rates will depend on the pulmonary rates (volume and

Inhalation - Through the nose
Ingestion - Through the mouth
Percutaneous - Absorption directly through the skin
Wounds - Injection through the skin

Fig. 3 - Routes of entry for radioactive materials

frequency of breathing), and the size of the particles (which determines where they become trapped in the airways of the respiratory tree, Figure 4). Since the particles don't dissolve, they ultimately are carried up the respiratory tree by action of the hair-like cilia which line the airways. Eventually they reach the epiglottis where they enter the gastrointestinal tract and pass on through the body.

The category of contaminants which are soluble in body fluids is usually further subdivided into three cases: contaminants which dissolve uniformly into body water, those which seek out a particular organ for deposition (like iodine in the thyroid gland) and those which enter the bone of the skeletal system. (See Figure 5).

The first case, soluble in body water, is the easiest to handle. It is usually assumed that these contaminants are uniformly deposited throughout all body fluids.

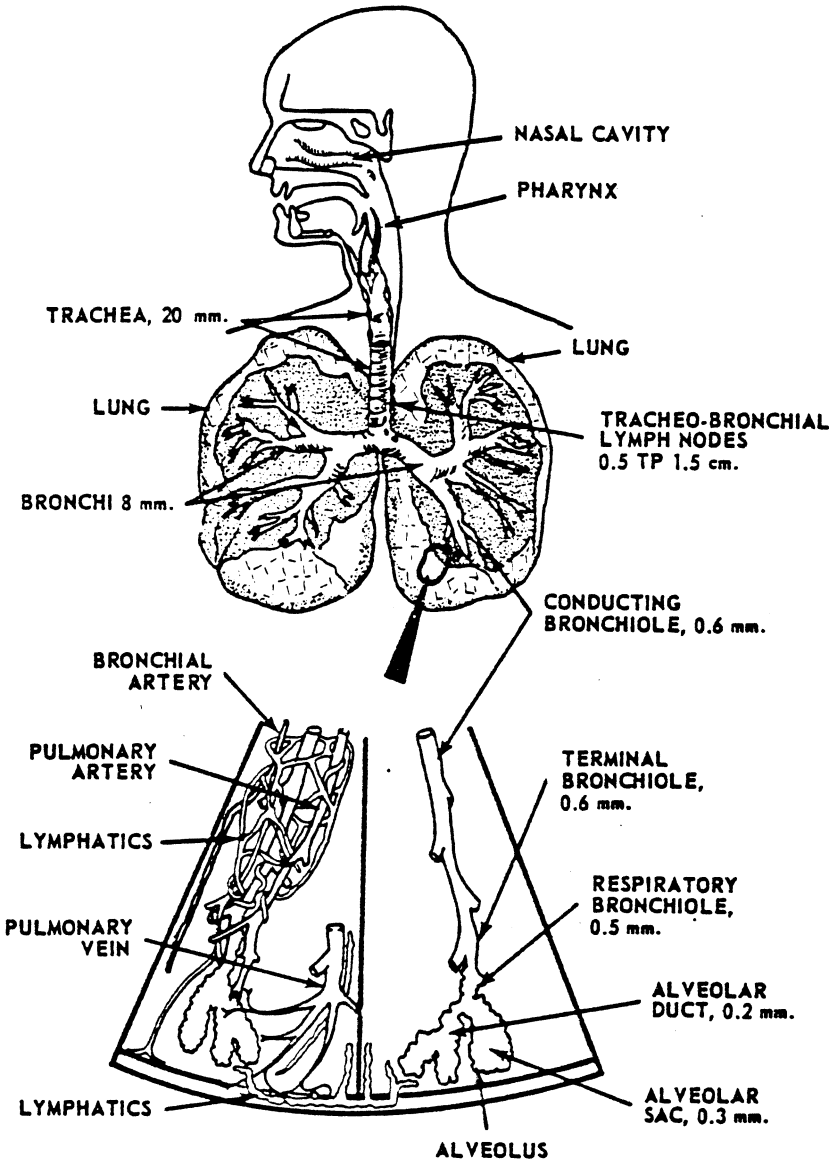


Fig. 4 - The human respiratory tree

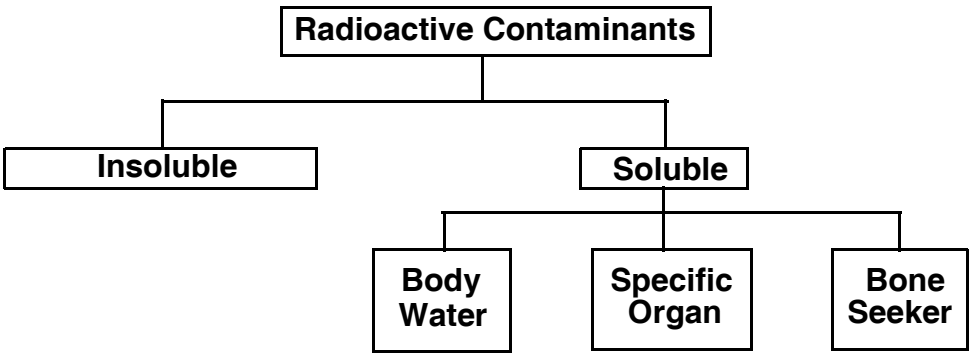


Fig. 5 - Classification of internally deposited contaminants

Thus, since urine is one of the body fluids, the concentration of a soluble radionuclide in urine is considered to be identical to the concentration in the rest of the body water. For purposes of radiation protection calculations in the U.S., anatomical values for internal dosimetry are usually taken from Publication Number 23 of the International Commission on Radiological Protection, ICRP, (**Report of the Task Group on Reference Man**). [For technicians working in the rest of the world, newer updates of the ICRP publications are used. These are discussed later in this Chapter.] ICRP 23 is an exhaustive compilation of masses, sizes and compositions of human organs and bodies from prenatal to adult. The "Reference Adult Male" and "Reference Adult Female" are considered to possess the "average" characteristics of all North American men and women. The total body water of "Reference Adult Male" is 42 kg while "Reference Adult Female" has 29 kg. Thus, in the case of a radioactive contaminant uniformly deposited in body water, the total body burden is found by taking the urine concentration and multiplying it by the total body water. An example calculation is done in Sample Problem 1. The conversion factor follows, of course, from the fact that water has a density of exactly 1 kg/liter. If the worker had been male, the body burden would have been 420 microcuries due to the higher mass of "Reference Man."

The clearance or removal of a radionuclide uniformly deposited in body water following a single uptake is a result of filtration in the kidney. This causes the concentration of the soluble radionuclide to decrease exponentially with time. A graph of

Sample Problem 1

GIVEN:
A female worker submits a urine sample with 0.01 μCi/ml ^{3}H.
FIND:
Calculate her body burden, in Bq at the time of sampling.
SOLUTION:
For a female worker, the body burden = activity in body water = urine concentration x body water mass = 0.01 μCi/ml x 29 kg x 1000 ml/kg = 290 μCi x 3.7 x 10^{4} Bq/μCi = 1 x 10^{7} Bq

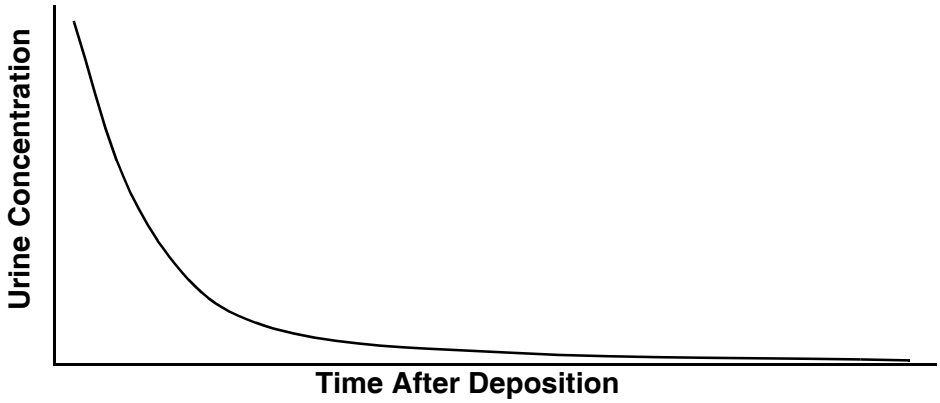


Fig. 6 - Clearance of a contaminant via body water

such behavior is given in Figure 6. Recall from Chapter 1 that raising e to some power to make an exponential function is the inverse operation from taking the logarithm. It should not be too surprising that when the above graph of Figure 6 is re-plotted on semi-log paper (which takes the logarithm of the quantity on one axis), the result is now a straight line. This identical behavior would be seen for a semi-log graph of the physical radioactive decay of a sample versus time. Figure 7 shows the result. A graph such as this last one is commonly constructed following an accidental uptake of a significant level of radionuclide. A series of urine samples are collected and measured. The graph is constructed by fitting the best straight line to the data plotted on the semi-log graph. The clearance half-life, the time for half of the radionuclide to be removed, is then easily calculated from the graph. It is just the length of time for the concentration to fall to one-half from some arbitrary starting point. This number is extremely important in the dose calculations made after accidental uptakes. To cite one example, the clearance half-life for tritium in humans is about 10 days.

The organ-deposited contaminants are the second of the three cases for soluble radionuclides. Many different chemical elements or compounds are concentrated into

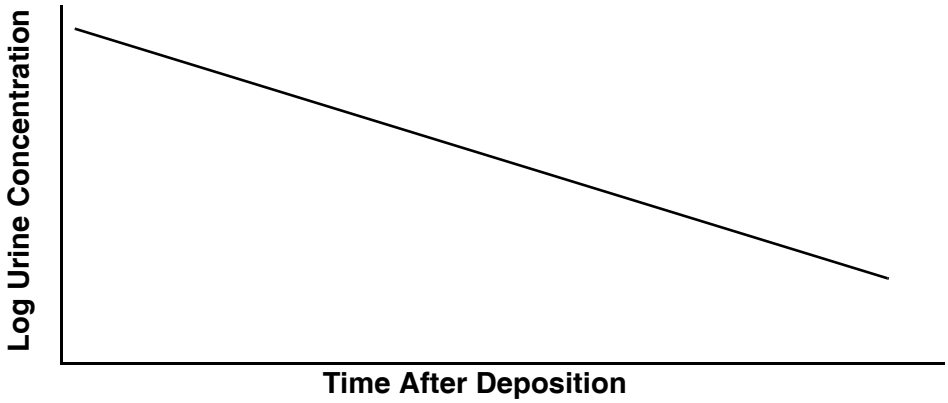


Fig. 7 - Data of Fig. 6 plotted on semi-log scale

certain body organs by normal metabolic activity. Iodine is probably the best known example. This element is needed by the thyroid gland to produce the normal thyroid hormones. As blood passes through the gland, iodine is extracted and concentrated. In a normal adult, about 25% to 30% of the iodine circulating in the blood following an acute intake will be removed and stored in the thyroid gland tissue. Uptake is completed within two days following ingestion or intravenous injection. Biological clearance from the thyroid gland itself occurs with a clearance half-life of about 70 days. The ICRP recommends a conservative value of 120 days for radiation protection calculations. Once the iodine leaves the thyroid, it is again in body water where it is subject to removal in the kidney and excretion in the urine. The ICRP assumes a 12 day clearance half-life for iodine in body water. Thus, the complete "life history" of iodine involves a two-step process which is typical of all organ-deposited radionuclides.

Generally, uptake from the blood to a specific organ takes place fairly rapidly. The nuclide then leaks out of the organ back into body water where it is quickly cleared. The clearance curve for urine concentration versus time, plotted on semi-log graph paper for the organ-deposited case of a soluble contaminant is shown in Figure 8. Initially, the rapid clearance is due to removal of the fraction of the radionuclide that doesn't become organ-deposited but remains in body water. After this fraction is cleared, then the remaining radionuclide that appears in the urine has slowly leaked out from the organ into the body water. This concentration in the urine is dependent totally on the clearance rate from the organ. Thus, the semi-log plot shows a curve rather than a straight line. In fact, the curve is really the sum of two straight lines - one for the fraction remaining in body water after the initial uptake and the other corresponding to the slow organ clearance. This behavior is clarified by Figure 9 which resolves the curve of Figure 8 into the two components. By graphically resolving the two straight lines in the case of an actual uptake by a worker, both clearance half-lives can be readily computed for use in the final dose calculation.

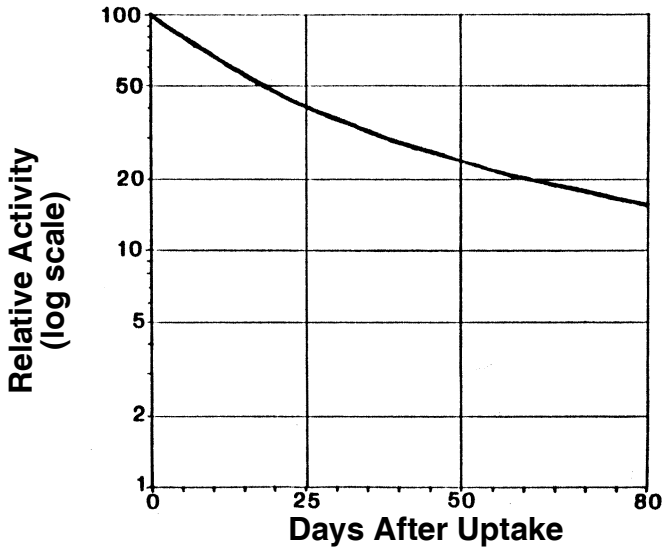


Fig. 8 - Semi-log clearance curve for an organ-deposited nuclide

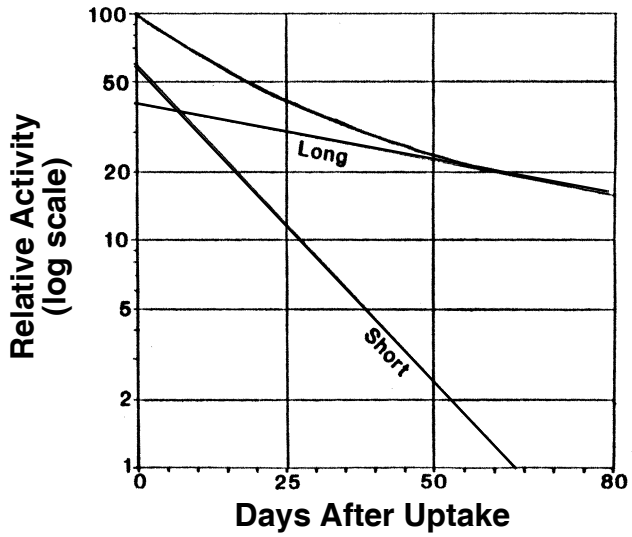


Fig. 9 - Clearance curve of Fig. 8 showing curve components

The third and final category of soluble contaminants has been given the name of bone seekers. Historically, the field of internal dosimetry began as a result of the unfortunate radium dial painters in the 1920s. The industry generally employed young women to apply radium luminous paint to watch, clock and instrument dials. It flourished in several states between about 1915 and 1925. The girls had a continuous daily uptake as a result of licking the paintbrush tip to produce the fine point needed for the small lettering. Radium was found to readily attach to bone in the exposed person where the high stopping power alpha emissions could pose a serious problem in terms of tissue destruction. The initial discovery of the relation between the dial painting occupation and health effects was due to an alert dentist, Dr. Theodore Blum. The first death of a dial painter as a result of radium uptake occurred in 1925. By the late 1920s, the dial painting industry prohibited the practice of "tipping" the brushes so further exposures ceased. However, by then, several hundred young women had been exposed to significant levels. Dial painter deaths were usually due to radiation-induced anemia, sinus tumors or bone cancer.

As a family, bone seekers have extremely long retention times after incorporation into bone tissue. Plutonium isotopes have a biological clearance half-life of about 200 years for bone tissue. To be strictly correct, bone seekers would be considered "organ-deposited." However, the fact of the extremely long clearance times coupled with the vast historical experience for these particular radioisotopes causes them to be placed in a separate category. For purposes of radiation protection calculations, radium and calcium isotopes are assumed to have biological clearance half-lives of 45 and 49 years, respectively, for bone.

Practical Bioassay Applications

In the case of a suspected radiation accident involving release of radioactive material, nasal swabs are commonly used. As a field technique, this can give early

indication of possible inhalation of contamination by the persons involved in the incident. A cotton swab is usually moistened with alcohol and then lightly rubbed over the inside surface of one nostril. The procedure is then repeated, with a new swab, for the second nostril. (Or the worker can blow their nose into a tissue which can be dried). After collecting the samples, they should be double-bagged and labeled. By placing a second bag over the first, the possibility of cross-contamination is reduced. This problem refers to the transfer of activity to a clean sample from a hot sample by leaking contamination (or contamination on the outside of the hot sample) when a whole set of containers are packaged together for transport to the counting lab.

Proper labeling is vitally important. Considering the confusion which surrounds any radiation accident, it will be impossible, AFTER THE FACT, to determine which sample was taken from which victim and when. The basic information that should be put on a label for a nasal swab sample is listed in Figure 10. In addition to the fact that the sample is legal evidence of internal contamination (or lack thereof) of an individual, the sample information will be used by the treating physician.

Sample Date	**Time Collected**
Victim's Name	**Victim's Identification #**
Sampling Person's Name	**Unusual Circumstances**

Fig. 10 - Information required on a bioassay sample

The other important rule for nose sampling is always collect a nasal swab before the decontamination shower is taken if this does not pose additional risk of significant skin dose to the contaminated person. During the shower, the person is instructed to thoroughly wash the nasal passage, thus, removing any deposited radioactive contamination. This destroys valuable information which could have been used as a basis for starting treatment of the victim, possibly dramatically reducing the internal dose that would be received.

Under accident conditions, it is often not possible to have an immediate accurate analysis made of bioassay samples. This might be caused by collecting logistics for an isolated accident site, lack of sensitive, calibrated counting equipment on-site or just the sheer volume of samples taken. Thus, it is helpful to have some "rules of thumb" to aid in "on the spot" decision making for processing exposed personnel. Potential inhalation of alpha emitting radionuclides is the most serious problem. It is generally felt that a reasonable action level is 200 disintegrations per minute of alpha contamination from the swabs of both nostrils counted together on a portable alpha survey meter. The Radiation Emergency Assistance Center/Training Site, REAC/TS, uses a rule of thumb that 10% of the intake is on the nasal swabs taken within one hour of exposure. Take particular care so as not to puncture the fragile window of the alpha detector. Swabs that have a higher reading than the action level indicate that the person very likely had a serious inhalation exposure. These persons should be immediately referred to a physician for analysis and should be put in a urine sampling program.

Urinalysis is used both under accident conditions and, at many facilities, as a routine monitoring technique to assure good work practices. Any internally deposited

contaminant which is even slightly soluble in body fluids will eventually show up in a person's urine. Under accident conditions, several special rules apply. If the accident is less than 1 hour old at the time of sample collection, the victim should be instructed to first void the bladder and then collect the first sample that is able to be produced. Physiologically, it takes some time for the contaminant to dissolve, be absorbed into the bloodstream, be filtered out in the kidney and appear in the bladder. As a rule of thumb, it takes about 4 hours before the maximum urine concentration is reached following an acute uptake. If the person does not first void the bladder, the urine present from before the accidental intake will dilute the contamination and so the sample will show a deceptively low concentration. If the initial sorting of victims is done based only on this first urine sample, many persons could be improperly dismissed without further treatment.

Urine samples are normally collected in waxed containers or polyethylene bottles. Once again, pay proper attention to labeling. The same information noted in Figure 10 is needed. Particular care should be taken to get the correct time of sample collection as urine concentration changes quite rapidly in these first few hours. If the accident is more than an hour old at the time of sample collection, the victim should collect the sample without first voiding the bladder. Another important rule is that urine samples should be collected after the decontamination shower. This reduces the chance of contaminating the collected sample from external body radioactivity which gets into it during collection. Such external contamination could easily lead to a false diagnosis of an extremely large body burden and could cause many problems for both the radiation protection technologist and the victim.

One final rule applies to urine collection if tritium is a possible contaminant. As mentioned previously, the molecule diffuses rapidly through many materials due to its small size. This can pose a major problem if a bunch of urine samples are stacked together for any length of time before analysis. The tritium from a single sample with a high concentration can diffuse out of the sample container and enter the adjacent samples, contaminating them to a significant level. This peculiar problem can be eliminated by use of GLASS BOTTLES for sample collection if tritium is suspected.

Routine urinalysis programs are common at many facilities. Industrial operations involving uranium in the nuclear fuel cycle or for the manufacture of armor piercing munitions can cause relatively high body burdens. Tritium is likely to appear in urine samples from workers involved in the labeling ("tagging") of organic chemicals or workers in nuclear power plants of the heavy water variety. (U.S. plants are exclusively light water moderated. Canadian and West German plants use heavy water, D_2O, as a moderator). In the DOE sector, tritium is a problem at sites handling the large quantities used in most nuclear weapons. Although new production is curtailed, the 12 year half-life means that tritium is being continually replaced in old weapons as they are refurbished.

Finally, in both nasal swab and urine sample collection, it is important to take some random samples from nonvictims. This provides a comparison set of samples for analysis and points out problems in counting equipment and/or procedures.

In Vivo Counting Techniques

Basic Principles

The second general method for determining the body burden of a deposited radionuclide is *in vivo* counting. This involves the placement of an external radiation detector near the body to measure radiations emitted from internally deposited radionuclides. (You may be more familiar with the term "whole body counting" to describe this procedure. Since frequently it is only part of the body that is counted, e.g., a thyroid count or lung count, the newer term *in vivo* counting is the preferred name.) In contrast to the bioassay method discussed above, *in vivo* counting clearly only works for those radionuclides which emit radiation that is able to penetrate body tissues and be externally detected. Practically, this means that the method is limited to gamma emitting radioisotopes (or in a few cases of very high Z elements, x-rays of reasonably high energy emitted following internal conversion). In general, a large sensitive scintillation counter or semiconductor counter is used with massive shielding to reduce background interference. As was the case with the bioassay methods just discussed, *in vivo* counting gives the burden at the time of measurement and so the actual earlier intake of radioactivity must be estimated by calculations which are discussed later.

Historically, the radium dial painters were the first occupational group needing the services of an *in vivo* counter. Due to the extremely long retention time of the bone-seeking radium, very little is excreted in the urine so bioassay techniques are unsuitable. An attempt was made to measure radium body burdens of dial painters in 1930 by use of a set of long Geiger counter tubes surrounding the person. Unfortunately, the inherently low sensitivity of the GM counter caused these efforts to fail.

In the 1950s, the development of solid and liquid scintillation counters opened the door to practical *in vivo* counters. The design differences between a solid crystal scintillation counter and a liquid scintillation counter (which uses a solution of scintillating chemicals in a light-tight tank surrounded by photomultiplier tubes) lead to two different general types of whole body counters. These are discussed separately below.

Liquid Scintillation Whole Body Counters

This type was the first whole body counter developed for practical use in radiation protection. The prototype model was built by the USA in 1955 for the first International Conference on the Peaceful Uses of Atomic Energy held in Geneva, Switzerland. It used a vertical tank in the form of a spiral "sheet" to hold the scintillation solution. The opening in the spiral was the "door." The tank was shielded by 10 tons of lead 3 inches thick. It had a gamma ray efficiency of about 25%. (See Figure 11 for an artist's sketch).

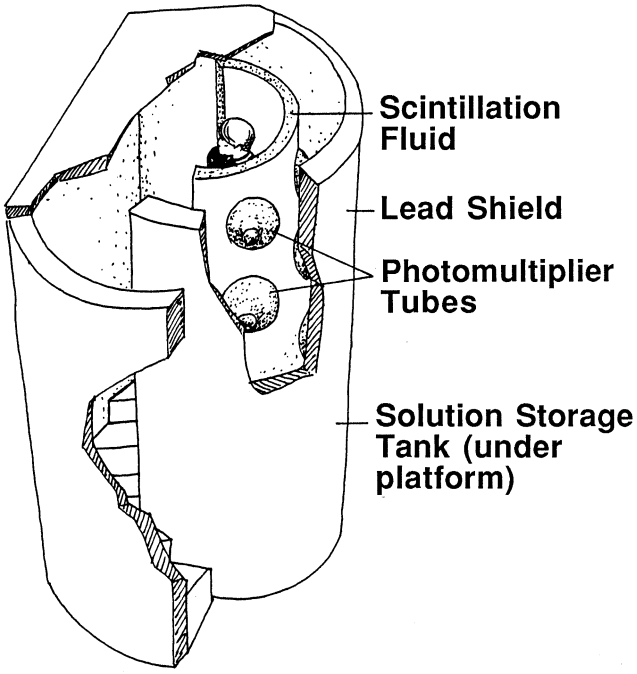


Fig. 11 - Sketch of world's first whole body counter

A modern liquid scintillation whole body counter typically contains at least several hundred gallons of scintillation fluid, several hundred photomultiplier tubes and several tons of surrounding shielding. A common configuration is a horizontal cylinder with a hollow cylindrical opening along the axis as shown in Figure 12. Typically, the gamma ray efficiency of such counters is between 15% and 30%. The energy resolution (defined in Chapter 7) is poor for this variety of counter. It may be about 1 MeV in a practical counter. This means that the liquid scintillation whole body counter is NOT USUALLY ABLE TO IDENTIFY PARTICULAR RADIONUCLIDES in the radioactive body burden. Gamma rays from two different radioisotopes would have to differ in energy by more than 1 MeV to show up separately in the analysis. On the other hand,

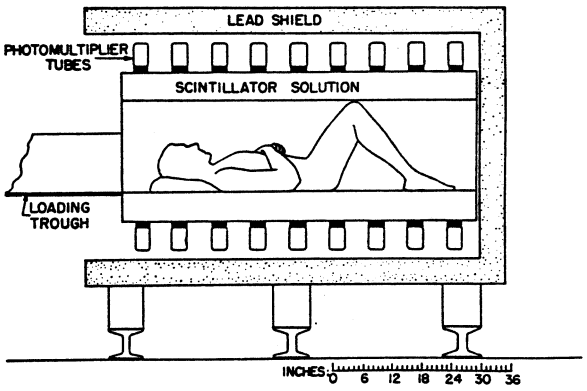


Fig. 12 - Typical configuration of a liquid scintillation whole body counter

the high efficiency of this detector type allows high sensitivity in detecting deposited internal radioactivity, and it allows relatively short counting times.

Solid Crystal Scintillation Whole Body Counters

The most common crystals used are the conventional NaI(Tl) assemblies covered in Chapter 7. To obtain the sensitivity necessary to measure the very low radiation levels associated with most internally deposited radioactivity, the crystals must be physically large. (For the sake of the curious, the "world's largest" factory grown single crystal of NaI(Tl) as of 2000 was a 21" thick by 20" diameter giant produced by Bicron). The large crystals used for solid crystal whole body counters employ several photomultiplier tubes in the complete detector assembly. Of course, even the largest crystals are still much smaller in volume than the common liquid scintillation whole body counters. A very common configuration in the stationary whole body counter is to seat the person being counted in a special chair which allows them to be "wrapped around" the crystal which is held in the lap. Geometrically, this is the exact inverse configuration of the liquid counter in which the counter is wrapped around the worker. An alternate configuration places the person supine on a flat table and the

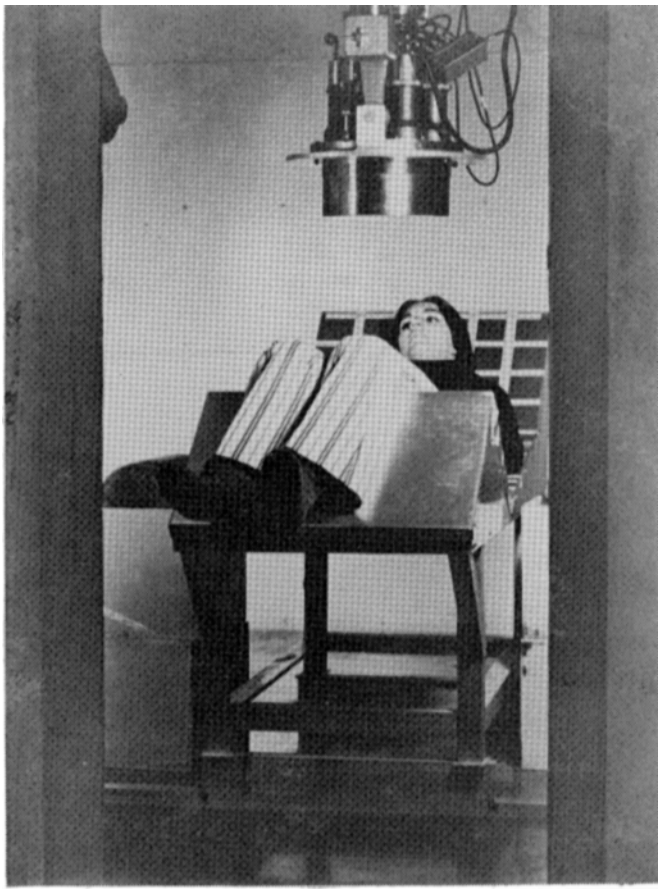


Fig. 13 - A solid crystal whole body counter and shielded room

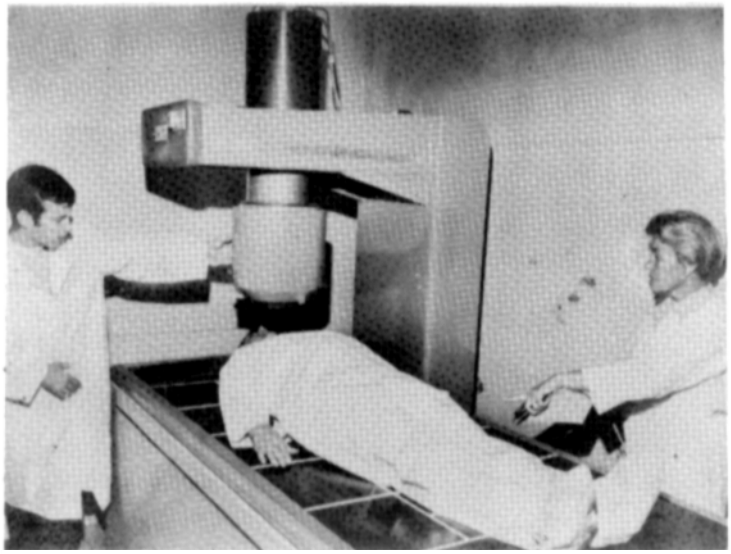
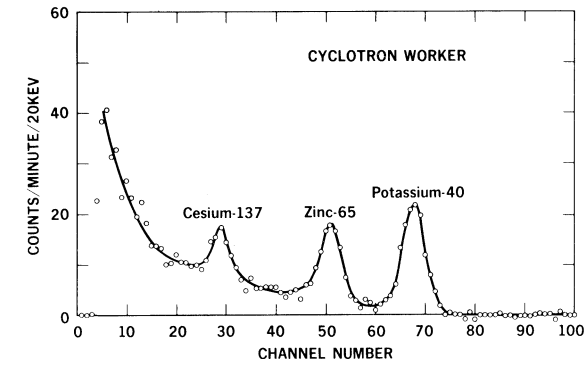


Fig. 14 - A solid crystal scanning whole body counter

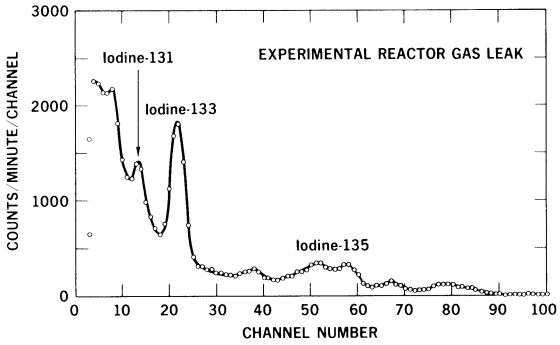
solid crystal is "scanned" over the body with a mechanical drive mechanism. An example of each is shown in Figures 13 and 14.

One big advantage of the solid over the liquid scintillation whole body counter is in energy resolution. With NaI(Tl), it is possible to distinguish gamma rays which differ by only about 50 keV in energy. This is a dramatic improvement over the 1 MeV resolution of the liquid counter. The 50 keV resolution is sufficient to identify most of the gamma emitting nuclides commonly encountered by the radiation worker. Thus, when the isotopes which make up a person's body burden are not known, the solid crystal whole body counter is the instrument of choice. The three spectra of actual accident victims shown in Figure 15 illustrate the energy resolution of a solid crystal whole body counter.

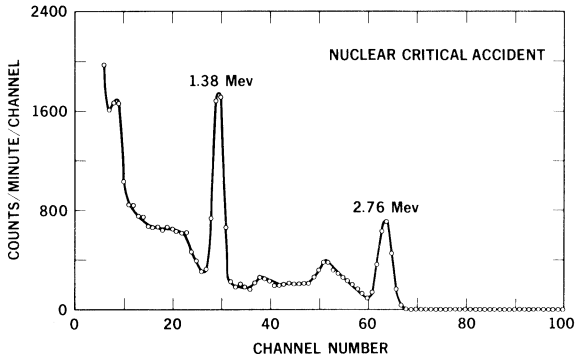
The disadvantage of the solid crystal counter is the lower counting efficiency. A common counter might show about 10% efficiency. The other problem in the solid counter is the need for shielded rooms. Placing the crystal assembly inside a heavily shielded room increases sensitivity by reducing background interference. It also gives good physical access to the counter. On the negative side, shielded rooms are very expensive. Early counters were built using below ground vaults in an attempt to cut costs. These proved not to be as advantageous as was hoped. The trace level of radioactive materials in the earth surrounding the vault causes a higher than desirable background. The preferred approach seems to be an above ground room shielded with steel or lead. A properly designed room will attenuate the background radiation level by 50 to 100 times depending on the gamma ray energies present in the background spectrum. The trick is to locate a source of steel plate which was smelted before 1945. More recently produced steel has unacceptably high levels of fallout radionuclides such as ^{137}Cs. Unfortunately, these fission product nuclides in the background spectrum are often precisely the nuclides being sought in a radiation worker. Armor plate from pre-World War II ships has been used successfully in many shielded room



Whole body counter record of evidence that a cyclotron worker has picked up higher-than-average quantities of radioactive zinc-65. Note how the graph shows peaks at specific energy levels identifying the radionuclides.



Graphic evidence that a radioactive gas accidentally inhaled by a worker at an experimental reactor contained 3 radioactive isotopes of iodine.



The number of 1.38-Mev gamma rays emitted by sodium-24 in the body of a reactor accident victim indicates he received about 900 rads of neutron irradiation.

Fig. 15 - Actual spectra from contaminated workers

designs. Figure 16 shows a large battleship steel plate shielded room at the Hanford Whole Body Counting Facility in the state of Washington. It has 10 inch thick iron walls, floor and ceiling.

The Hanford facility makes about 11,000 measurements annually, with 80% being whole body counts and 18% being lung counts. They make use of NaI(Tl) scintillation crystals up to 6 inches thick by 11.5 inches in diameter as well as germanium detectors, lithium drifted silicon detectors and CdTe. As an example of state-of-the-art capabilities, the minimum

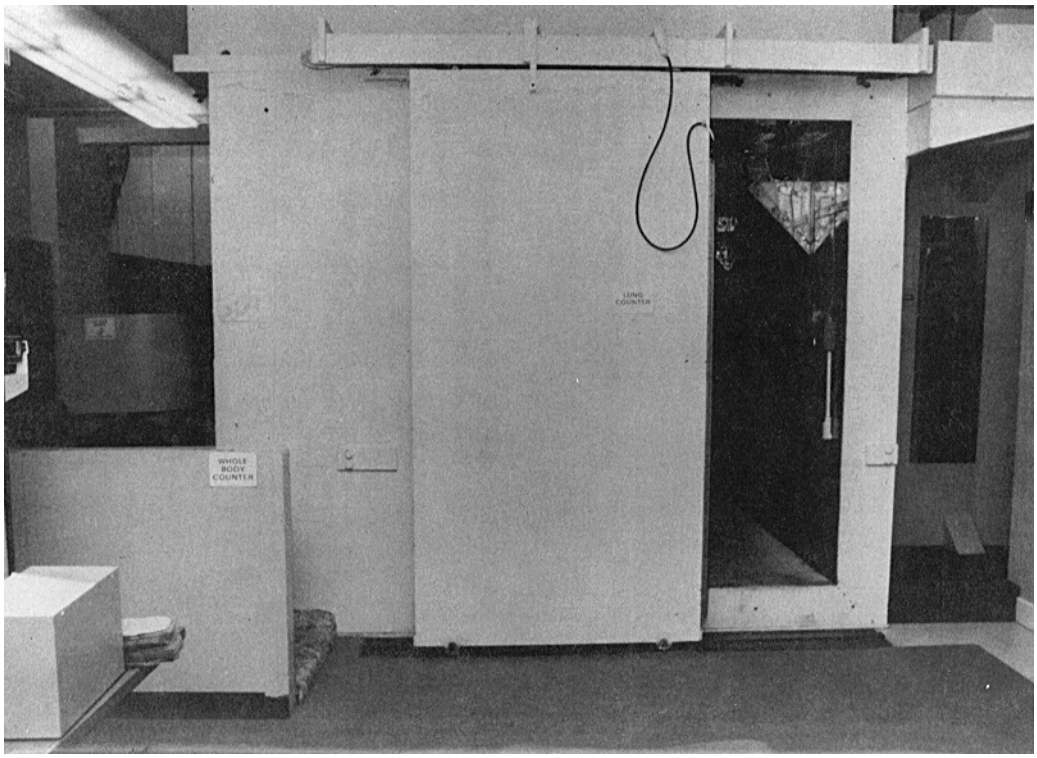


Fig. 16 - The Hanford iron shielded room

Courtesy, Hanford Laboratory

detectable amount of radioactivity that can be measured at Hanford is listed for several isotopes and organs in Figure 17.

A cheaper alternative to a shielded room is to make use of a "shadow shield." In this setup, the solid state detector is placed in a shielded collimator so it is viewing photons only from a narrowly defined angle. The body tissue being counted is placed in the acceptance angle and then the "shadow shield" is placed behind the body tissue to create a shadow region that shields the detector from all radiation that does not

NUCLIDE	TIME (sec)	ORGAN	DETECTION LIMIT (nCi)
Pu-239	2000	Lung	2.4
Pu-239	3000	Bone	5.4
Am-241	2000	Lung	0.18
I-125	2000	Thyroid	0.004
Cs-137	200	Whole Body	3
Co-60	200	Whole Body	3
K-40	200	Whole Body	15

Fig. 17 - Typical detection sensitivities of the Hanford Whole Body Counter

Courtesy, Hanford Laboratory

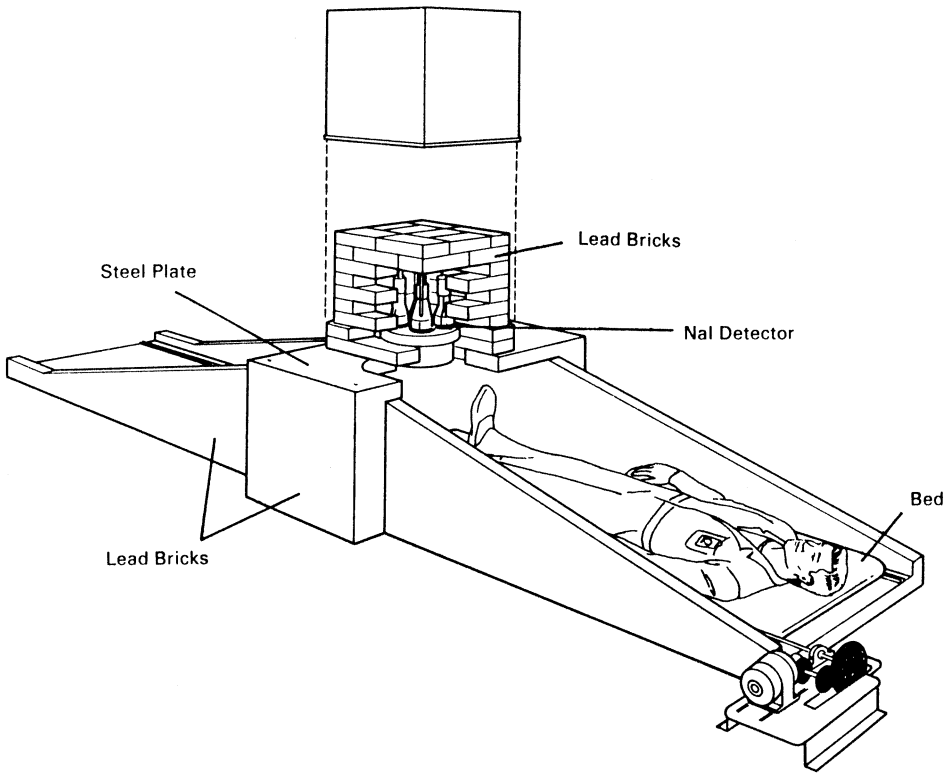


Fig. 18 - A shadow shield whole body counter

originate in the tissue. Figure 18 illustrates this technique. The shadow shield is below the patient bed in this case.

Partial Body *In Vivo* Counters

There are many practical cases in radiation protection practice in which circumstances are well enough known that it is acceptable to determine only an organ burden rather than a body burden. The classic example is, again, iodine. In the case of exposure to one of the iodine isotopes, it is necessary to determine the thyroid burden. A variety of special counters and procedures have been developed to measure organ burdens of commonly encountered radioisotopes. Thyroid counters and lung counters will be briefly mentioned here.

A typical thyroid counter setup is illustrated by Figure 19. This particular counter is designed for measuring thyroid burdens of Iodine-125. It consists of a 1 inch diameter by 1 mm thick NaI(Tl) crystal and a photomultiplier tube connected to a single channel pulse height analyzer which is adjusted to count only the pulses corresponding to the I-125 photon energies (about 22 – 40 keV). The output of the pulse height analyzer is fed to a scaler – timer – power supply unit. The system is calibrated using the neck phantom shown in Figure 20. Because of the low energy photons

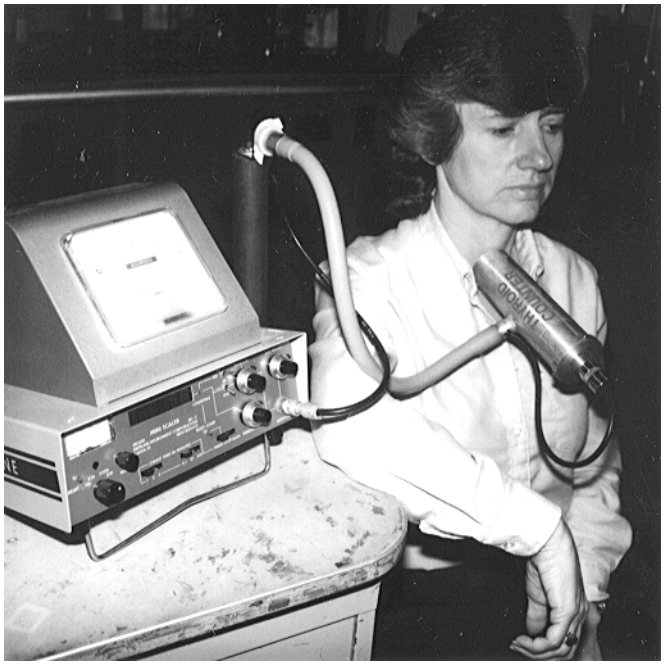


Fig. 19 - An example of a thyroid partial body counter

involved, a correction for the thyroid gland depth must be made. This is accomplished by first measuring the count rate from the I-125 directly over the gland and then taking a second reading at 90 degrees from the first, i.e., along the side of the neck. The ratio of these two counts can be used to locate the gland depth.

For those persons interested in more details on I-125 thyroid counter calibration, the following supplemental material is included. The water phantom must have an adjustable source depth. In the phantom

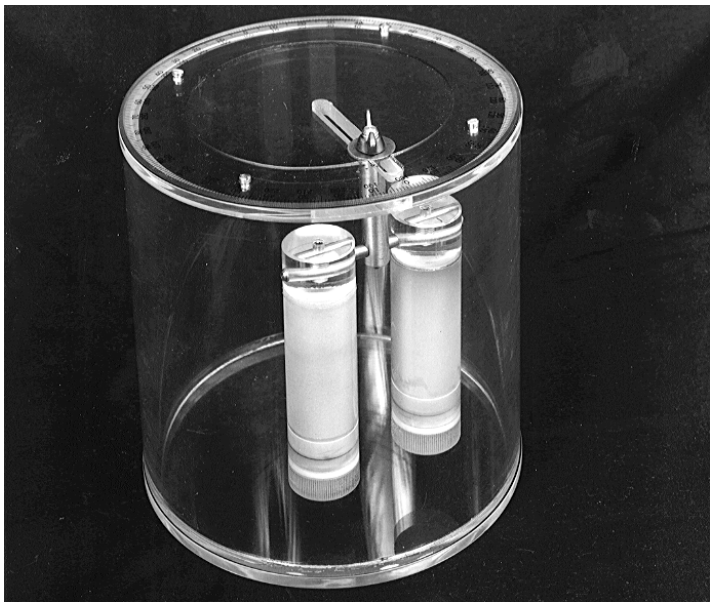


Fig. 20 - A neck phantom for calibration of thyroid counters

shown in Figure 20, the hollow "lobes" slide forward and back in the cover slot. The NaI(Tl) scintillation counter is placed in contact with the front of the water tank and counts are taken at several different source depths. If calibrated I-125 solution is not available, a solid rod source of I-129 is usually used. The "magic" conversion factor, taking into account only those photons which can penetrate the water, is:

Equivalent I-125 Activity (μCi) = I-129 Activity (μCi) X 0.528.

After the zero degree counts are taken, the phantom tank is rotated 90° and the second series of counts is taken at the same source depths as with the 0° readings. The ratio of the 90° to 0° readings is plotted on semi-log graph paper vs. source depth. Dividing the 0° counts by the number of I-125 photons emitted by the source gives the efficiency at the various depths. This data is also plotted on semi-log paper. The best fit straight lines are drawn through the data and the equation of each line is then obtained from the slope and intercept. Solving the two equations in two unknowns simultaneously gives a single equation for the counter efficiency that involves only the 90° to 0° count ratio as a variable. (As an example, the Pacific Radiation 2" diameter thyroid counter has an efficiency = $8.1 \times 10^{-3} R^{-0.95}$).

The worker's neck is counted at 0° and then 90° and the ratio, R, calculated. The R is then substituted in the above equation to get the efficiency. Finally, the thyroid activity is obtained as A (μCi) = Net cpm/[Efficiency x 2.22×10^6 dpm/μCi x 1.46 γ per disintegration]. Typically, a 3 or 4 minute count will produce a detection limit of less than one nanocurie of I-125.

Another example of a thyroid counter is the hand-held model shown in Figure 21. It was used by the Institute of Radiation Hygiene of St. Petersburg, Russia and

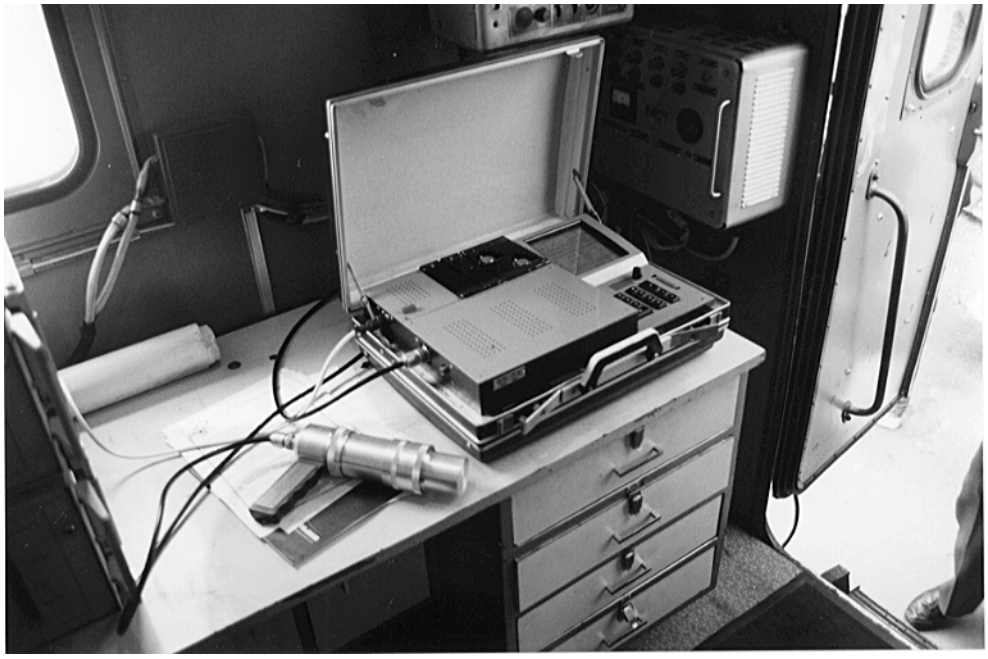


Fig. 21 - Hand held thyroid counter used at Chernobyl

Fig. 22 - Mobile lung counter in operation

Courtesy, Helgeson Nuclear Svcs. Inc. Mobile Unit

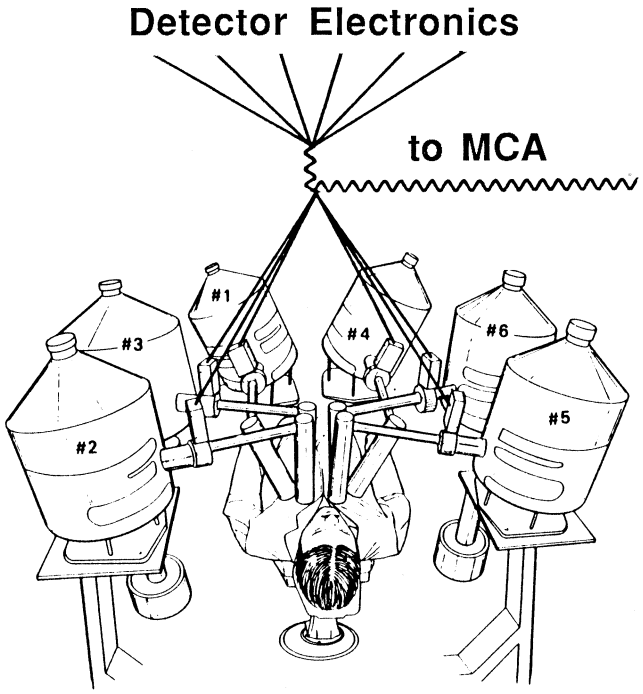
Fig. 23 - Mobile lung counter control room area

Courtesy, Helgeson Nuclear Svcs. Inc. Mobile Unit

measured I-131 thyroid burdens in Russia and Ukraine following the Chernobyl accident. The accompanying briefcase computer was used for on-the-spot data analysis.

Lung counters are useful under occupational conditions in which workers might inhale airborne radioactive contaminants. An example of a trailer mounted mobile laboratory lung counter is shown in Figures 22 and 23. The first photo demonstrates a worker receiving a lung count. Each of the two scintillation counters on the left is positioned directly over one of the lungs. The two scintillation counters over the worker's thighs are used to obtain a room background count which is subtracted from the lung count. The second photo shows the control console and waiting room area of the trailer. The minimum sensitivity of this system is reported to be about 40 micrograms for U-235, 8 nanocuries for Pu-239 and 0.3 nanocurie for Am-241. Normal counting time is from 20 to 40 minutes per person. Corrections for chest wall thickness variations will be discussed shortly.

Figure 24 shows the Hanford lung counter which makes use of 6 germanium counters, three placed over each lung. The layout on the chest is shown by the sketch in Figure 25. This arrangement has a detection limit of 2.4 nanocuries of ^{239}Pu.

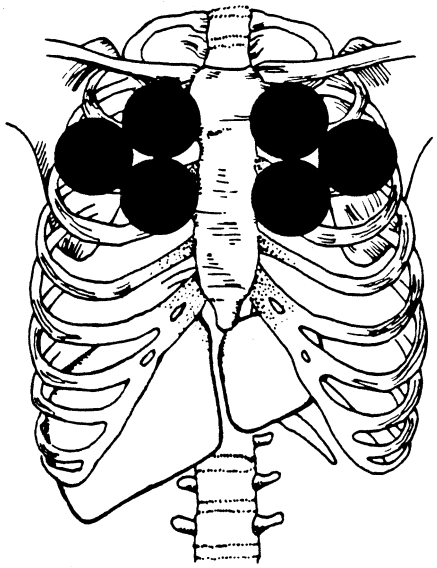


Courtesy, Hanford Laboratory

Fig. 24 - The Hanford lung counter

Data Analysis Complications

There are several factors which must be taken into account before a reasonable body burden or organ burden estimate can be arrived at. One factor is the shielding effect of the body tissues between the radioactive material and the detector. If the deposited activity is close to the surface, the count rate will be much higher than for the same activity that becomes deposited deep inside the body. This problem is particularly



Courtesy, Hanford Laboratory

Fig. 25 - Placement of Hanford lung counter detectors

acute for the case of relatively low energy photon emitters. As discussed in Chapter 3, the attenuation of gamma rays by the photoelectric effect is strongly energy dependent. Low energy photons are strongly absorbed by tissue. As an example, the 28 keV x-rays emitted by iodine-123 are 99% attenuated in passing through 10 cm of soft tissue. This is half the thickness of Reference Man. Correction factors must be developed to account for the self-shielding effect.

Another problem is to make corrections for unusually small or large persons which do not approximate Reference Man or Reference Woman. Use of different size "phantoms" filled with radioactive solutions to simulate these conditions is one approach toward a solution to the problem. The bottle phantom shown in Figure 26 uses metal containers of approximately the same size and shape as the ICRP "Reference Man" phantom. This particular phantom is at the Scientific Research Institute of Medical Radiology in Obninsk, Russia and was used to calibrate the whole body counter used extensively for Chernobyl accident victims and cleanup personnel.

Phantoms constructed of tissue equivalent plastic with removable plastic organs (see Figure 27) are also used. The replaceable organs can be molded with different amounts of radioactivity incorporated in them to allow the counter sensitivity to be accurately measured. Different chest wall thicknesses are available for this phantom to allow a correction factor to be determined.

The correction for chest wall thickness is most important in the case of low energy photon emitters deposited in lung tissue. In men, the wall thickness varies from a minimum of 2 cm to a maximum of 4 cm. In women, the variation is greater. A common problem for lung counting is determining the amount of deposited plutonium, most frequently ^{239}Pu. The radioactive decay information in Appendix A-1 indicates that the decay is 100% by alpha particle emission. However, the decay daughter product is U-235 which emits an occasional 17 keV x-ray. Thus, the lung

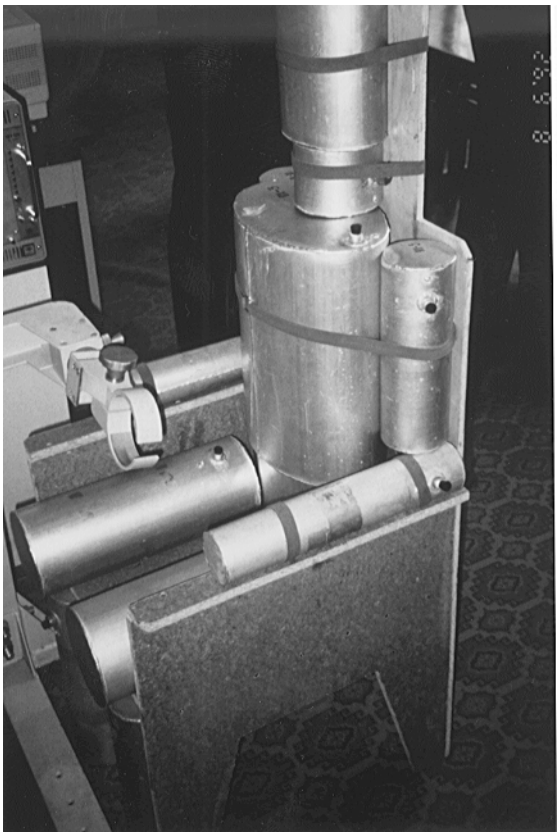


Fig. 26 - A bottle phantom for calibration, Obninsk, Russia

Courtesy, Donald Philpotts

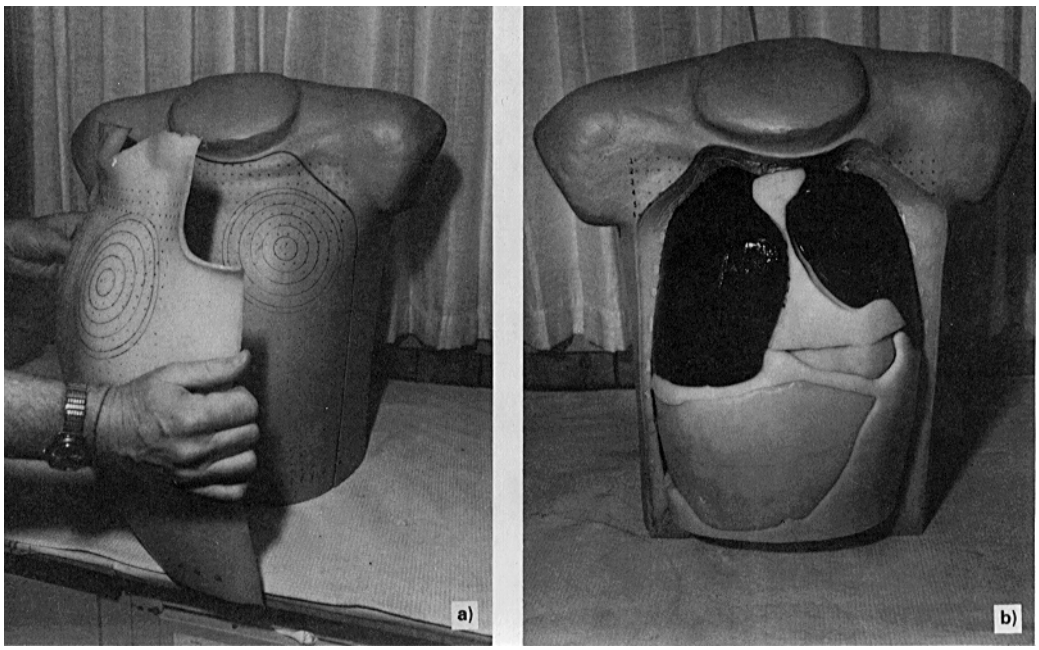


Fig. 27 - A tissue equivalent torso phantom

Courtesy, Hanford Laboratory

burden is measured by counting the 17 keV emissions. In human tissue, half of these 17 keV photons are stopped by each 0.7 cm of travel. Under these conditions, it becomes critically important to have an accurate measure of the chest wall thickness overlying the lung tissue. The method of choice is to use a medical ultrasound unit. By transmitting a high frequency sound pulse and then detecting the reflected sound waves, a direct, non-invasive measurement is made. The chest wall thickness can be determined to an accuracy of ± 1 mm by this technique. In the event ultrasound equipment is not readily available, it is possible to estimate the chest wall thickness from a person's weight and height. The following equation is used (note the mixture of metric and English units):

Wall Thickness (cm) ={0.68 + [0.974 X Weight (lbs.)/Height (in.)]}

In addition to chest wall corrections, another problem in lung counting is the possibility that the activity is actually in the ribs rather than the lung tissue. To account for this possibility, a count is usually made of bone activity by measuring the forehead of the person. Then, a correction is applied which allows the total activity measured to be divided into the lung contribution and the rib contribution. If the head is contaminated, the knee or ankle is substituted.

Sophisticated computer models have also helped develop realistic correction factors for nonstandard size workers and to account for non-uniform distribution of radionuclides in the human body. To conclude, whole body counting can be a valuable tool in the overall management of internal dose cases, both under routine and accident conditions, to keep worker doses as low as reasonably achievable.

Intake Calculations

Single Uptake Events

The results of a measurement from a bioassay lab or an *in vivo* counter lab can give information on body burdens or organ burdens at the time of the measurement but that still leaves the question of how much activity was taken in prior to the measurement. Thus, the next step in the complete bioassay or *in vivo* counting procedure is the calculation of radionuclide intakes based on the measurement results. The techniques to be described are simplified, conservative methods recommended by the U.S. NRC for a routine screening type of bioassay or *in vivo* counting program and to demonstrate compliance with regulatory limits on worker uptake. They are not intended for use in case of actual accidental uptakes. More sophisticated models are required that take into account the circumstances of the accident and the specific physiology of each victim.

The radioactivity concentrations in excreta, such as urine, or the whole body counter results must be converted to obtain the size of the radioactive intake. A number of methods have been used over the years. Some of these use computer programs that are commercially available. Other techniques allow hand calculation of results and employ complicated models that allow for many variables to be specified by the user. For purposes of screening measurements and for demonstrating regulatory

compliance, a number of simplifying assumptions can be made. One of the easiest calculational methods makes use of intake retention fractions or IRFs. An IRF gives the fraction of the initial intake activity which is present in the whole body, an organ, or in excreta at various times after the intake. For example, if the IRF for a 24 hour urine sample is 10% on day 6, then by collecting the urine and assaying it on day 6, the intake activity is 10 times the total urine sample activity (as 1/10 of the intake amount is excreted in the urine on day 6). To formalize this concept, the intake, I_0 in μCi or Bq is given by

$$\mathbf{I_0(\mu Ci, Bq) = A_t(\mu Ci, Bq) / IRF_t} \qquad \textbf{[Eqn. 1]}$$

where, A_t is the measured activity in the body or organ from a bioassay or *in vivo* count at some time t after the intake and IRF_t is the corresponding IRF value at time t. An exhaustive table of IRF values is available in NUREG/CR-4884 published by the U.S. Nuclear Regulatory Commission.

This Eqn.1 is merely another way of writing the Radioactive Decay Law first shown as Eqn. 7 in Chapter 2, where "I_0" is the renamed A_0 of Eqn. 7 and the "IRF_t" is revealed to be an exponential decay term that accounts for both the physical decay and the biological metabolism of a radioactive material.

When more than one value of IRF is given (multiple values are frequently listed to account for the differences in elimination observed for different chemical compounds of a radionuclide), the NRC recommends using the most conservative value (the smallest IRF). This will produce a result for the estimated intake that is probably higher than the actual intake, but can be used for screening or compliance demonstration purposes. Sample Problem 2 illustrates the calculation. Figure 28 is a sample of the listings for IRFs from the NUREG document.

Since the body is such a complex structure physiologically, and so many simplifications are made in the intake calculation, it is reasonable to assume that better accuracy could be obtained if several successive

Sample Problem 2

GIVEN:

A worker has an annual whole body count that shows 0.014 μCi of Cs-137 and 0.052 μCi of Co-60.

FIND:

What is the estimated intake for this worker?

SOLUTION:

Since the date of intake is unknown, the NRC's Regulatory Guide 8.9 on internal dosimetry suggests using the midpoint of the time span, i.e., six months ago. A linear interpolation between the 300 and 400 day listings for the radionuclides in NUREG/CR-4884 (not included here) gives the following whole body IRFs: Cs-137 = 5.93×10^{-2}, Co-60 = 9.37×10^{-2} or 1.16×10^{-2} depending on chemical form. The conservative assumption requires the smallest IRF when a choice is available, so the 1.16×10^{-2} is used for the Co-60. These IRFs are then simply inserted in Eqn. 1 above to give the results for the two nuclides as follows: For Cs-137, $I_0 = 0.014\ \mu Ci / 5.93 \times 10^{-2} = 0.24\ \mu Ci$. For Co-60, $I_0 = 0.052\ \mu Ci / 1.16 \times 10^{-2} = 4.5\ \mu Ci$.

CLASS D AMAD = 1 MICRON HALF-LIFE = 1.43E+01 DAYS PHOSPHORUS 32

TIME AFTER SINGLE INTAKE	FRACTION OF INITIAL INTAKE IN:			
	24-HOUR	ACCUMULATED	24-HOUR	ACCUMULATED
DAYS	URINE	URINE	FECES	FECES
1.00E-01		5.77E-03		6.83E-04
2.00E-01		1.39E-02		1.92E-03
3.00E-01		2.19E-02		3.52E-03
4.00E-01		2.96E-02		5.38E-03
5.00E-01		3.69E-02		7.39E-03
6.00E-01		4.38E-02		9.49E-03
7.00E-01		5.03E-02		1.16E-02
8.00E-01		5.64E-02		1.37E-02
9.00E-01		6.22E-02		1.58E-02
1.00E+00	6.76E-02	6.76E-02	1.78E-02	1.78E-02
2.00E+00	4.17E-02	1.06E-01	1.53E-02	3.23E-02
3.00E+00	2.50E-02	1.26E-01	7.62E-03	3.83E-02
4.00E+00	1.64E-02	1.36E-01	3.68E-03	4.02E-02
5.00E+00	1.17E-02	1.42E-01	1.98E-03	4.03E-02
6.00E+00	8.98E-03	1.44E-01	1.24E-03	3.96E-02
7.00E+00	7.19E-03	1.44E-01	8.84E-04	3.86E-02
8.00E+00	5.95E-03	1.43E-01	6.91E-04	3.75E-02
1.00E+01	4.34E-03	1.39E-01	4.86E-04	3.51E-02
2.00E+01	1.55E-03	1.05E-01	1.72E-04	2.37E-02
5.00E+01	1.24E-04	3.10E-02	1.38E-05	6.26E-03

Fig. 28 - A sample listing of IRF values as they appear in NUREG/CR-4884

results from bioassays or *in vivo* **counts were available. This would be the case in an accidental uptake where the date of intake is known.**

The recommended procedure is to take several samples or counts at intervals which are short compared to the effective half-life of the nuclide clearing from the body. This series of measurements then gives a more accurate intake estimate. The NRC recommends use of a "minimum chi-squared statistic" formula. In this case of multiple measurements, Equation 1 is replaced by Equation 2:

$$I_0 = (\Sigma_i IRF_i \times A_i) / (\Sigma_i IRF_i^2) \quad \text{[Eqn. 2]}$$

Here, the "i" subscripts identify the A and IRF values for a particular sequential measurement at some time t, i.e., the i^{th} measurement. Sample Problem 3 might clarify the arithmetic.

In actual cases, it should not be surprising that results for a specific individual show variations from the expected behavior. The elimination of the radionuclides will be affected by the food eaten by the worker, the worker's state of health, age, state of activity and by any unique physiological factors (overweight, inactive thyroid, etc.). It is also important to note that if urinalysis is the method of choice, the complete 24

Sample Problem 3

GIVEN:
A male research technician inhales a P-32 labeled compound following a broken flask accident. A series of 24 hour urine samples, corrected for decay since sampling, showed the following concentrations, in μCi/liter: 2 day = 1.5, 10 day = 0.13 and 20 day = 0.06.
FIND:
What was the intake activity of P-32?
SOLUTION:
This differs from the previous problem in that we know the intake date and have a series of bioassays. Thus, we use Equation 2. The IRF values are obtained from Figure 28 as 0.0417, 0.00434 and 0.00155 for the 2, 10 and 20 day samples respectively. In this problem, the A_i represents the total activity in each 24 hour urine. Using the default volume of 1.4 liters (see text), A_1 = 1.4 liter x 1.5 μCi/liter = 2.1 μCi for the day 2 sample. Similarly, A_2 = 0.182 and A_3 = 0.084 μCi. Applying Eqn.2 to the above data,

$$I_0\ (\mu Ci) = \frac{(2.12 \times 0.0417) + (0.182 \times 0.00434) + (0.084 \times 0.00155)}{(0.0417)^2 + (0.00434)^2 + (0.00155)^2}$$

or, I_0 = 51 μCi of P-32.

hour total day's urine output constitutes the sample. This is called a "24 hour urine" sample, and is the basis for the published IRF table values. The actual volume is used in calculations for a specific case where information is available. If not, the NRC suggests a value of 1.4 liters for the 24 hour urine volume for "Standard Man" and 1.0 liters for "Standard Woman."

Multiple or Continuous Uptakes

A single intake event is usually the result of an accident. The more common practical problem is the worker who has regular intakes of small amounts of material or is exposed continuously. The simplified IRF approach can still be used, but it is not as simple as before. Before proceeding, note that if multiple intakes are separated by more than 4 effective half-lives (the combined effect of physical decay and biological elimination, to be defined later in this chapter), each intake can be treated separately as a "single intake" and the results added together.

Multiple intakes closer together than the 4 half-lives are treated as a continuous intake by NRC recommendation. In this case, equation 1 is modified to a more complex form involving the integral of the IRF over the exposure time period. NRC Regulatory Guide 8.9, Revision 1, provides an approximation technique to evaluate the integral. It will not be discussed here.

Once the intake activity and date or dates of intake are known, then the final step in the whole process is the calculation of the dose to the worker from the deposited radioactivity. There are a variety of ways to calculate the dose from internal emitters. The next section discusses the ICRP method which is used under the 10 CFR 20 regulations in the United States.

Internal Dosimetry Calculations

Basic Principles

With the information supplied by a bioassay lab or an *in vivo* counter lab, the health physicist then must compute the actual doses that will be delivered to the person. This involves a knowledge of body physiology and then a choice of a mathematical model (set of formulas). We will begin with a discussion of some basic principles.

The human body has no built-in radiation detector. It cannot distinguish stable isotopes from radioisotopes of the same element. This just means that radioactive contamination inside the body will be treated to the same physiological processes that would occur should the stable forms of those elements be taken in. Thus, the organs of deposition are determined by the chemical compound (or element) and the physical form of the radiocontaminant.

> **The "critical organ" has been defined to be the body organ which receives the greatest damage as a result of a radioactive intake. This is normally the organ having the highest concentration of radioactivity. Since iodine tends to concentrate in the thyroid, the critical organ for radioiodine intakes is the thyroid gland. The first comprehensive internal dosimetry system available in radiation protection technology, introduced in 1959 by the International Commission on Radiological Protection, determined the critical organs for each element and then calculated the doses to those organs as a result of an intake. Protection standards were based on delivery of no more than 5 rem per year to the critical organ.**

In the late 1970s, the ICRP recognized the need for a more comprehensive approach to internal dosimetry that considered many organs – not just the critical organ for a particular isotope. Through the use of newly available sophistication in computer technology, a complete revision of the 1959 system was developed. It was released in 1979 as ICRP Publication 30, **Limits for Intakes of Radionuclides by Workers**. The necessary decay scheme information was placed in a companion report, ICRP Publication 38, **Radionuclide Transformations**. The 8 volumes total 3,893 pages of information, mostly in the form of computer printouts, and cover all the elements in the periodic table. This model gave dose information on many organs, not just the critical organ, as a result of an intake. This methodology and data are used to perform internal dose calculations under current 10 CFR 20 regulations in the USA. Details of the ICRP models will be given later in this chapter.

The last basic principle to be covered before proceeding to the calculational models is the concept of compartment modeling. This refers to picturing the human body as a collection of tanks interconnected by a plumbing system of pipes and valves. Inhalation of radioactive material is pictured as a flow of activity down a pipe leading to the tank representing the lung. Ingestion is pictured as a flow of radioactivity into the "stomach" tank. Since the tanks or compartments are interconnected, the activity becomes diluted and flows to other compartments as time passes. For purposes of radiation protection calculations, the activities are assumed to be deposited into the various starting compartments instantaneously. The concentrations of radio-

activity in the tanks are then assumed to follow the laws of physics, i.e., they decrease exponentially with the passage of time.

For the sake of accuracy, it might be noted that the ICRP has continued to update it's models since 1979. However, the regulatory standards in effect in the U.S. in 2011 use numerical results based on the old 1979 model, and there is no current interest in making any changes.

The first big ICRP change was the release of ICRP Publication 66 in 1994. This publication replaced the human respiratory model with a newer version. It incorporated new data previously unavailable on the behavior of inhaled radionuclides, gave numerical results for infants and children (Publication 30 was only for adults), and changed the default inhaled particle size to 5 microns vs. the former 1 micron particle. Instead of averaging the dose over the entire lung, the new model treats each region of the respiratory tree separately.

Another change from the ICRP included a 2 rem annual dose limit instead of the 5 rem basis for Publication 30. The gastrointestinal model of Publication 30 was replaced in 2006 with ICRP Report 100, "Human Alimentary Tract Model for Radiological Protection." Three additional model compartments were added to more accurately describe the movement of radionuclides through the body. Another detail added was data on the time spent in each compartment for ages other than adult males.

The first update of ICRP Report 23 was issued in 1995 as ICRP Report 70. It contained current information on the skeletal system. ICRP Report 88, released in 2001, provided comprehensive information on the embryo/fetus. The revision of the remainder of ICRP 23 on Reference Man was issued as ICRP Report 89, "Basic Anatomical and Physiological Data for Use in Radiological Protection: Reference Values." It appeared in 2002 and provided recommended values for organ masses and functional data for males and females of six different ages: newborn, 1 year, 5 year, 10 year, 15 year and adult.

Mathematics Of Clearance

Clearance of internally deposited radioactivity involves two totally independent and separate processes. First is loss of activity due to the physical decay of the radionuclide (covered in Chapter 2). Second is the biological removal caused by the action of normal body physiology. Since physical decay of a nuclide is independent of any outside influence, chemical, physical or biological, the normal radioactive decay law is followed by internally deposited radionuclides. The radioactive atoms are completely unaware that they are surrounded by your liver or thyroid cells. Similarly, as stressed above, the normal biological removal processes act on any compound that enters the body. The fact that this particular compound is radioactive is of no consequence – it is not even recognized by the body. In performing clearance calculations, both biological and physical clearance are assumed to follow exponential laws. The physical clearance is written as shown in Equation 3:

$$A_t = A_0 \, e^{-\lambda_p t}. \qquad \textbf{[Eqn. 3]}$$

The A_t represents the activity of the deposited radioactivity (e.g., measured in becquerels or microcuries) at some specified time, "t," in the body or in some organ, i.e., either the body burden or organ burden. The subscript "p" on the decay constant shows that this is the physical decay constant. Similarly, Equation 4 shows the biological clearance term:

$$A_t = A_0 \ e^{-\lambda_b t}. \qquad \text{[Eqn. 4]}$$

Again, the activities at the starting time and measurement time are represented by A_0 and A_t respectively. Here, the "b" subscript means that this λ (lambda) is the biological decay constant.

It was indicated above that the biological and physical clearances are completely independent. Therefore, we can write the equation for the body or organ burden vs. time, A_t, due to the combined effects of both biological and physical clearance, as follows:

$$A_t = A_0 \ (e^{-\lambda_p t}) \ x \ (e^{-\lambda_b t}) \qquad \text{[Eqn. 5]}$$

Following the rules of algebra, that exponents are added when multiplying, Equation 5 can be rewritten as:

$$A_t = A_0 \ e^{-(\lambda_p + \lambda_b) t} \qquad \text{[Eqn. 6]}$$

To simplify the arithmetic, it is convenient to define a new decay constant, the effective decay constant, λ_{eff}, as shown in Equation 7:

$$\lambda_{eff} = \lambda_p + \lambda_b \qquad \text{[Eqn. 7]}$$

Thus, Equation 6 can be rewritten as shown:

$$A_t = A_0 \ e^{-\lambda_{eff} t} \qquad \text{[Eqn. 8]}$$

Here is Equation 1 of this Chapter again, with the IRF_t ($= e^{-\lambda_{eff} t}$) shown even more explicitly to be an exponential term involving the effective decay constant. This would be correct in the simple case of a whole body IRF in which the initial intake, I_0, actually was 100% deposited uniformly in the entire body. However, in a realistic case, the λ_{eff} can be hard to determine as most radionuclides deposit in more than one organ, each with a different biological half-life.

It must be kept in mind that a decay constant (such as lambda) is always reciprocally related to a half-life. In the present case, the biological, physical and effective decay constants thus each have an associated half-life. These are illustrated in Figure 29. The effective half-life would be defined as the time it takes for the body or organ burden to fall from some starting activity (at t = 0) to half of that activity, taking into account both biological and physical clearance.

Biological Half-life = T_b = ln 2/λ_b = 0.693/ λ_b

Physical Half-life = T_p = ln 2/λ_p = 0.693/λ_p

Effective Half-life = T_{eff} = ln 2/λ_e = 0.693/λ_e

Fig. 29 - The three decay constants and half-lives

$$T_{eff} = \frac{T_b \times T_p}{T_b + T_p} \qquad \text{[Eqn. 9]}$$

Fig. 30 - Relationship between the three half-lives

T_{eff} plays an extremely important role in internal dosimetry. The internal dose of a person is DIRECTLY PROPORTIONAL to the effective half-life. The longer it takes for the radioactivity to diminish, the greater the dose to the person. The mathematical relationship between the half-lives is shown in Figure 30. Also, see Sample Prob. 4.

Sample Problem 4

GIVEN:
The biological clearance of iodine from the thyroid has a half-life of 110 days according to NRC recommendations.
FIND:
What is the effective half-life for I-125?
SOLUTION:
From Appendix A-1, the physical half-life of I-125 is 60.2 days. From Figure 29, the effective half-life is the product divided by the sum of the biological and physical half-lives. Thus,
T_{eff} = [110 (da) x 60.2 (da)] ÷ [110 (da) + 60.2 (da)] = 6622 / 170.2 (da) or
T_{eff} = 38.9 days.

The ICRP Internal Dosimetry Models

Committee II of the International Commission on Radiological Protection developed the first comprehensive mathematical model for internal dosimetry calculations for radiation protection purposes. It was released as ICRP Publication 2 in 1959. A brief description of some of the assumptions used follows.

In this model, X and gamma rays were assumed to be absorbed exponentially with distance through tissue. Particulate radiations (alpha particles and beta particles) were assumed to deposit ALL of their energy in the ORGAN WHERE THEY DEPOSITED. Body organs were all assumed to have the density of water and the shapes were either spheres or cylinders. No allowance was made for low density lung tissue or high density bone. Isotopes were assumed to deposit uniformly throughout an organ. With these assumptions, a mathematical model was developed to estimate the dose that was delivered to an organ as a result of the presence in that organ of a unit quantity (one microcurie) of activity.

This model served the needs of health physicists for some 20 years. During that time, limitations of the model became evident. Problems with choice of a too simplistic approach to lung dosimetry were solved with

release of a revised lung model in 1965. Another fundamental limitation could not be fixed as easily. The need arose, particularly in the field of nuclear medicine, of being able to determine the dose to the gonads (sex organs) as a result of radioactivity being deposited in OTHER body organs. This information was used to determine the relative suitability of different radiopharmaceuticals for nuclear medicine procedures. Before a new radioactive drug can be approved for human use the radiation safety implications must be evaluated. These radioactive drugs are administered in activities which are so low that there is no risk of direct somatic radiation damage to the patient, but the possibility of increased risk of genetic damage cannot be ruled out. Unfortunately, the 1959 ICRP model only allows the dose to be calculated in the organ where the activity deposits. Since virtually no commonly used radionuclides concentrate primarily in the gonads, the 1959 model was not usable.

With the 1979 release of ICRP Publication 30, the old internal dosimetry model was completely replaced. The new model provided much additional information compared to the old.

The major change was in the way photon absorption was dealt with. The new model used a computer to trace the histories of hypothetical photons randomly released from different organs in a human body. Organ shapes assumed were anatomically correct. The correct composition and density were also used for bone and lung. Figure 31 is a list of the organs used and their masses. By the computer keeping track of where each photon deposited energy, it was possible, after tracking millions of hypothetical photon histories, to construct a reasonable list of probable doses to all body organs as a result of a unit quantity of a photon emitting radionuclide being deposited in a single body organ. This technique is called a Monte Carlo calculation since it involves playing "a game of chance" with the photons. Figure 32 shows some of the features of the ICRP mathematical phantom used for the new model.

Source organs	Mass (g)	Target organs	Mass (g)
Ovaries	11	Ovaries	11
Testes	35	Testes	35
Muscle	28 000	Muscle	28 000
Red marrow	1 500	Red marrow	1 500
Lungs	1 000	Lungs	1 000
Thyroid	20	Thyroid	20
ST content	250	Bone surface	120
SI content	400	ST wall	150
ULI content	220	SI wall	640
LLI content	135	ULI wall	210
Kidneys	310	LLI wall	160
Liver	1 800	Kidneys	310
Pancreas	100	Liver	1 800
Cortical bone	4 000	Pancreas	100
Trabecular bone	1 000	Skin	2 600
Skin	2 600	Spleen	180
Spleen	180	Thymus	20
Adrenals	14	Uterus	80
Bladder content	200	Adrenals	14
Total body	70 000	Bladder wall	45

Fig. 31 - ICRP Model organs and masses

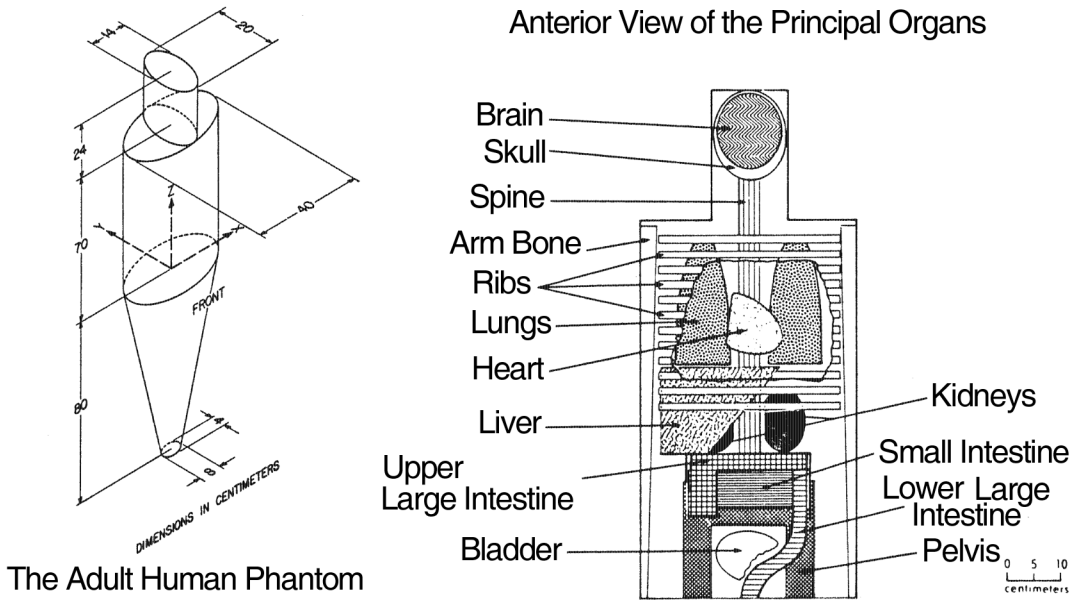


Fig. 32 - The ICRP phantom

A simplified drawing of the compartment model used for calculations is shown in Figure 33. The boxes represent the different major body regions involved and the connecting lines are possible pathways that radioactivity can follow in transporting from region to region as it clears. The two entry routes of inhalation and ingestion are also indicated.

Once the basic principles and assumptions were established, the ICRP used supercomputer technology to calculate dose data for virtually every known radioisotope with a half-life greater than 10 minutes. These computer printouts comprise the major portion of the ICRP Publication 30 report series. The committed dose equivalent for a unit uptake, Sv/Bq, is tabulated for each nuclide for any body organ that receives at least 10% of the dose.

The term "Committed Dose Equivalent" or CDE, first discussed in Chapter 5, was defined in Publication 30 by the ICRP as "the total dose equivalent averaged throughout a tissue in the 50 years after intake of a radionuclide into the body." The recommended symbol is H_{50}. This fundamental quantity is the basis of internal dosimetry. Multiplying the CDE by the weighting factors, w_T, defined in Chapter 5, leads to the committed effective dose equivalent, or CEDE, which is the internal component of the total dose to a worker.

A closely related quantity is the annual limit on intake or ALI. The ALI is the activity of a radioactive material, in becquerels or microcuries, that would have to be taken into a human body in order to exactly deliver 0.05 Sv (5 rems) of committed effective dose equivalent to the body or a committed dose equivalent of 50 rems to any individual organ or tissue. To keep internal doses within the dose limits, workers cannot take more than the ALI of a radionuclide into the body each year. If more than one

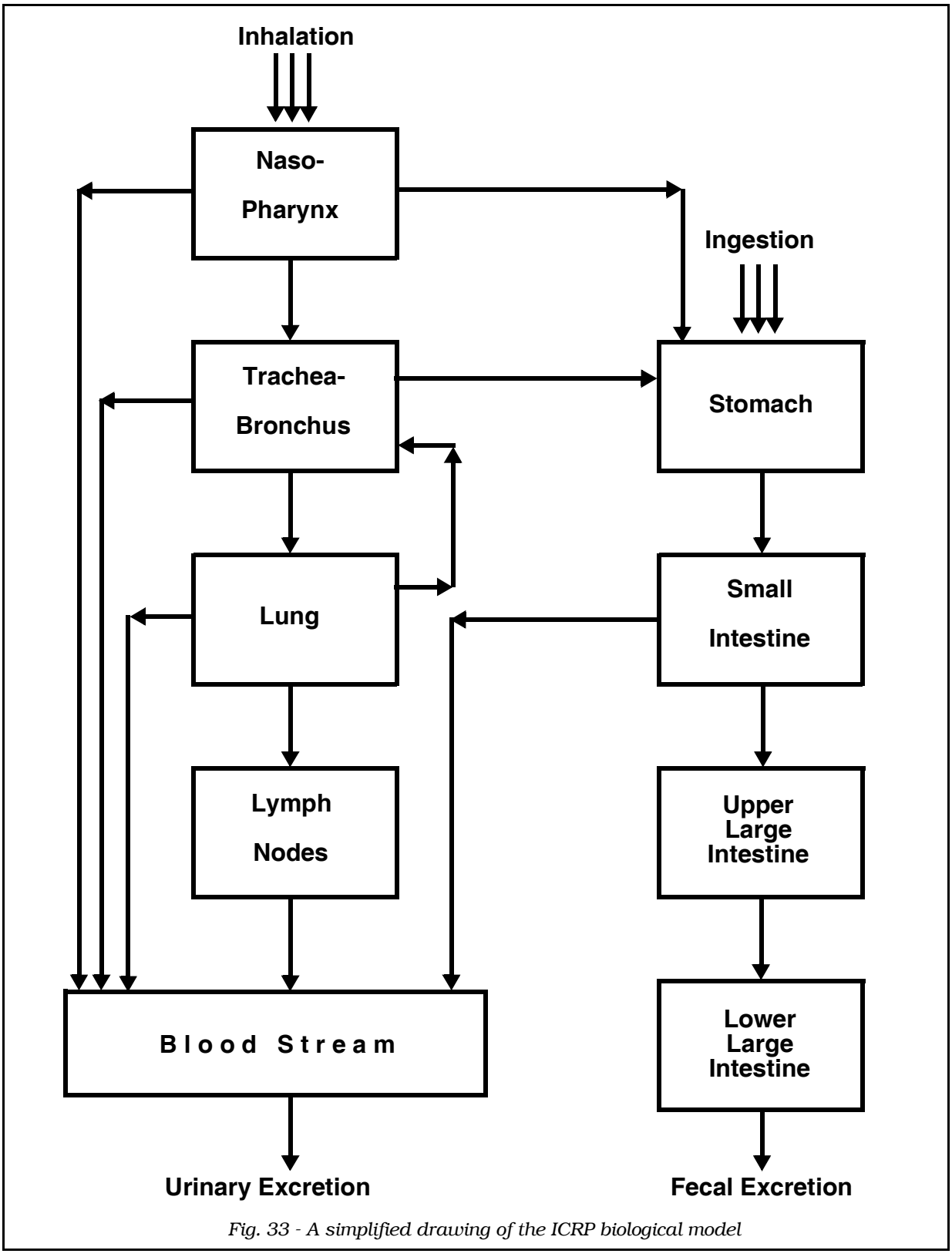


Fig. 33 - A simplified drawing of the ICRP biological model

nuclide is taken in, the various activities divided by their respective ALI values cannot add to a fraction greater than 1.0.

To expand on this idea a bit more, Figure 34 is an excerpt of selected radionuclides from Appendix B of the 10 CFR 20 regulations for U.S. workers. Two values of ALI are listed for each radionuclide. The ingestion ALI is the limit for oral intakes by mouth and the inhalation ALI is the limit for intakes which are breathed in through the nose. Sometimes the inhalation ALI has more than one value because different chemical compounds behave quite differently in the body. In that case, Appendix B identifies the compounds which correspond to the different values of ALI. Some Sample Problems in the next section will make use of these ALI values and show the practical usefulness of the ALI concept to a radiation protection technologist.

For the sake of non-U.S. readers, the ALI concept is used, of course, in the rest of the world since it was "invented" by the ICRP. But there is a slight problem - the ICRP currently bases its ALI values on a CEDE of 0.02 Sv (2 rems), not the 0.05 Sv used in the USA. The new ALI values based on 0.2 Sv are given in ICRP Publication 61.

For completeness, a comment is in order on the internal dose produced by radionuclides injected into tissues as a result of a wound. This question was raised in Chapter 5 and mentioned briefly. If a technologist needs information on wound dosimetry, the definitive work was published in 2006, as NCRP Report Number 156, **Development of a Biokinetic Model for Radionuclide-Contaminated Wounds and Procedures for Their Assessment, Dosimetry and Treatment.** The work was a collaboration between both the NCRP and the ICRP.

One final point before proceeding with the dose calculations. Since we have seen that the dose equivalent received as a result of an internal uptake of radioactive

Nuclide	Class	Ingestion ALI (μCi)	Inhalation ALI (μCi)
Hydrogen 3	Water	80,000	80,000
Phosphorus 32	Most compounds	600	900
Potassium 40	All compounds	300	400
Cobalt 60	All except oxides, hydroxides, halides and nitrates	500	200
Cobalt 60	All others	200	30
Technicium 99m	All except oxides, hydroxides, halides and nitrates	80,000	200,000
Technicium 99m	All others	–	200,000
Cesium 137	All compounds	100	200
Plutonium 239	All except PuO_2	0.8	0.006
Plutonium 239	PuO_2	–	0.02
Americium 241	All compounds	0.8	0.006

Fig. 34 - Some ALI values from Title 10 Code of Federal Regulations Part 20

material is proportional to the effective half-life of clearance, it might occur to you that the dose to an internally contaminated person could be reduced if we could alter the effective half-life. Of course, the physical half-life component is unalterable. However, in many cases it is possible to affect body physiology to some extent. For example, in the case of a contaminant distributed in body water, the biological clearance time can be reduced by force feeding liquids or by the use of diuretics which speed up kidney function. Since these are medical procedures, they must be performed under the direct advice and care of a licensed physician. This topic will be covered in more detail in Chapter 14 on nuclear emergencies.

Practical Internal Dose Calculations

Finally, the time has come to illustrate the actual calculation of the dose. The methods discussed here are acceptable to the U.S. NRC for occupational worker calculations. There are two types of internal components needed to satisfy the law. The Committed Effective Dose Equivalent, CEDE, must be calculated to show compliance with a 0.05 Sv (5 rem) total body limit and the organ-specific committed dose equivalent, CDE, must be computed to comply with a 0.5 Sv (50 rems) annual organ dose limit.

The concept for calculating the CEDE following an intake is quite simple. By definition, the ALI will deliver a CEDE of 5 rems. Thus, if a worker takes in the ALI amount of a nuclide, the CEDE is 5 rem. However, the CEDE is proportional to the intake activity. For example, if a worker inhales 1/10 of an ALI, the internal dose (CEDE) would be 0.5 rem, i.e., 1/10 of 5 rems. In general, the equation for internal dose to the body is

$$\textbf{CEDE } (H_{50}) = \textbf{5 rems x I } (\mu Ci) / \textbf{ ALI } (\mu Ci). \qquad \textbf{[Eqn. 10]}$$

To use the above formula, the ALI must be for the correct chemical compound ("Class" in Figure 34) and from the correct column, inhalation vs. ingestion. Two examples are given, in Sample Problems 5 and 6.

Sample Problem 5

GIVEN:

A worker is found to have a body burden of 0.187 mCi of Cs-137 immediately following an ingestion accident.

FIND:

Calculate the CEDE for this case.

SOLUTION:

Since the body burden was measured at the time of uptake, the intake = the body burden = 187 µCi = I_0. From Fig. 34, ingestion ALI = 100 µCi. So, using Eqn. 10,

H_{50} = 5 rems x 187 µCi/100 µCi = 9.35 rems

Sample Problem 6

GIVEN:

The technician from Sample Problem 3, this chapter, inhaled P-32.

FIND:

What is the CEDE for this worker?

SOLUTION:

The bioassay calculations in Sample Problem 3 showed an intake value of I_0 = 51 μCi of P-32. From Fig. 34, inhalation ALI = 900 μCi. So, using Eqn. 10,

H_{50} = 5 rems x 51 μCi/900 μCi = 0.28 rems

The second determination, the organ-specific dose, is not always a requirement. The purpose of the limits is to assure that no single body organ exceeds an annual dose of 0.5 Sv (50 rems). This requirement is met if the CEDE for a worker does not exceed 0.012 Sv (1.2 rems) in a year. Thus, the NRC does not require any organ-specific calculations unless the internal component, CEDE, is above 1.2 rems. If the calculation is required (CEDE > 1.2 rems for a worker) then the preferred method is to use the organ-specific committed dose equivalent conversion factors tabulated in "Federal Guidance Report #11" published by the Environmental Protection Agency as EPA-520/1-88-020 in 1988. The document has separate conversion factor tables for inhalation and for ingestion of radionuclides. The dose per unit intake conversion factor is listed for the various organs in units of Sv/Bq. To convert to the old units, multiply the factor by 3.7 x 10^6 rem/μCi per Sv/Bq. Note that the CDE without weighting factors is the correct dose for this situation. Sample Problem 7 continues the Cs-137 example of Sample Problem 5 and illustrates this calculation.

One word of caution. The organ-specific dose calculation above is strictly correct only if no external deep dose equivalent was received by this worker. If an external dose was received, then the total annual reported external deep dose equivalent for the worker must be added to the organ-specific CDE to give the total organ dose equivalent, TODE. This total is then compared against the 50 rem limit.

A real world problem that sometimes confronts a practicing radiation protection technologist is the requirement to calculate the total body or organ dose for a time period less than the regulatory value of 50 years. This can be done quite easily if the H_{50} dose, either CDE and/or CEDE, is known. The dose for a period of time from intake, i.e., time = 0 until a time t is just H_{50} x $(1 - e^{-\lambda_{eff} t})$. Similarly, the dose delivered to a person during an interval starting at time t_1 and running until t_2 is just H_{50} x $(e^{-\lambda_{eff} t_1} - e^{-\lambda_{eff} t_2})$.

Dose Calculation for Embryo/Fetus

The U.S. 10 CFR 20 regulations limit the dose that an embryo/fetus can receive to 5 mSv (0.5 rem) during the entire pregnancy. This limit applies to the sum

Sample Problem 7

GIVEN:

The worker with the 187 μCi Cs-137 ingestion in Sample Prob. 5 exceeded the 1.2 rem CEDE guideline.

FIND:

Has the 50 rem organ-specific limit been exceeded for this worker?

SOLUTION:

The ^{137}Cs conversion factors, obtained by consulting Table 2.2 (ingestion), Federal Guidance Report #11 (not available here) are reproduced as follows:

Dose Per Unit Intake (Sv/Bq)					
Gonad	**Breast**	**Lung**	**R Marrow**	**B Surface**	**Thyroid**
1.4E-8	**1.2E-8**	**1.3E-8**	**1.3E-8**	**1.3E-8**	**1.3E-8**

The CDE (dose with NO weighting factor) is calculated by simply multiplying the intake, I_0, by the conversion factors for the organs AND by the SI to old unit conversion factor. Thus,

CDE (rem) = I_0 (μCi) x Dose Per Unit Intake (Sv/Bq) x 3.7 x 10^6 rem-Bq/μCi-Sv.

Choosing "gonad" as an example, CDE = 187 (μCi) x 1.4 x 10^{-8} (Sv/Bq) x 3.7 x 10^6 rem-Bq/μCi-Sv = 9.7 rems. Similarly, the other organ-specific doses are "breast" = 8.3 rems, "lung," "red marrow," "bone surfaces," and "thyroid gland" = 9.0 rems each. The highest exposed organ in this example is, thus, the gonads. With a dose of 9.7 rems, the 50 rem limit was not exceeded.

of the deep dose equivalent to the pregnant worker and the internal dose to the embryo/fetus. External doses must be monitored if it is likely that the worker will exceed 0.5 mSv (50 mrem) for the entire gestation period. Radionuclide intakes by a pregnant worker must be monitored if they are likely to exceed 1% of the ALI values during the pregnancy.

Determining the dose to the embryo/fetus from internal uptake of radioactive material by the mother is extremely difficult. These intakes may be from occupational or environmental exposure as well as from medical administration. The maternal radioactivity can be transferred to the embryo/fetus through the placenta. The ease with which this occurs depends markedly on the age of gestation, the radionuclides involved and their chemical form. Much of the biological information comes from animal experiments using rats, guinea pigs, rhesus monkeys, mice and hamsters.

Even the physics of the interactions is complicated. In the case of adult organs, alpha and beta emitters are usually absorbed in the organ of deposition. In the embryo/fetus, the tiny organ sizes may well be smaller than the range of these particles so the organ dose is relatively less. (At 30 days gestation, the embryo is only 4 mm long. By 90 days, it is about 10 cm long.)

The NRC has accepted a two level approach for the internal dose component. A simplified, worst case set of dose conversion factors has

been presented in Regulatory Guide 8.36 for compliance purposes. If the worst case calculation shows a result close to or exceeding the 5 mSv limit, a more detailed time-dependent approach is suggested by the NRC. Tables of dose conversion factors that take into account the actual time of intake during the 9 month pregnancy can be used to get a more accurate estimate of the true dose to the embryo/fetus. Consult the Regulatory Guide for more details on this time-dependent technique.

The simple worst case method is based on the working equation:

$$DE_{9\ months} = I\ (\mu Ci)\ x\ f_1\ x\ DF \qquad [Eqn.\ 11]$$

In this equation, the quantity calculated, $DE_{9\ months}$, is the unweighted dose equivalent to the embryo/fetus for the entire 9 month gestation period. The I represents the uptake at any point during the pregnancy (recall, this is a worst case calculation), and the f_1 is the fraction of the inhaled or ingested intake activity that reaches the mother's blood. The appropriate f_1 value is found in Regulatory Guide 8.36 in Table 1 for inhalation uptakes and in Appendix B for ingestion uptakes. Finally, DF is the embryo/fetus dose conversion factor from Table A-1 of the Reg. Guide.

To illustrate the worst case method, a brief excerpt of data, from the Reg. Guide, for a few nuclides is given in Figure 35. To make the figure manageable, inhalation f_1 values are only given for Class D compounds (those which clear the lungs with less than a 10 day biological half-life) except for the cobalt nuclides and plutonium which have no Class D compounds. The arithmetic is shown by the calculations included in Sample Problem 8.

Nuclide	Ingestion f_1	Inhalation f_1	DF (rem/μCi)
H-3	1.0	0.63	5.87E-5
Co-57	0.3	0.27	2.12E-3
Co-60	0.3	0.27	3.80E-2
Sr-90	0.3	0.525	5.22E-2
Tc-99m	0.8	0.60	3.32E-5
I-125	1.0	0.63	1.38E-3
I-131	1.0	0.63	3.64E-3
Cs-137	1.0	0.63	5.94E-2
Pu-239	0.001	0.12	5.22E-2

Fig. 35 - Excerpts from Regulatory Guide 8.36

Summation of External & Internal Dose

Introduction

At this point, the discussion returns to the general topic of both Chapters 8 and 9, that is, personnel dosimetry. The last remaining question is how to enter dose

Sample Problem 8

GIVEN:

A pregnant nuclear medicine technologist accidently inhales 30 MBq of Tc-99m while preparing the morning's radiopharmaceuticals.

FIND:

Does the internal dose exceed regulatory limits for the embryo/fetus?

SOLUTION:

The intake, in μCi is 30 MBq x 27 μCi/MBq = 810 μCi. The f_1 (for inhalation) and DF factors are taken from Figure 35.

Applying Eqn. 11:

$DE_{9\ months} = I \times f_1 \times DF = 810\ \mu Ci \times 0.60 \times 3.32 \times 10^{-5}\ rem/\mu Ci = 1.6 \times 10^{-2}$ rem.

Since this is less than the 0.5 rem limit, the accidental dose does not exceed regulatory limits in this worst case calculation. A more accurate time-dependent calculation is not warranted for this situation.

information on the employee's personnel dose record. As mentioned above, for internally deposited nuclides the committed dose is the total internal dose received by the person over the next 50 years. But it must be remembered that the dose rate is decreasing exponentially with time as the radionuclide clears and decays. Half of the total dose that would ever be received will be delivered in the first half-life after uptake, 75% will be received in two half-lives, etc. It can be shown mathematically that over 99% of the total dose will be received in the first 7 half-lives following intake. Based on experimental evidence of clearance of radionuclides in humans, it turns out that, with the exception of a few bone seekers, most radionuclides have pretty well cleared the body within one year after uptake. Thus, the ICRP recommends, and the U.S. Nuclear Regulatory Commission agrees, that the Committed Dose Equivalent be assigned to the employee in the year of intake as if the entire dose equivalent is actually delivered that year.

To comply with radiation protection regulations in the USA, it is necessary to add together the external and internal doses for a worker to arrive at a Total Effective Dose Equivalent. This value is then compared with the 50 mSv (5 rem) limit to determine whether the worker is in compliance for the year.

Compliance Reporting of Dose

For purposes of compliance with U.S. federal regulations designed to protect workers in the occupational environment, annual reporting of the total annual dose and the lifetime accumulated dose must be made to the worker and, in specified cases, to the NRC.

There are two NRC Forms which are useful in personnel dosimetry. NRC Form 4 is basically a summary of the lifetime radiation dose history, year by year, by employer. The form includes internal and external doses for the current year and the total effective dose equivalent for past years. Provision is made to record both routine doses and those received as "Planned Special Exposures." NRC Form 5 is a very detailed report of all doses of regulatory interest for the current year. Together, NRC Forms 4 and 5 constitute the complete up-to-date dose history of a worker. Individual workers should maintain these forms in a safe place. Under current law, a radiation worker is not eligible for planned special exposures until Forms 4 and 5, or equivalent, are presented to an employer.

The case of an embryo/fetus is a special circumstance. The licensee is required to keep a record of the 9 month dose but does not have to file Forms 4 or 5 for the embryo/fetus.

Some comments on the correct way to fill out a Form 5 are in order. Form 4 is just a less detailed summary of data from Form 5. The Deep Dose Equivalent (Whole Body) dose entry would normally be made directly from the personnel badge processor report. There are separate entries for "eye dose equivalent" (specifically to the lens of the eye) and "shallow dose equivalent" which is further subdivided into "skin" and "extremities."

Radionuclide intakes are entered separately, one entry per nuclide. The solubility class must be noted. Class D compounds biologically clear with a half-life less than 10 days, Class W compounds have between 10 and 100 day clearance half-lives and Class Y compounds are over 100 days. The intake activity, estimated from bioassay or *in vivo* counter measurements, is recorded along with the intake "mode," inhalation, ingestion, injection by wound or absorption. The committed effective dose equivalent, CEDE, is calculated using Equation 10 in this chapter. If the CEDE exceeds 1.2 rems, then the organ-specific committed dose equivalent, CDE, for the organ with the highest dose is entered. The tables in Federal Guidance Report #11 are used to make this determination (see Sample Problem 7).

Don't forget to add the annual deep dose equivalent to the whole body (Block 11) to the CDE for the maximally exposed organ to obtain the "Total Organ Dose Equivalent," TODE. Finally, the "Total Effective Dose Equivalent," TEDE, is calculated. It is merely the sum of the deep dose equivalent (Block 11) plus the CEDE (Block 15). With that, the topic of personnel dosimetry is closed!

Problem Set

1. Name the two different types of procedures that are used to measure the amount of radioactivity internally deposited in a person. What are the advantages and limitations of each?

2. What kinds of body tissues and/or elimination products are most often used for analysis.

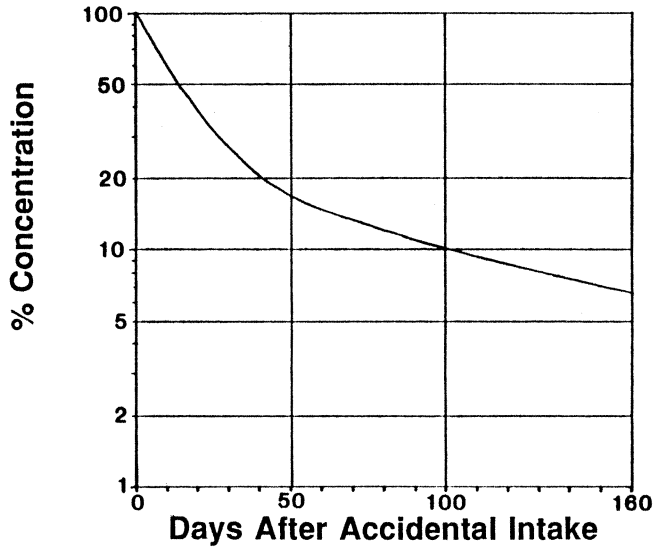
3. Under what conditions is fecal analysis required?

4. Name the possible entry routes for radionuclides going into the body. Which of these routes can be eliminated through effective use of protective clothing?

5. What does the term "bone seekers" refer to? Name some. Why are they placed in a separate category for internal dosimetry purposes?

6. What is a "Reference Adult Male" and a "Reference Adult Female?" Are you likely to ever meet either one?

7. A worker has received an accidental internal uptake of radioactivity. Urine samples are collected daily and counted. When the concentration is plotted on semi-log paper, the curve shown below is obtained. What conclusions can be drawn about the contaminant? Estimate the relevant half-lives for this worker. Are these effective, biological or physical half-lives?



Graph for problem 7

8. Briefly describe the occupation of "radium dial painting." How did the practice of this profession lead to the need for internal dosimetry models?

9. Under accident conditions, list the actions to be taken, and the order in which they are performed, in collecting urine samples and nasal swabs for bioassays.

10. Why should urine samples be collected from persons who were not involved following a radiation accident involving release of soluble contaminants?

11. Name some types of facilities where tritium might likely be handled. What special internal dosimetry problems would be involved in such facilities? What is the usual bioassay method for tritium?

12. Name the two types of whole body counters. List the advantages and disadvantages of each type.

13. Why is pre-WW II steel preferred for shielded rooms that are used for whole body counting?

14. A worker submits a urine sample which shows a concentration of 57 microcuries of H-3 per liter. He recalls an incident 4 days earlier in which he might have accidentally received the contamination. Calculate the initial body burden at the time of the accident. Calculate the activity in his body water at the time the sample was submitted.

15. Why do internally deposited radioisotopes follow the same routes as stable isotopes of the same elements?

16. How was the concept of critical organ used to establish radiation protection standards for internal dose situations?

17. Calculate the effective half-life for a nuclide with a physical half-life of 27 hours and a biological decay constant of 0.008 per minute.

18. A single 24 hour urine sample of 1.1 liters was collected from a worker who had inhaled P-32 10 days before the sample was obtained. The sample was sent to the counting lab via the U.S. Postal Service which misplaced it, and then finally delivered it 14.3 days after the sample was collected from the victim. It was promptly counted to give 75 Bq/ml. If this worker had no other internal or external exposures for the year, has the annual 50 mSv dose limit been exceeded?

19. A worker at a nuclear power station is found to have a body burden of 83 microcuries of Cs-137 during a routine quarterly whole body count. His count in the preceding period was at background. Calculate the CEDE associated with this situation, assuming he inhaled the material. The 30, 60 and 90 day total body IRFs for inhalation (from NUREG/CR-4884) are 0.476, 0.394 and 0.325 respectively.

20. It has been determined that the likely isotope to be involved in a significant internal uptake accident at a radioactive source manufacturer is Am-241. This nuclide is used to make americium-beryllium neutron sources. Calculate the ratio of the dose equivalents to bone surfaces compared with the gonads for both inhalation and ingestion intake routes. Dose Factors from Federal Guidance Report 11 are tabulated as follows:

CDE (Sv/Bq) for Americium-241		
Organ	Inhalation	Ingestion
Gonad	3.25×10^{-5}	2.70×10^{-7}
Breast	2.67×10^{-9}	2.62×10^{-11}
Lung	1.84×10^{-5}	3.36×10^{-11}
Marrow	1.74×10^{-4}	1.45×10^{-6}
Bone Surfaces	2.17×10^{-3}	1.81×10^{-5}
Thyroid	1.60×10^{-9}	1.32×10^{-11}

21. How much americium-241 could a worker with an annual DDE of 0.8 rem to date inhale during the remainder of that calendar year and not exceed the TODE limit? Use Problem 20 chart for Federal Guidance Report 11 values.

22. Show by calculation that less than 1% of a deposited radionuclide remains in the body after 7 effective half-lives have elapsed.

S-1. Why are <u>two</u> neck counts taken per worker when measuring thyroid uptake of ^{125}I?

S-2. What are two practical methods of estimating chest wall thickness for the purpose of making a correction in lung counting data?

S-3. Why are tritium urinalysis samples often collected and stored in <u>glass</u> containers?

S-4. Name two different types of "correction factors' which must be applied to data taken in an *in vivo* counter facility. What techniques might be used to obtain values for each of these factors?

S-5. What were two of the limitations in the 1959 ICRP Model? What major change in calculational technique was incorporated in the newer 1979 ICRP Model?

S-6. A pregnant worker with an external deep dose equivalent since conception of 135 millirem accidently ingests 30 μCi of I-125. Has she exceeded her dose limit for the embryo/fetus?

Other Resources

1. "Hanford Whole Body Counting Manual," H.E. Palmer, et. al., U.S. Department of Energy Report PNL-6198, Pacific Northwest Laboratory, Richland, WA, 1987. (Available from National Technical Information Service.)

2. “Use of Bioassay Procedures for Assessment of Internal Radionuclide Deposition,” NCRP Report Number 87, Bethesda, MD, 1987.

3. “General Concepts for the Dosimetry of Internally Deposited Radionuclides,” NCRP Report Number 84, Bethesda, MD, 1985.

4. “Interpretation of Bioassay Measurements,” E.T. Lessard, *et. al.*, NUREG/CR-4884, Brookhaven National Laboratory, July 1987. This two volume report contains the IRF values used in internal dose calculations in the USA.

5. “Interpretation of Bioassay Measurements,” Regulatory Guide 8.9, Revision 1, U.S. Nuclear Regulatory Commission, 1993. Current as of 2011.

6. “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” Federal Guidance Report #11, EPA-520/1-88-020, Environmental Protection Agency, 1988.

7. “Radionuclide Exposure of the Embryo/Fetus,” NCRP Report Number 128, Bethesda, MD, 1998.

8. “Radiation Dose to the Embryo/Fetus,” Regulatory Guide 8.36, U.S. Nuclear Regulatory Commission, 1992. Current as of 2011.

9. “Doses to the Embryo and Fetus from Intakes of Radionuclides by the Mother,” ICRP Publication 88, 2002 Corrected Version.

10. “Instructions for Recording and Reporting Occupational Radiation Exposure Data,” Regulatory Guide 8.7, Revision 2, U.S. Nuclear Regulatory Commission, 2005.

11. “Criteria for Monitoring and Methods for Summation of Internal and External Occupational Doses,” Regulatory Guide 8.34, U.S. Nuclear Regulatory Commission, 1992. Still current as of 2011.

10

Environmental Monitoring Programs and Equipment

Outline of this Chapter

Chapter Summary

Environmental monitoring is the last chapter in the Instrumentation Unit. It covers principles of program design, the choice of appropriate instruments and then discusses some current issues.

Monitoring programs allow for the calculation of population doses. A pathway analysis is used to relate environmental sample concentrations to human doses. A preoperational monitoring program is characterized by comprehensiveness. Critical nuclides and pathways must be identified, along with collection of meteorology data. In the postoperational phase, the key principles of sensitivity and selectivity are used to extract the desired data out of a background of interfering activities.

External gamma ray levels are usually measured with a TL dosimeter. To get high sensitivity, non tissue-equivalent phosphors are used with energy compensating filters. Surface fallout activity is usually measured with gummed-film. Airborne particles can also be sampled using filtration. Dust loading can be avoided with a moving filter sampler.

Radioactive gases are often sampled with a continuous flow instrument. In some cases, a grab sample is used. Adsorption is useful for radioiodine sampling and for radon monitoring at low concentrations.

The dose to a worker exposed to airborne radionuclides can be calculated quite simply. By comparing the air concentration to the DAC, the dose is easily found from the accumulated DAC-hours of exposure.

Radon gas in buildings continues to be something of a public health controversy. Charcoal canisters and alpha track detectors are sensitive and accurate at environmental concentrations. Long-term studies underway will hopefully allow better estimation of risks of radon exposure at low concentrations. Such studies are difficult because most lung cancers are caused by smoking, not radon.

Dose reconstruction is a growing sub-field within environmental health physics. This involves new modeling to recalculate doses to individuals or populations with more realism than past worst case calculations.

Now that we are in the post cold-war era, public demands are resulting in more and more efforts to restore and reclaim contaminated sites. The Marshall Islands and the former Soviet Union are two areas where such efforts are underway. Success in these types of projects will pave the way for similar efforts at other sites.

Monitoring Programs

Introduction

There are a number of reasons why a facility establishes a program to measure the radiation levels in the vicinity of their site. These are briefly summarized in Figure 1. It would be impractical to issue monthly radiation badges to the millions of persons who live in the vicinity of nuclear facilities. In addition, the likelihood of obtaining any useful data would be minimal since ordinary badges are not sensitive enough to detect changes in environmental radiation levels. At the same time, to be consistent with the ALARA philosophy, we need to be aware of changes in population radiation exposure due to nuclear facilities. Thus, the only practical way to estimate population exposure is by careful measurement of both the external radiation level in a region and the radioactivity present in air, food and water. When these measured values are combined with a knowledge of the drinking water sources (e.g., private well vs. city water) and types of food consumed in the region by the population, the estimated population dose can be computed quite accurately using internal dosimetry models.

A rapidly evolving sub-field of environmental radiation protection is dose-reconstruction. This is the speciality of a growing number of scientists, researchers and technicians who are attempting to calculate the radiation dose to populations exposed to radionuclide releases occurring several years to several decades in the past. The original dose estimates were usually based on the maximally exposed individual. Much of the driving force behind the new efforts has been pressure brought to bear on state and national legislatures by consumer interest groups representing exposed populations. In the U.S., the Freedom of Information law passed some time ago gave added impetus since many formerly classified records of weapons production facilities are now available to the public. Obviously, the key to successful calculation of these population doses in the past is accurate and complete environmental sampling results. Thus, future generations are dependent on current sampling programs if it becomes necessary to perform such “back calculations” in times to come.

1) Measure human population doses
2) Allow for future reconstruction of population dose
3) Determine radiological impact of a facility
4) Detect an unknown release
5) Quantify accidental off-site radionuclide releases
6) Meet specific license requirements
7) Foster a credible public image
8) Obtain pathway data to refine models
9) Test the adequacy of on-site control measures
10) Study air and water mixing patterns

Fig. 1 - Purposes of environmental monitoring

Dose reconstruction efforts near the Hanford facility in Washington state and the Nevada Test Site were briefly mentioned in Chapter 6. Further discussion of other such projects will be found near the end of this chapter.

A question related to population exposure is the determination of the radiological impact of a nuclear facility on the existing environmental levels. This involves measurements similar to those described above. Now, however, extensive environmental samples must be collected both before and after the new facility has begun operations with radioactive materials. The difference in the calculated population doses between the before and after cases is then called the radiological impact of the new facility.

The fourth goal in Figure 1 is the detection of an unknown release of radioactivity from a facility. Clearly, if an adequate radiation safety program is being maintained at a site, there would never be an "unknown" release because appropriate alarming monitors would warn of the condition as it occurred. Unfortunately, this is not always the case. Lack of funding, improper training, Murphy's Law and a host of other conditions make the ideal radiation safety program hard to achieve in practice. Thus, the environmental monitoring program functions, to an extent, as a "backup system" in that it would detect unusual levels in the event of a release of activity that was not sensed by some other early warning technique.

Historically, this was the case in at least one major radiation accident. The fact that the British Windscale reactor core was on fire in 1957 was discovered only after a routine air sample collected downwind from the reactor site showed huge levels of Iodine-131. More recently, the first indication that western nations had of the problem at Chernobyl was a result of dramatically elevated levels on Scandinavian routine air samplers. These incidents will be discussed in detail in Chapter 14.

A monitoring program is maintained because the law requires it. The radioactive materials license issued to a facility spells out the extent of the required measurements. To illustrate, Figure 2 shows the diversity of monitoring stations positioned on the DOE facilities at the Idaho National Engineering Laboratory. In many licenses, environmental monitoring plays only a very limited role. The activities or nuclides involved or the form in which they are handled are such as to pose little risk of contaminating the environment. The public health and safety can be protected with little effort. On the other extreme, facilities such as radioactive waste burial grounds, DOE weapons production facilities, nuclear power stations and fuel reprocessing plants are required to maintain continuous extensive monitoring programs to protect the public. For most of these large facilities, specific guidelines are spelled out in the appropriate state and federal regulations, in regulatory guides or in DOE Orders. A sample program for a nuclear power station will be examined later in this chapter.

Another goal of environmental monitoring is to increase the credibility of the image of a facility in the eyes of the public. Many facilities have adopted the approach that they will routinely conduct a more extensive environmental program than the minimum specified in the law. This demonstrates that they are willing to do more than their share to protect the public. Also, the extra monitoring data is frequently useful in cases of lawsuits or media coverage alleging off-site radiation damage.

The eighth goal listed, obtaining pathway data, is important in the continuing objective of improving environmental monitoring programs. It was mentioned above that there are two basic steps in arriving at a population dose estimate – physical

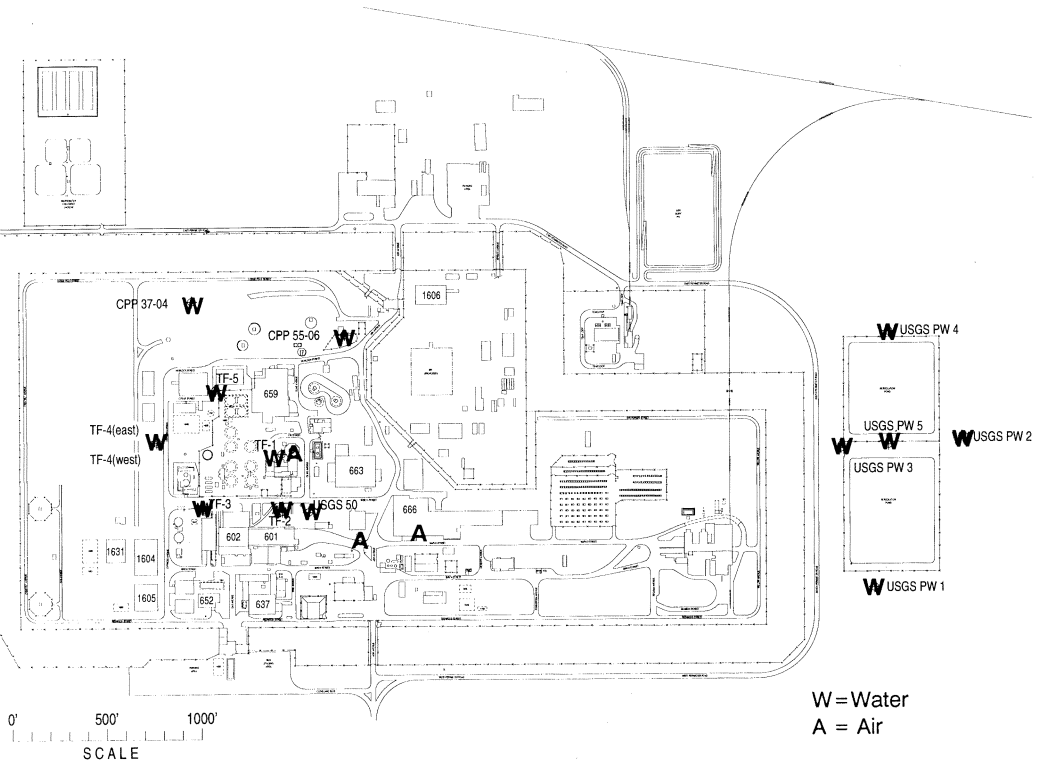


Fig. 2 - Environmental monitoring stations near the Idaho National Engineering Laboratory

measurements followed by mathematical calculations. This goal is closely tied to the second step.

A "pathway" is any route that radioactivity can follow in passing from a licensed facility to a person in the general population where it becomes internally deposited or contributes external dose (See Figure 3). Since most pathways have several steps involved with different possible branches at each of the "crossroads," there

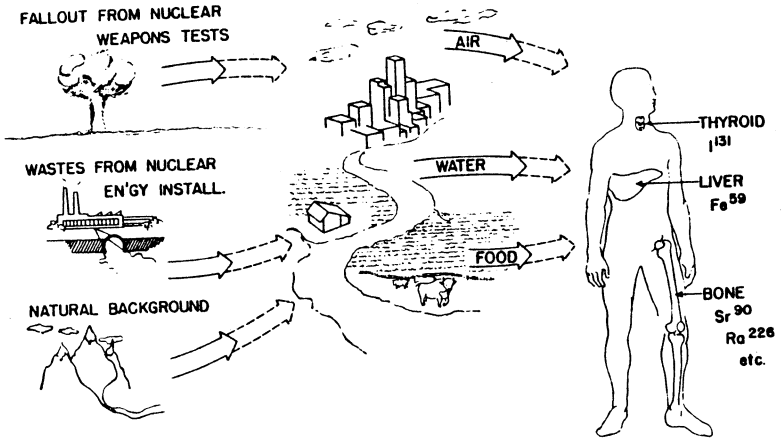


Fig. 3 - Radionuclide pathways from source to population

are literally hundreds of different possible pathways for most facilities. In order to arrive at realistic population dose estimates, it is necessary that the computer be informed as to the fraction of the activity that proceeds by each new branch at the intersections. Theoretically, a lot of information has been developed on environmental transport mechanisms, i.e., how the various radionuclides move through soil, waterways, the atmosphere, plants and animals. Actual environmental monitoring results can then be used to verify (or reject, as the case may be) the predictions of the theory. Where the theory breaks down, the actual measurements can be used to point the way toward improved theories.

The next goal is the test of control measures. Radioactive materials must always be used under "controlled" conditions, i.e., sealed sources are confined to specified shielded containers and unsealed sources are used with protective equipment such as glove boxes, chemical fume hoods, remote pipettors, absorbent paper, special waste receptacles, etc. These control procedures are designed to prevent the release into the environment of concentrations of radioactivity above specified legal limits and to maintain doses to workers "as low as reasonably achievable," ALARA. If certain radionuclides begin showing up in environmental samples, this indicates that the radiation protection staff must determine where the present control measures are lacking. As new measures are instituted, progress can be followed by watching for those nuclides in future environmental samples.

The final goal is to study the movement of large air masses or currents in bodies of water. Knowledge of these movements is crucial in performing the population dose estimate calculations. From time to time, many facilities release small amounts of radioisotopes into the environment under controlled conditions. These known activities can be used as tracers for air and water masses. Environmental samples taken after the release can then be used to demonstrate the flow patterns.

A few other general comments are in order. To set up a successful environmental monitoring program for a given radionuclide, some conditions must be met. The isotope must be present in the environment in a concentration (e.g., becquerels per liter) that is higher than the minimum detectable concentration, MDC, for the counting system used to process the sample. This counter performance index of sensitivity will be discussed further in Chapter 12. Secondly, the radioisotope must be capable of being efficiently transported through some environmental pathway. For example, this might be by groundwater, or via air currents to a pasture where it enters the cow-milk-man chain. Finally, the radioisotope sought must have a sufficiently long physical half-life. This will allow time for environmental transport, sample collection, sample processing and counting. If all of these conditions are met, a successful program is possible.

Preoperational And Postoperational Programs

A preoperational monitoring program is one which is put in place prior to the use of radioactivity at the site. It is governed by the overriding principle of COMPREHENSIVENESS. In the earliest stages of developing a program, there are thousands of nuclides and pathways which will eventually be shown to be of no significance, but

Locate radiation anomalies
Document ambient levels
Identify critical nuclides and pathways
Document seasonal meteorology patterns

Fig. 4 - Preoperational program objectives

work is needed to identify them. There are several distinct objectives which must be accomplished before nuclear operations begin. These are summarized in Figure 4.

The first objective involves a grid survey of radiation levels over the complete site geography. A crosshatch pattern of imaginary lines is laid out over the facility and measurements made at the intersecting points on the "grid." Particular attention is paid to the uniformity of the readings. For most sites, the land area is small enough so that the underlying rock strata are identical at all locations. Occasionally, however, there is a significant enough variation that the background radiation will be anomalous at some location, i.e., it will read higher or lower than the adjacent area. It is important to be aware of these "normal" variations in the background radiation so that the new nuclear facility is not blamed for the unusually high level at some location. Once any anomalies are located, then the various contributions to the natural radiation environment at the site are documented extensively. This information will be used as a reference condition against which environmental samples collected after nuclear operations are begun are compared.

The objective concerning critical pathways and critical nuclides is related to the discussion of pathway analysis above. A critical pathway is the route taken, from point of release to body entry, of a critical radionuclide which causes human exposure. The expression "critical" refers to those nuclei which cause the largest dose contribution to the actual population at risk near the facility, and the particular pathways they follow. In a realistic case, there may be close to a hundred possible radionuclides that could potentially be released by a facility. Each nuclide might have a hundred different possible pathways. One of the time-consuming tasks of the preoperational monitoring program is to develop the necessary mathematical models for calculating the population dose estimates for each nuclide and pathway. When that project is completed, the critical nuclides and pathways become those which are at the "top of the list" when all combinations are ranked in decreasing order of population dose. Some common radioisotopes which often qualify as being critical nuclides are strontium-89, strontium-90, cesium-137, iodine-131, radium-226, hydrogen-3, manganese-54 and barium/lanthanum-140.

Finally, the preoperational monitoring program must collect enough data to fully document the average meteorological patterns and hydrological patterns in the vicinity of the site. This information is used for input data in the mathematical models for estimating population doses. It is necessary to know average wind speed and direction and water flow patterns as these determine where and how fast the released radioactivity is transported. The same information on average conditions is also needed in emergency planning for the facility. This allows realistic dose estimates to be made and helps determine evacuation routes.

A wind rose is a graphical representation of the wind speed and direction. A typical wind rose might represent data for a one-month period at the observation point. The basic data is obtained from a wind speed and direction recorder or may be generated manually by reading speed and direction meters for a one-minute period each hour and tabulating the results for the one-month interval. Figure 5 is a sample result. The wind direction is divided into 16 sectors. The lengths of the projecting bands show the fraction of the time that the wind blows from that direction and the width of the band shows the wind speed. The key below the rose shows the scale values.

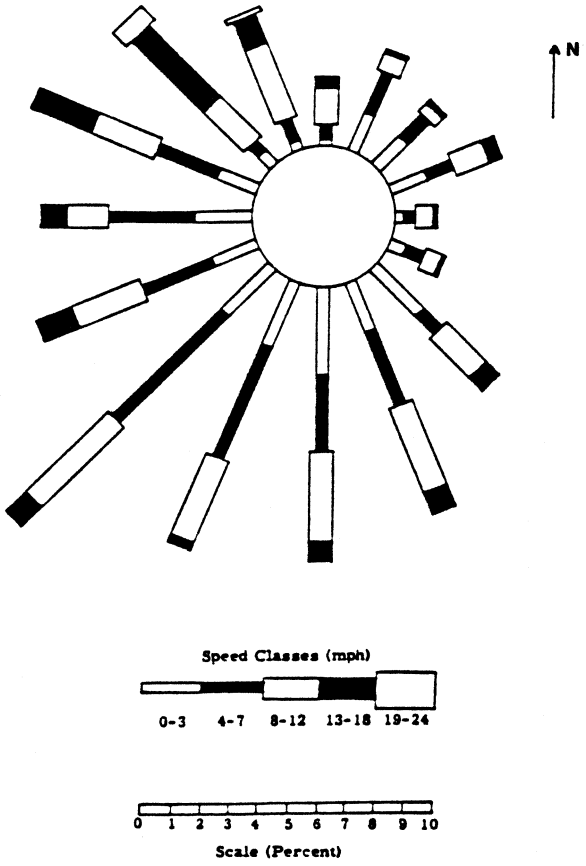


Fig. 5 - An example of a wind rose

The atmospheric stability is measured, in meteorology, by the change in temperature with altitude above the measuring station. It is usually expressed as a "stability class." Meteorologists recognize six classes. The class determines the behavior of a "plume" of radioactivity emitted from a ventilation stack and is used quantitatively when calculating the concentration of airborne activity at some downwind point following a release. Class A is extremely unstable. It is characteristic of clear, daytime conditions with low wind speed. Class B is moderately unstable, often found in the daytime under partly cloudy conditions. Class C is slightly unstable, such as an overcast day with moderate wind speed. Class D is neutral. It occurs most often during the nighttime. Class E is slightly

stable and is representative of overcast nights at low wind speed. Finally, class F (temperature inversion) is moderately stable. It is found only on clear nights with very low wind speed. Plume dispersion is determined by the stability class in effect at the time. Example patterns are shown for several cases in Figure 6.

Looping would indicate unstable air – Classes A through C. Fanning occurs under inversion conditions, Class F. Little dilution takes place. Lofting is observed near sunset and fumigation is likely as the morning sun dissipates nighttime inversion conditions.

A general rule of thumb is that a preoperational monitoring program is conducted for at least one year before start-up of a significant nuclear facility. This allows time for weather observations over the four seasons. In the case of a nuclear power station, federal guidelines require that some aspects of the preoperational program must be functioning for two years before arrival of fuel to the site. When the preoperational program is completed, it should be possible to answer the four important questions listed in Figure 7.

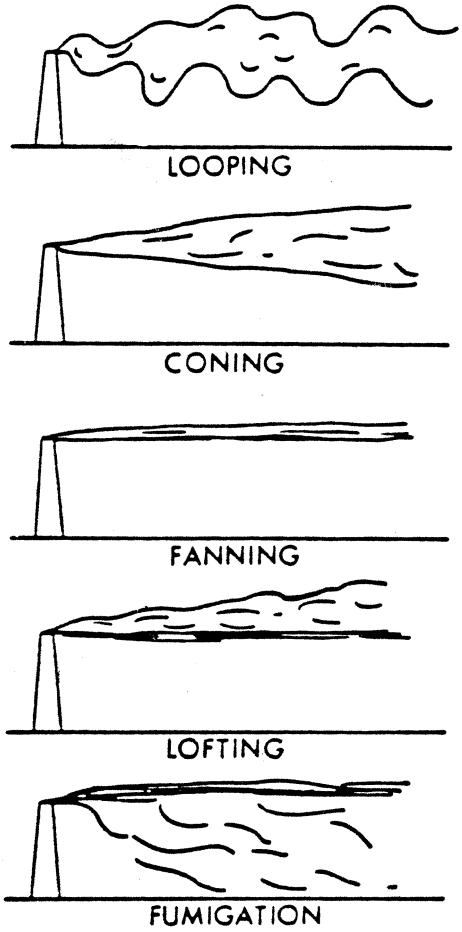


Fig. 6 - Sample plume dispersion patterns

What radioisotopes should be measured?

Where should the samplers be located?

How often should a sample be collected?

Which equation is used to calculate population dose?

Fig. 7 - Preoperational program questions to be answered

The normal postoperational monitoring program is governed by the two principles of SENSITIVITY and SELECTIVITY. Ideally, we would like to be able to detect the presence of a radionuclide in the environment at a sensitivity corresponding to a concentration of 1% to 10% of the applicable limits set forth in the federal or state radiation protection standards. In practice, this is often difficult to achieve. Some radionuclides emit radiations which are hard to detect, even with modern equipment.

Another common reason why the detection limit may not be as low as desired is the presence of interfering natural radioactivity in the environment. It may be necessary to select the radionuclide of interest out of a background of competing activities. Radon gas is a good illustration of this problem. The isotope Rn-222 decays, as part of a chain of radioactive daughters, with a succession of alpha and beta particles being released. The complete chain is illustrated in Figure 8. This natural alpha environment interferes with radiation protection measurements that are looking for artificially introduced alpha emitters. The problem, in theory, could be solved by making a

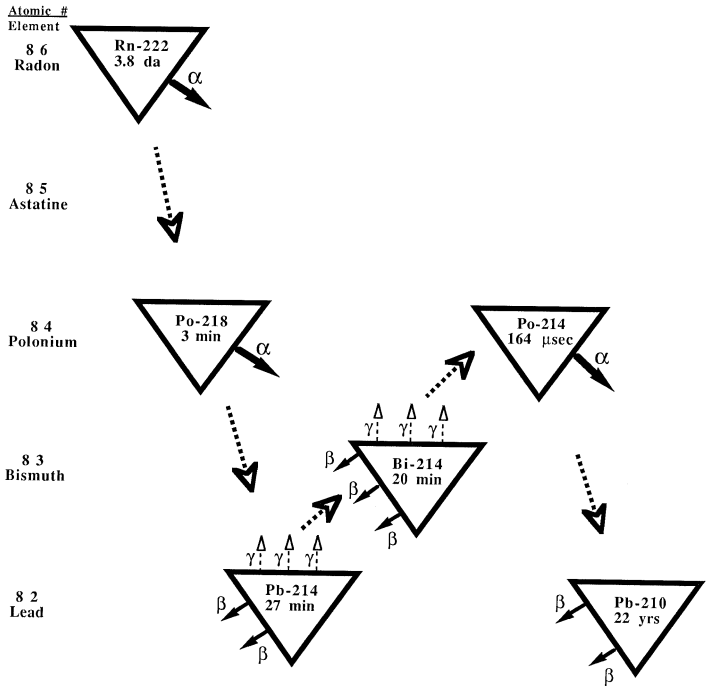


Fig. 8 - The radon-222 decay chain

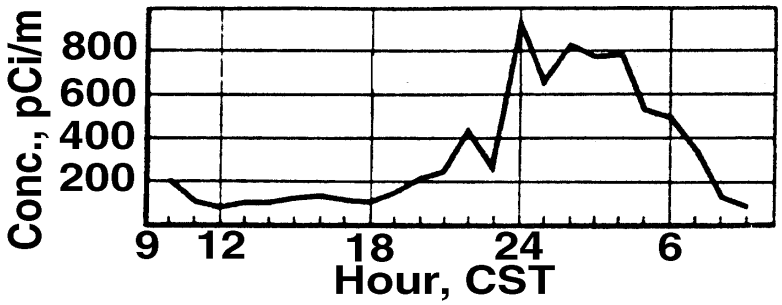


Fig. 9 - Typical daily variation in radon air concentration

background correction – subtracting the radon alpha level from the instrument reading. In practice, this is not feasible for the reason shown in Figure 9.

This graph shows a very typical behavior for the radon air concentration. The concentration can fluctuate by several hundred to a thousand percent in a matter of an hour or two. This is caused by the fact that radon emanation through rocks and soil to reach the atmosphere is strongly dependent on barometric pressure. (This fact is used in most hardrock uranium mines to reduce miner's lung doses from the radon and daughter products in the mine atmosphere.) Since the "background" concentration is not constant, a simple background subtraction is meaningless.

Two practical techniques have been developed to get around the problem of radon interference in environmental alpha monitors. The radon and its daughters have quite short half-lives. Thus, merely introducing a delay time between sample collection and counting will reduce the interference. By employing detectors such as surface barrier diodes (Chapter 7) which have excellent energy resolution, it is possible in many applications to eliminate or reduce radon interference. A single channel pulse height analyzer is set on the energy of the alpha emitter being sampled. Radon monitors, themselves, will be discussed later in this chapter.

The principle of SELECTIVITY means that we would like to be able to separate each desired nuclide out from the background interference cleanly enough to be able to measure the activity in a sample. For photon emitters, this is now practically achievable as a result of the common availability of semiconductor counters such as the HPGe counters discussed in Chapter 7. The photon energy resolution is so good that individual photon peaks can be seen for virtually all known gamma emitters. The counter is usually connected to a multichannel pulse height analyzer which is itself interconnected to a small dedicated computer. The computer is programmed with gamma ray energy spectra for all known gamma sources. It then analyzes spectra run on environmental samples, compares the located peaks with its library of spectra, and prints out a report of the nuclides and activities in the sample. Figure 10 shows an example of a spectroscopy report produced with the Canberra Series 90 dedicated multichannel analyzer.

By determining the critical nuclides and pathways in the preoperational monitoring program, the postoperational program is usually of much smaller scope. In the

```
Tag no.= 11000065   CANBERRA      Series 90              Page  1
Live =    4000 s    True =     4000.6 s  Readout: 12:01 23-Apr-88

User 1  Group 1  Unit 1.1     Collected at: 10:00:13.9 23-Apr-88
Range, Gain=  4096, 4096  Offset=     0  LLD, ULD=   2.0, 110.0 %

X-cal = + 3.8327-08 * Ch^2 + 4.5653-01 * Ch + 1.0846+01 keV

PEAK: Statistics = 2.00     Min width =   6      Max width =  15
AREA: Background =    2     % Error =  1.65

Iso ID: Window=  2.0 keV   Library: RESIDENT
Age (days)=          0.0

Equ #  1    Crossover=  279.2 keV
Lo-eff= e^(- 2.1017+00 *ln(E)^2 + 2.0708+01 *ln(E) - 5.6262+01 )
Hi-eff= e^(+ 9.1495-02 *ln(E)^2 - 1.4256+00 *ln(E) - 1.3508+00 )


ROI#     Start/      Integral/     ICPS/       Peak/
          Stop          Area       %Err        FWHM
           keV                                  keV

  1      11.76          14555       3.6       16.79
         21.35           2741     14.23       0.470


  2      85.72          24955       6.2       88.02
         91.20           8435      4.18       1.415

            Cd-109   AT    88.00keV  = 4.3835-01   uCi
            Np-237   AT    86.50keV  = 1.3143-01   uCi

  3     119.96          32400       8.1      122.19
        124.53          17786      1.93       1.420

            Co-57    AT   122.10keV  = 2.7701-02   uCi

  4     135.03          13122       3.3      136.43
        138.68           2180      9.53       1.406

            Co-57    AT   136.50keV  = 2.6587-02   uCi

  5     163.79          21808       5.5      166.04
        169.27           8710      3.70       1.507

            Ba-139   AT   165.90keV  = 6.4338-02   uCi
            Ce-139   AT   165.90keV  = 1.5120-02   uCi

  6     236.38           8977       2.2      238.69
        240.95           1184     16.97       1.374
```

Fig. 10 - A sample spectroscopy report

Courtesy, Canberra Industries Inc.

postoperational phase, attention is focused almost entirely on the critical nuclides and instruments are located on the critical pathways. Fission products such as Sr, Cs and iodine follow the pasture-cow-milk-man pathway or the water-fish-man pathway. Food sampling on milk and fish would be appropriate. The radium isotopes are often found in drinking water so it becomes the sampling medium. The critical pathway for carbon-14 is usually through food or the nuclide is inhaled as C-14 dioxide.

Monitoring Program Examples

The basic decisions which must be made in setting up a program to monitor the environmental levels of radioactivity are listed in Figure 11. Once the critical

Which isotopes?	**Collection frequency?**
Sampling location?	**Sample preparation?**
Sampling medium?	**Counting equipment?**
Sample size?	**Calculational model?**

Fig. 11 - Postoperational environmental monitoring program decisions

nuclides and pathways have been established, the type of medium (air, water, food) is chosen. Procedures should then be developed to specify the manner in which the sample is taken, its size, and the times that samples are to be collected. Next, the analysis laboratory procedures should be specified. Sample preparation might involve evaporation of liquid samples or possibly a radiochemical separation to remove interfering activities. In some cases, all that is required is a delay between sample collection and counting to allow short-lived nuclides to decay. Choice of counting equipment is governed by the volume of samples to be processed, the types of radiations emitted by the sample and the desired lower limit of detection (the MDC) for the critical nuclide, discussed below.

To illustrate the various principles discussed up to this point, a sample environmental monitoring program appropriate to a nuclear power reactor site will be covered in some detail. The information given is based on Regulatory Guide 4.8 issued by the Nuclear Regulatory Commission. The preoperational phase is begun two years before start-up. This allows sufficient time to evaluate the natural radiation environment at the site, to purchase and evaluate suitable sampling and counting equipment, establish analysis lab procedures and train personnel in all aspects of the program. In the first year, ambient gamma levels, concentrations of radioactivity in food and in shoreline sediments are made. In the second year, airborne radioactive particulates, and concentrations in milk and water are added to the program. During the final 6 months, radioiodine levels are monitored in air and milk.

Some of the requirements for the postoperational phase of the program at a nuclear power station are shown in Figure 12. The program must continue at this level of measurement for the first three years of plant operation. After that time, changes in scope, based on the 3 years of data, can be implemented following permission from the NRC.

A few comments on Figure 12 entries are in order. Airborne samples are collected "in the least prevalent wind direction" to provide a "site-specific control sample" which is subtracted from downwind sample counts. In the event some other facility releases radioactivity, this will be shown by these upwind readings. Water sampling "upstream from discharge point" serves an identical purpose. "DIRECT Exposure Pathway" refers to the external radiation levels at the site. Acceptable instruments will be discussed below. The term "X/Q" used in the second column is the airborne concentration of radioactivity per unit activity release rate. Regulatory Guide 1.109 contains the equations needed to calculate population doses near a nuclear power reactor site based on the environmental concentrations.

Exposure Pathway	Number of Samples and Locations	Sampling and Collection Frequency
AIRBORNE Particulates	3 samples from locations in different sectors of highest offsite ground level concentrations. 1 sample from the residence having highest $\chi/Q + 1$ from each community within 10 mile radius. 2 samples from control locations in least prevalent wind direction	Continuous sampler operation with sample collection weekly or as required by dust loading.
AIRBORNE Radioiodine	2 samples from locations in different sectors of highest offsite ground level concentrations.1 sample from the residence having highest $\chi/Q + 1$ from a community within 10 mile radius. 1 sample from control location in least prevalent wind direction	Continuous sampler operation with canister collection weekly. [Canisters subject to channeling. Check carefully before operation or mount several in series to prevent iodine loss]
DIRECT	2 or more dosimeters to be placed at the same locations as for air particulates, as well as 2 additional control locations. 2 or more dosimeters to be placed at each of 3 other locations of highest offsite ground level dose	Quarterly
WATER-BORNE Surface	1 sample upstream 1 sample in immediate area of discharge	Composite sample
WATER-BORNE Drinking	1 sample of each of 1 to 3 supplies obtained within 10 miles which could be affected by plant discharge	Composite sample [Should be collected with equipment capable of collecting an aliquot at time intervals short relative to compositing period]
INGES-TION Milk	1 sample at offsite dairy farm or milk animal at location having highest χ/Q. 1 sample from milking animals in each of 3 areas where doses are calculated > 1 mrem/year 1 sample at control location in least prevalent wind direction	Weekly or semimonthly depending on calculated dose. [Weekly if dose to child's thyroid >15 mrem/yr. Semimonthly if less than 15 mrem/yr]
INGES-TION Fish	1 sample each commercially & recreationally important species in vicinity of discharge point. 1 sample of same species in areas not influenced by plant discharge.	Semiannually or in season

(Excerpted from U.S. Nuclear Regulatory Commission Regulatory Guide 4.8)

Fig. 12 - Excerpts from a nuclear power plant post operational program

As mentioned earlier, the principle of SENSITIVITY is very important. In the case of a nuclear power plant, the NRC has established some sensitivity criteria for 11 particular radionuclides that are likely to be among the critical nuclides. The table in Figure 13 lists the nuclides along with the required Minimum Detectable Concentration (MDC) for each in several different sampling media. The meaning of the MDC and the equations for calculating it will be covered in Chapter 12.

The concentrations specified in Figure 13 were chosen on the basis that they would deliver 4 mrem per year to the population at risk via airborne routes or 2.5 mrem per year via waterborne routes. For each nuclide, the calculations were done for the most sensitive human organ and age group.

Finally, for the nuclear power plant, the NRC regulations require an annual report of the measurement results. The data are subjected to a statistical analysis. Comparisons must be made of currently measured values to those measured during the preoperational phase. If the environmental concentration of a particular nuclide, averaged over three months, exceeds 4 times the MDC, then a report must be filed to the NRC within 30 days. The value of 4 X MDC is given the name "Reporting Level" when used in this way.

Environmental Instruments

External Gamma Radiation

Environmental levels of external gamma radiation ("direct radiation") are most often measured using TL dosimeters. As discussed in Chapter 8, several types of thermoluminescent dosimeter phosphors are unusually sensitive to radiation. Calcium fluoride and calcium sulphate dosimeters are available commercially and are able to measure doses below 1 mrem per month. By encapsulating the individual dosimeter phosphors in evacuated glass bulbs, very reproducible results can be obtained in the sub-mrem range.

A thermoluminescent dosimeter for environmental monitoring is available from Landauer., Inc. It makes use of a carbon-doped aluminum oxide (Al_2O_3:C) phosphor which has an effective atomic number of 10.2, much closer to 7.5 than the calcium phosphors. It has a maximum fading of 10% in three months at extreme environmental temperatures and no fading at room temperature for up to nine months. The minimum detectable dose is 1 microsievert (0.1 mrem), with dose being reported to the nearest tenth of a millirem. The phosphor is similar to the one used in the Landauer Luxel® personnel dosimetry optically stimulated luminescence badge.

Recall that many TL phosphors show significant energy dependence. The energy compensating shields covered in Chapter 8 can be used effectively in these cases. Corrections may also need to be made for fading of dosimeters. Often, an external gamma ray dosimeter will be left in place in a remote monitoring location for a three-month period before collection and processing. Careful choice of a TL phosphor will result in a dosimeter with the necessary sensitivity and acceptable fading characteristics.

REQUIRED DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS

Analysis	Minimum Detectable Concentration (MDC)*					
	Water (pCi/l)	Airborne Particulate or Gas (pCi/l)	Fish, Meat, or Poultry (pCi/kg, wet)	Milk (pCi/l)	Vegetation (pCi/kg, wet)	Soil (pCi/kg, dry)
gross beta	2	1×10^{-2}				
H-3	330					
Mn-54	15		130			
Fe-59	30		260			
Co-58, Co-60	15		130			
Zn-65	30		260			
Sr-89	10	5×10^{-3}	40	10		
Sr-90	2	1×10^{-3}	8	2		150
Zr-95 Nb-95	10					
I-131	0.4	7×10^{-2}		0.4	80	
Cs-134, Cs-137	15	1×10^{-2}	130	15	80	150
Ba-140, La-140	15			15		

*The nominal Minimum Detectable Concentration is defined at the 95% confidence level. The MDC levels are decay corrected to the end of the total sampling period. These MDC levels should be used as minimum criteria for objectives for instrumentation and analytical procedure selection.

Fig. 13 - Some mandated MDCs for critical nuclides

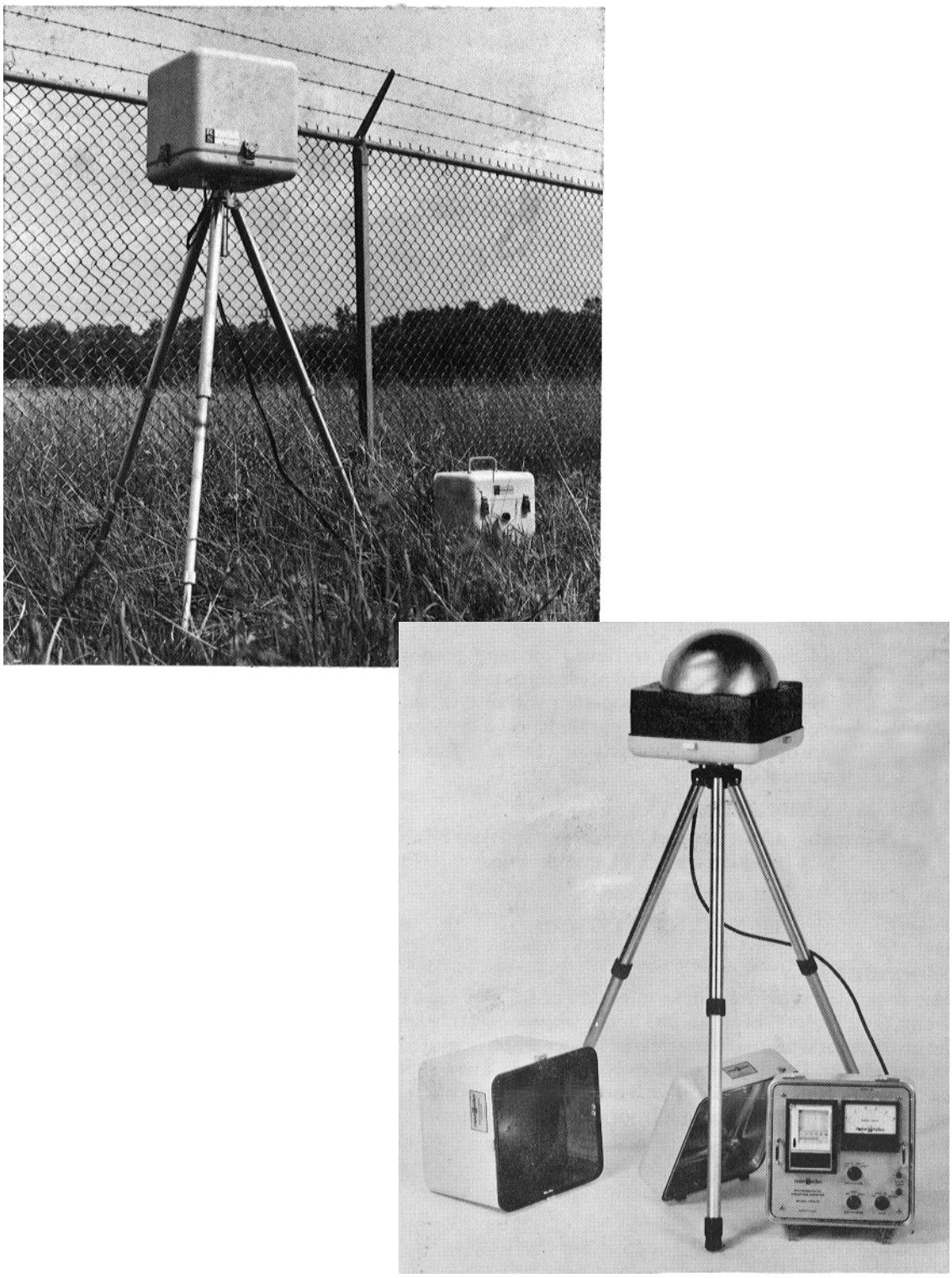
Occasionally, it is helpful to have a "real time" record of direct gamma ray levels at a monitoring station site. For instance, this would be valuable in the event of an accidental release of radioactivity. The TL dosimeters just discussed are of the integrating type. This means that they accumulate radiation dose as it occurs but can only provide information on the cumulative total dose when processed. No indication is given as to when the dose was received by the badge. A series of small releases could not be distinguished from a single large release with a TL dosimeter. This problem can be solved by use of a pressurized ion chamber. Figure 14 shows such an instrument. The spherical detector is operated in the ion chamber region of the characteristic curve (Chapter 7) but the filling gas is pressurized to 40 atmospheres (about 600 psi) with argon gas. This "extra" loading of gas molecules makes the chamber more sensitive – the chamber has the sensitivity of one which has 40 times the volume of the sphere operated at ambient pressure. The manufacturer claims that the instrument pictured can detect a CHANGE in the gamma ray background of as little as 1 mrem per year. The real time data for dose rate is displayed by use of a chart recorder with a 30 day supply of paper. The entire instrument is portable to the extent that it can be operated at remote locations from a battery pack. In the event that an accidental release of radioactivity occurs, the dose rate information is recorded as the plume of activity passes over the pressurized ion chamber. These dose rate changes give valuable information that can be used to predict exposures to the population in the path of the plume.

A number of nuclear power plants have carried this one step further. They have connected, via radio telemetry or hard wiring, a widely dispersed series of these pressurized ion chambers to read out dose rates into the emergency control center. This eliminates the need to retrieve the recordings under accident conditions and gives instantaneous information to emergency planners.

Surface Deposited Activity

Another category which is monitored is the level of radioactive particulates that settle out onto the ground. Of course, such "fallout" is common to any release of radioactive particulates and does not necessarily imply that a nuclear detonation has occurred. The radioactivity that reaches the ground could be due to planned, legal releases from a nuclear licensee. It could also be due to accidental releases from a facility or result from distant atmospheric testing of nuclear devices. As of 2011, there have been no atmospheric tests of nuclear weapons for over a decade. Surface deposited activity can be correlated with its origin if the sample is analyzed as to the radionuclides present. An example of how fallout from continued atmospheric testing might affect routine environmental sampling results is shown in Figure 15. The graph plots measured tritium concentration at a U.S. monitoring station over a 2.5 year period.

Several instruments are used in collecting samples for surface deposited activity measurements. The "old flypaper technique," though simple, is still in common use. Flypaper, also called "gummed-film" consists of a sheet of waterproof paper or



Courtesy, Reuter-Stokes

Fig. 14 - A commercial pressurized ion chamber instrument

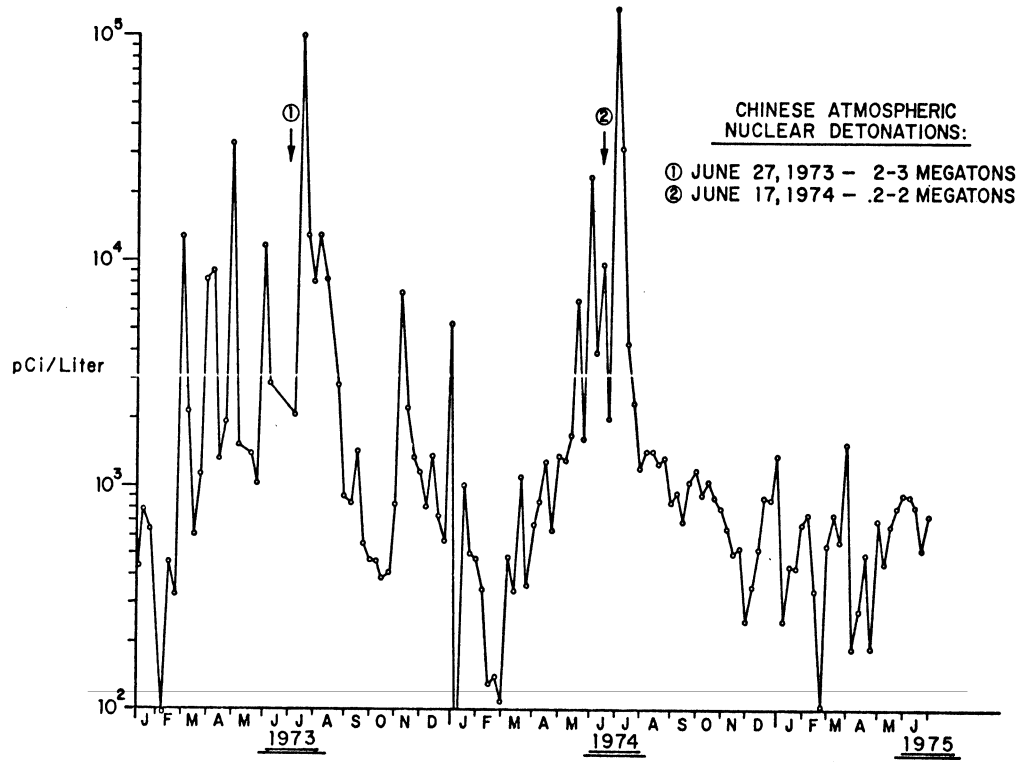


Fig. 15 - Influence of fallout on the background level

acetate which has been coated with a waterproof adhesive layer on one side. A typical sheet is a 12 inch square. It is attached to a flat-plate holder located typically on the roof of a building. The sticky side is pointing up. At the end of the sampling period, (for example, monthly) the paper is removed from the holder and folded in half to trap the deposited particulates. It is then placed in an envelope and mailed to the analysis laboratory. There, the flypaper is "ashed" (burned at high temperature). This gets rid of the paper backing and adhesive. The ash that remains is then counted for radioactivity. Alpha and beta analyses are usually performed with a gas flow proportional counter. The gamma ray energy spectrum is determined using a germanium semiconductor counter. Under a variety of weather conditions, the gummed-film traps an average of 10% to 40% of the particles that fall on it. Figure 16 shows the gummed-film as used a few decades ago. Figure 17 shows flypaper monitoring sites during atmospheric testing in the 1950s in the USA.

Rainwater is another medium from which to collect fallout samples. A relatively large diameter funnel passes collected water through an ion exchange column. The column is nothing more than a long cylindrical tube packed with ion exchange resin beads. The beads have the property that they will attract and attach to their surface various ions carrying a positive or negative charge (cation and anion resins, respectively). In the present case, both columns are used, back to back, to extract all of the fallout ions as the rainwater flows through. Next, the ion exchange resin

Fig. 16 - Gummed-film monitor in use (1950s)

beads are removed from the column and the radioactivity attached to the beads is measured using conventional counting techniques.

Occasionally, soil samples are taken for the purpose of assessing the fallout activity. There are several problems with using soil as a collection medium. The bulk of the soil cannot be easily removed to release the radioactivity. It thus acts as a radiation shield if counting is attempted. The obvious answer of washing out the radioactivity also turns out not to be so simple. Most soils contain a clay component. Clay is a natural "ion exchange resin" in that it attracts and attaches to its surfaces stray ions in the vicinity. Thus, that material will not easily wash out for counting purposes.

Finally, grass turns out to be an excellent trap for fresh fallout activity. In general, broad leaf vegetation tends to accumulate such particulates. It can be readily collected. Little self-shielding is present for gamma emitters so gamma activity can be counted directly with a germanium counter. Many samples can be processed in a short time which is helpful under accident conditions when grid surveys are performed to find the extent of the spread of the contamination.

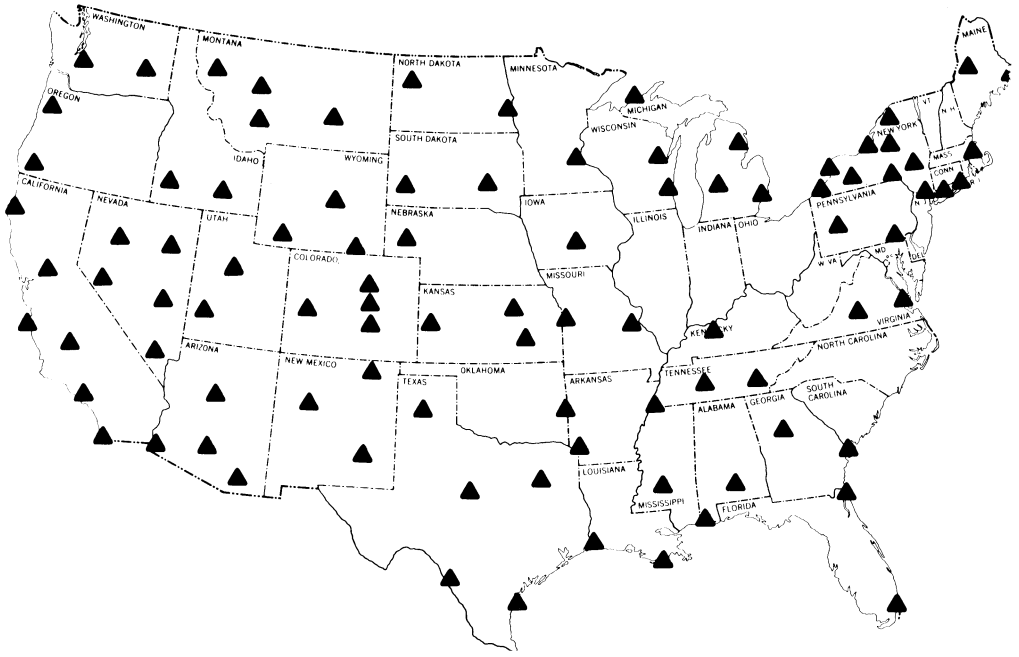


Fig. 17 - Gummed-film monitoring stations in the 1950s

Air Sampling - Particulates

Airborne radioactivity has two general classes of contributors – the particulates and the gases. The instrumentation needed for each is quite different, so they will be discussed as separate cases. Then, some special air sampling cases will be addressed. Separate sections will cover radon gas in buildings, sampling for radioiodine and tritium air sampling.

Figure 18 is a list of some techniques used for monitoring airborne particulates. The first technique, sedimentation, merely means allowing the force due to gravity to act on the mass of the suspended particles. They are then collected as they settle out. The flypaper discussed earlier is an example of the use of this technique.

Since particulates possess a mass, they will then have inertia. If they are moving, they will tend to travel in a straight line. Sampling devices using inertia to separate particulates from an air stream force the air stream to make a sharp turn in

Technique	Example
Sedimentation	Flypaper
Inertial Separation	Cascade Impactor
Filtration	High Volume Sampler
Electrostasis	Precipitator

Fig. 18 - Some particulate air sampling techniques

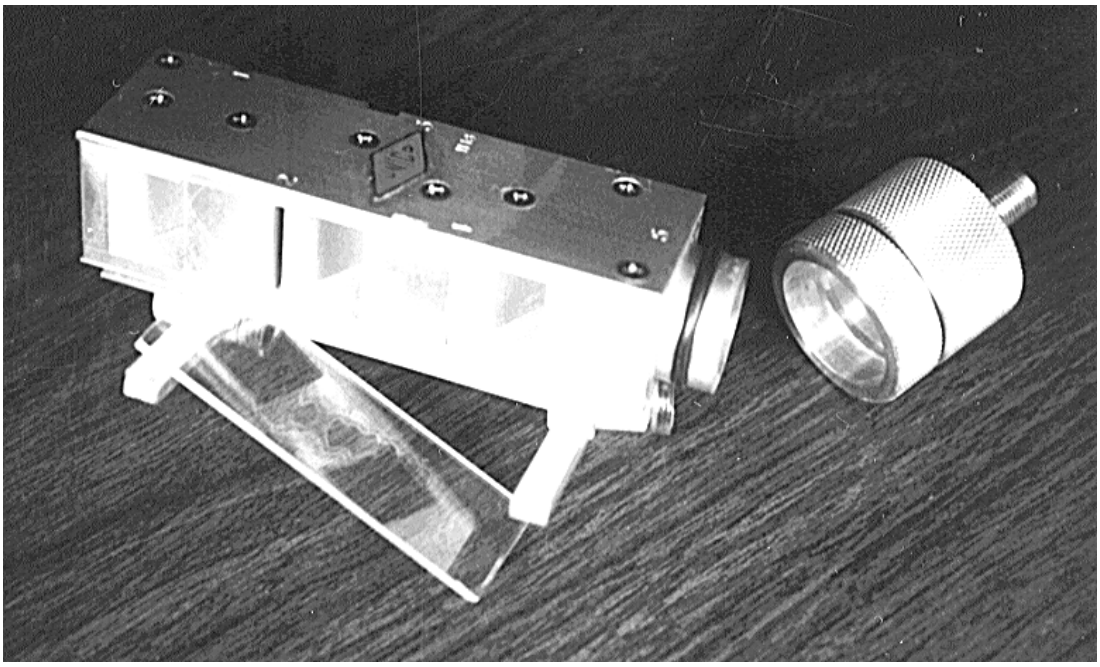


Fig. 19 - A cascade impactor to measure particulate size distribution

direction. The gaseous component can easily change direction, but the inertia of the airborne particulates prevents them from executing a sharp bend. They are removed at the bend and later counted to determine the air concentration of radioactive particulates. A cascade impactor is one particular instrument which illustrates this technique. A photo of an impactor is shown in Figure 19.

The particles which are unable to negotiate the sharp turns of the cascade impactor are "plated out" on the grease-coated glass microscope slide. The restrictions become smaller as the air stream proceeds down the device. This increases the stream velocity at each stage and so smaller and smaller particles are removed. The final stage is a membrane filter. The cascade impactor is particularly useful in radiation protection for determining the size distribution of airborne radioactive particulates. This information is needed to correctly determine the internal dose to a person who inhales radioactive particulates. The sizes of the particles determines how they distribute throughout the nasal passage, bronchi and lungs.

Filtration means the entrapment of radioactive particles in some matrix. The conventional filter paper used in chemistry experiments is often the matrix of choice. The tangle of paper fibers catches particles as the air stream is drawn through by an air pump. The filter is later removed and counted. As usual, alpha and beta activities are determined using conventional counters such as a gas flow proportional counter. Corrections for self-shielding are made if necessary.

Two factors need to be considered in operating a filter sampler. Dust loading refers to the accumulation of nonradioactive material on the filter element as time passes. In a dusty environment, this might become significant in a matter of minutes to hours. As the filter becomes plugged up with dust, the airflow rate usually drops off

but the collection efficiency (the fraction of particles passing through which are trapped) usually increases. These effects must be taken into account in calculating the air concentration of the activity.

The other factor to be noted is the change in efficiency of a filter matrix as the airflow velocity changes. The graph in Figure 20 gives the general behavior of the efficiency of collection of a filter for some size distribution as the velocity of impact of the particles changes.

Most filters show the effect illustrated. At very low and very high velocity, the filter gives high collection efficiency. At intermediate velocities of the air stream, the filter performs less satisfactorily. The reason for this strange behavior is made clearer in Figure 20 in which the total efficiency curve is "resolved" into two component curves.

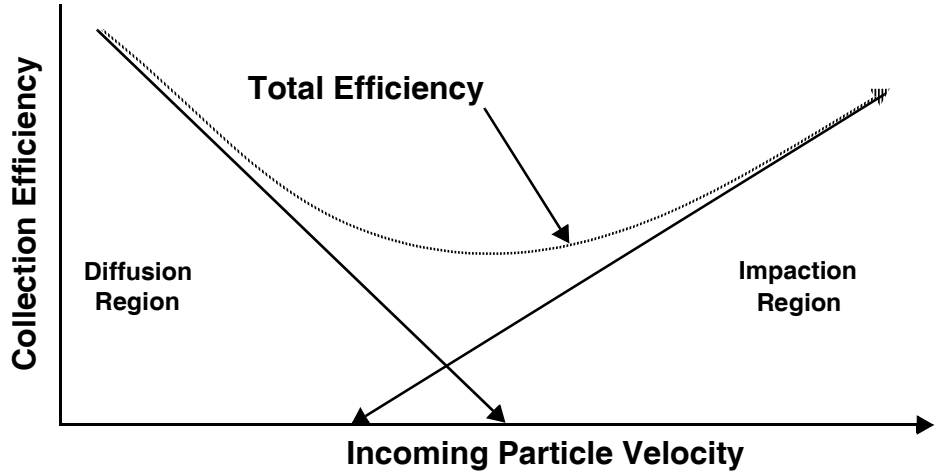


Fig. 20 - The two phenomena which influence filter efficiency

The high efficiency at low velocities is due to the phenomenon of diffusion (the slow moving particles drift off the path and become trapped in dead air spaces in the filter matrix) while the high efficiency at high velocities is due to impaction (the inertia drives the particles into the filter matrix where they are buried). The practical implication for the practicing radiation protection technologist is that THE FILTER OF AN AIR SAMPLER MUST ALWAYS BE REPLACED WITH THE IDENTICAL TYPE SPECIFIED BY THE SAMPLER MANUFACTURER. The manufacturer has matched the airflow characteristics of a particular sampler to the filter matrix supplied with it. If the technologist chooses a different filter matrix (perhaps a cheaper one?) then the chances are that the collection efficiency will no longer be as stated in the equipment specifications. This will introduce an error into the air concentration calculations.

An example of a portable fixed filter sampler is the high volume unit pictured in Figure 21. This instrument consists of a high velocity turbine air pump, a flow meter and a filter paper holder. It is useful for spot sampling situations. Under extended running conditions, it is subject to dust loading since it operates with a high air volume flow rate. The model pictured runs at 2 to 12 cfm, and has a convenient digital

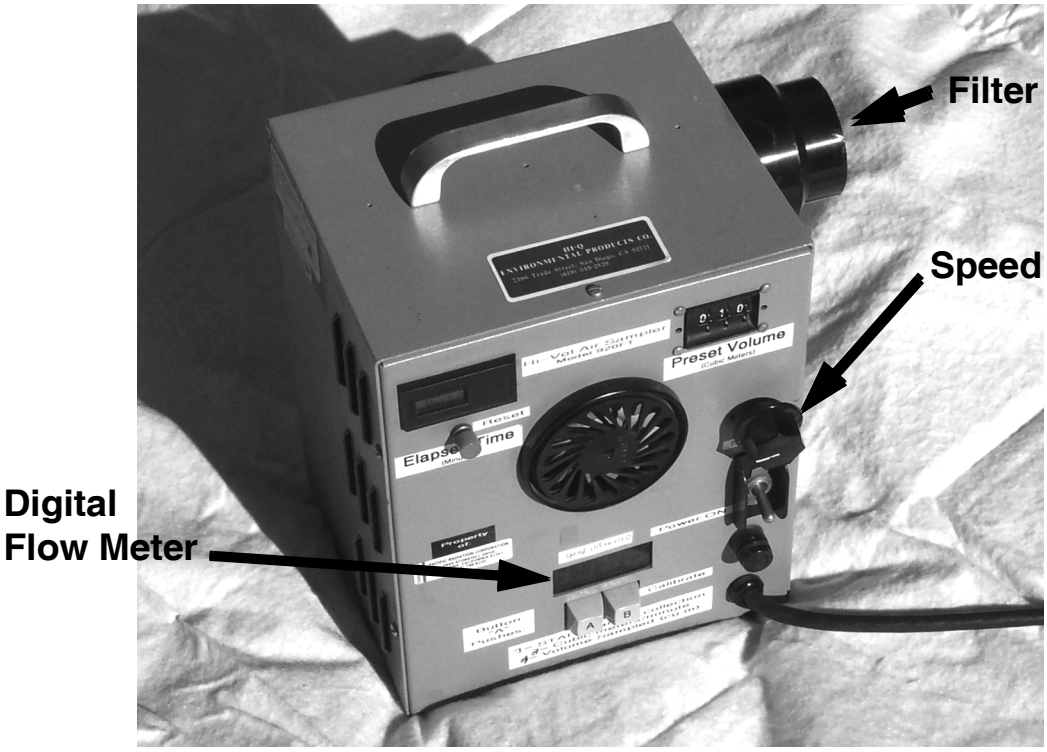


Fig. 21 - A high volume air sampler

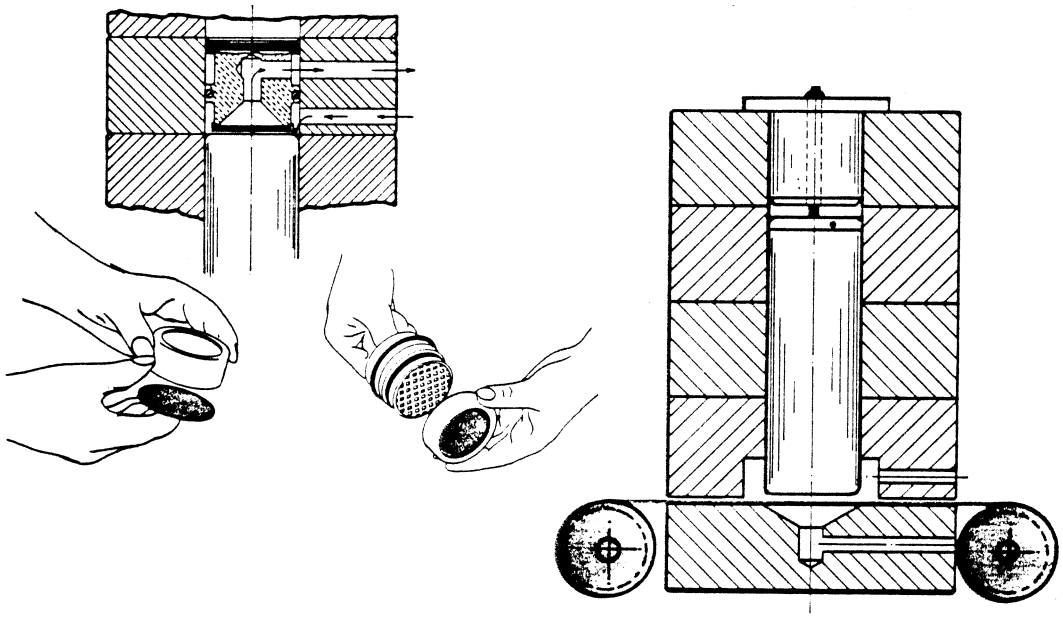


Fig. 22 - Operation of fixed and moving filter samplers

Courtesy, Nuclear Measurements Corp.

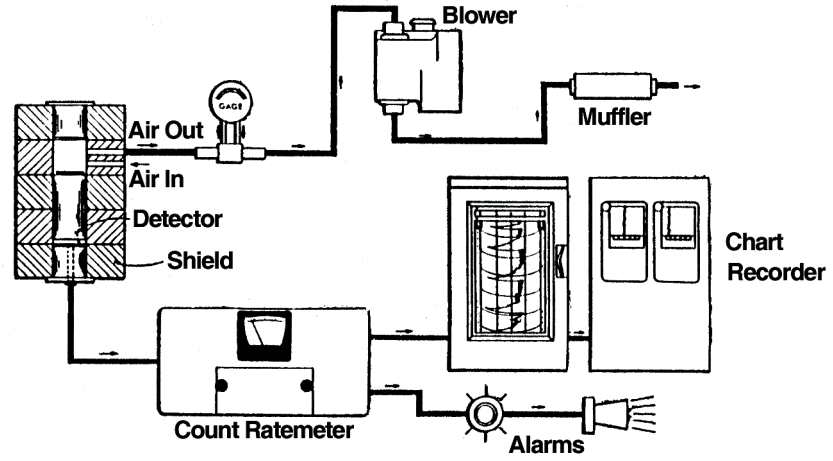


Fig. 23 - Block diagram for sampler of Figure 24

Courtesy, Nuclear Measurements Corp.

readout of the total volume of air collected. It can be preset to collect a given volume of air and then shut down automatically.

One clever solution to the problem of dust loading on a fixed filter sampler is the moving filter sampler. In this device, the filter is, in effect, replaced continuously. The filter medium is usually in the form of a roll. It is pulled slowly across the airflow orifice by a motor drive and so the filter is no longer subject to dust loading. Figure 22 is a sketch of the principles of fixed and moving filter samplers. Figure 23 is a block diagram of a commercially available moving filter air particulate sampler. The sampler itself is shown in the photo of Figure 24.

The last technique mentioned for sampling air particulates is electrostasis. An electrostatic precipitator uses this technique. A high voltage electrode wire releases electrons which attach to dust particles. The Coulomb force of repulsion then drives the charged dust particles to the outside collecting plate. Such samplers have occasional use in radiation protection technology. They are characterized by a very constant airflow rate and by very high collection efficiency.

Finally, consider the case of sampling particulates in a moving air stream. This is the topic of stack sampling. A stack (as in smokestack) is a vent pipe or duct carrying a stream of air molecules and particulates, possibly for release into the environment. If the particulates can possibly be radioactive, their concentration, ($\mu Ci/cm^3$), must be measured before release. This is done by inserting a pickup nozzle into the air stream inside the stack and withdrawing a sample for analysis. The chief problem in this operation is to extract a "representative sample" of the stack concentration. Does your collected sample have the same activity concentration as the average value flowing down the pipe?

This involves three considerations - the type of nozzle, the placement of the nozzle in the air stream and the transport of the particulates to your collection point. The nozzle has to be designed to meet the conditions of "isokinetic sampling." These conditions are met if the velocity of the air stream entering the nozzle is exactly equal to the velocity in the duct at the sampling point. Alternatively, the air pressure inside

Courtesy, Nuclear Measurements Corp.

Fig. 24 - A commercial moving filter sampler

the nozzle must equal the pressure just outside in the air stream. If the pressure inside the nozzle is lower than the duct air stream, a suction is created that will drag in particulates that would have normally missed the nozzle opening. Hence, the collected sample will have a higher activity concentration than the air stream. Conversely, if the nozzle pressure is higher than ambient, a volume of air will build up at the nozzle that will disperse the arriving particles (analogous to too many cars trying to exit a freeway ramp at the same time). Fewer than the representative number will be collected and the activity concentration will be underestimated.

The other problem with the nozzle is placement in the duct. The central axis of a cylindrical pipe is NOT the location of the average velocity in the pipe. Due to the friction exerted by the pipe wall, the layer of air nearest the wall moves slower than the layers closer to the centerline. Also, any sharp bends in the ductwork will cause a disruption in the air stream. Thus, someone with training in the engineering aspects of air stream flow in duct work should be consulted for the correct placement of the nozzle in the stack to assure isokinetic sampling.

The last of the three considerations is how to transport the sample to the collection point. In a complicated facility, e.g., a nuclear power reactor, the sampling line between the nozzle and the collection point may well show losses of close to 100% of the particulates. The actual value is determined by the air velocity in the line, by the line diameter, by the length of the run and by the number of changes in direction (bends) in the sampling line. Useful information on the design of air sampling systems can be found in NRC Regulatory Guide 8.25, **Air Sampling in the Workplace**.

Air Sampling - Gases

There are several different approaches that can be taken to monitor the environment for radioactive gases. These gases result from the decay of natural radionuclides (e.g., radon) or are a result of artificial production in reactors (fission products such as krypton and xenon) and accelerators (argon, oxygen, nitrogen, etc.). Four techniques used to detect radioactive gases are listed in Figure 25 along with a practical example for each.

Technique	Example
Continuous Flow	Stack Monitor
Grab Sample	Lucas Cell
Adsorption	Charcoal Canister
Condensation	Tritium Monitor

Fig. 25 - Some air sampling techniques for gases

The continuous flow counter principle is widely used for radioactive gas monitors. The air to be sampled is pumped through a chamber containing the appropriate radiation detector and then exhausted. The counter output can be connected to a rate meter to obtain instantaneous readout of radioactive gas concentration. An airflow meter and filter stages to remove particulates which would cause interfering activity complete the design. A block diagram of a typical unit is shown in Figure 26.

A commercially available instrument designed to monitor levels of Kr-85 gas in room air is shown in Figure 27. This instrument, adapted by Pacific Radiation Corp., has a block diagram identical to Figure 26. It can be calibrated by connecting the monitor in a "closed loop." This just means that the output exhaust port is connected by tubing to the inlet port so that the same trapped air is continuously recirculated. A known concentration of radioactive gas is injected into the loop and the instrument is adjusted to read the correct concentration.

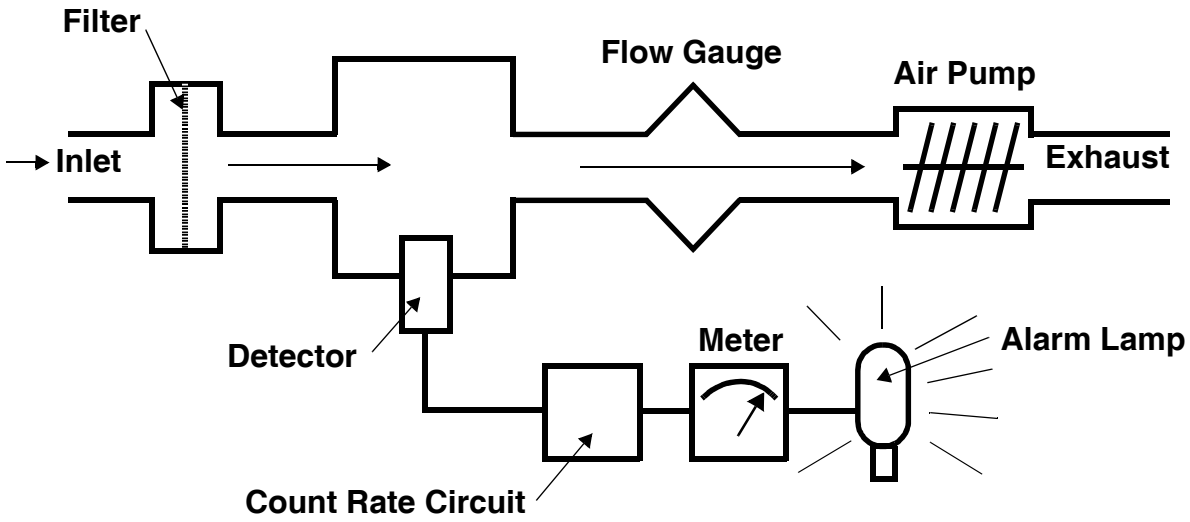


Fig. 26 - Block diagram of a continuous flow sampler

Continuous flow samplers are often calibrated in terms of "DAC" or Derived Air Concentration units. The DAC concept was introduced by the ICRP to assist radiation protection technologists in determining the hazard associated with air concentrations of radionuclides in the workplace. The particular value, expressed in microcuries per milliliter of air in the USA or becquerels per cubic meter in the rest of the world, depends on the radionuclide and on the chemical solubility class. As an example, for the radioactive gas Kr-85, the DAC is 1×10^{-4} μCi/ml.

The numerical value is calculated very easily. Recall from Chapter 9 that the Annual Limit on Intake, ALI, is the radioactivity taken into the body in one year which would deliver a committed effective dose equivalent, CEDE, of 50 mSv (5 rem). If the ALI is divided by the total volume of air that Reference Man breathes (2.4×10^9 ml or

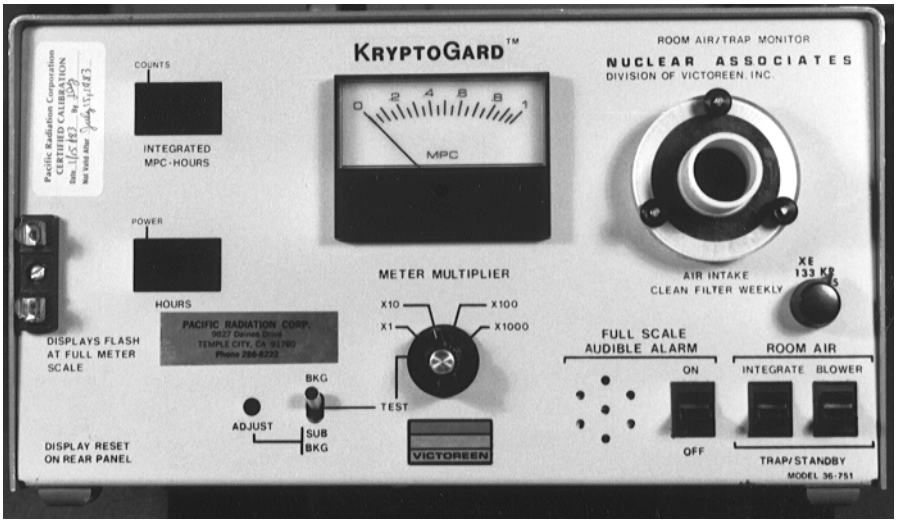


Fig. 27 - A room air monitor for krypton-85

Sample Problem 1

GIVEN:
A laboratory technician is exposed to airborne potassium-40.
FIND:
What is the calculated DAC for this case?
SOLUTION:
From Chapter 9, Figure 34, the inhalation ALI is 400 µCi. Substituting in Eqn. 1 gives:
DAC = 400 µCi / 2.4 x 10^{9} ml = 2 x 10^{-7} µCi/ml. Note that due to the many assumptions in the models, ALI and DAC values are expressed to only one significant figure.

2400 m^3) during a 2000 hour work year (40 hours/week x 50 weeks = 2000), the result is the DAC (See Sample Problem 1):

DAC (µCi/ml) = ALI (µCi) / 2.4 x 10^{9} ml [Eqn. 1]

In other words, if the air concentration is exactly equal to the DAC for one year, a worker will just reach the legal intake of radioactivity, one ALI, on December 31st. If the air concentration never exceeds the DAC in a given workplace, then workers will never exceed the maximum inhalation limits. The DAC is thus a convenient reference for comparison purposes when performing air sampling. One further clarification – a worker not receiving any other radiation doses except by inhalation would be allowed a cumulative exposure of 2000 DAC-hours in one working year. This idea of keeping track of the total DAC-hours of exposure is a convenient way to assure compliance with the radiation control regulations. See Sample Problem 2.

Another type of instrument that uses the continuous flow principle is a gas stack monitor. This instrument measures the concentration of a radioactive gas in a ventilation stack where it is being released to the environment. An example is shown in Figure 28. The connection of the detector to the stack is illustrated by Figure 29. This particular detector is a scintillation counter using a thin beta sensitive plastic

Sample Problem 2

GIVEN:
A DOE weapons facility worker is exposed for 8 hours to an air concentration of 6 x 10^{-11} µCi/ml of PuO_2.
FIND:
What is the dose (CEDE) from this exposure?
SOLUTION:
From Chap. 9, Fig. 34, the inhalation ALI = 0.02 µCi. Thus the DAC is, from Eqn. 1, 0.02 µCi / 2.4 x 10^{9} ml = 8 x 10^{-12} µCi/ml. The air concentration represents 6 x 10^{-11} µCi/ml / 8 x 10^{-12} µCi/ml per DAC = 7.5 x DAC. Thus, the cumulative exposure (concentration x time) was 7.5 DAC x 8 hrs = 60 DAC-hrs. Since 2000 DAC-hrs delivers a CEDE of 5 rem (0.05 Sv), this cumulative exposure delivered (60 / 2000) x 5 rem = 0.15 rem of committed effective dose equivalent.

Fig. 28 - A radioactive gas stack monitor

scintillator coupled to the photomultiplier tube. The detector is then surrounded by lead shielding to reduce the background count rate. The ratemeter has a logarithmic response to allow coverage of levels from below background to disaster concentrations without range switching. The 30 day strip chart recorder gives a continuous record and makes it possible to calculate the amount of radioactivity released.

Note that the gas stack sampler is not plagued with the many problems discussed earlier that characterized a particulate stack sampler. The question of isokinetic sampling does not even come up since there are no radioactive particulates involved in gas sampling.

The grab sampler approach is to collect a fixed volume of air at the sampling point and then transport it to a laboratory for analysis. The sample is actually taken

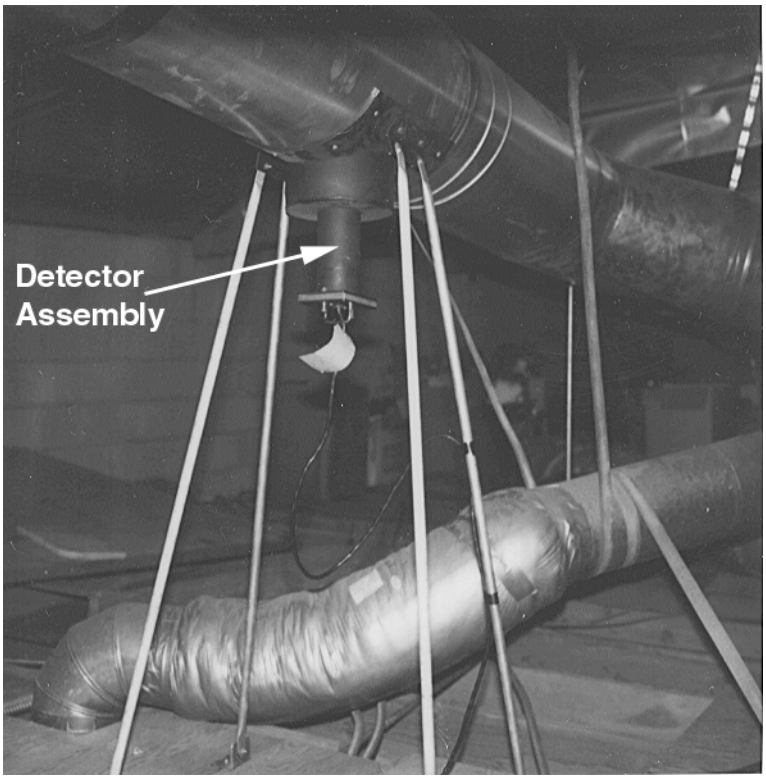


Fig. 29 - Detector connection to the stack

by opening an evacuated vessel at the desired location. The airflows in to fill the vacuum and is then sealed in. This principle is commonly used in uranium mines. The mine atmosphere has high levels of alpha emitters from the various radioactive daughters of radon. Figure 30 shows a sample of this type of monitor called a "Lucas Cell."

The device is simply operated. The air is pumped out through the stopcock. In the mine, the stopcock is opened to admit the sample and then closed. Back in the laboratory, the Lucas cell is placed on top of a photomultiplier tube inside a light-tight box. The sidewalls and top surface inside the cylinder are coated with ZnS(Ag) scintillation phosphor making it an alpha particle scintillator. The radon gas decays inside the cylinder and produces light flashes that are counted to give the air concentration. It can be simply calibrated by injecting a known concentration of radon from a calibrated radon bubbler.

Figure 30 also shows a commercially built Lucas cell counter. The photomultiplier tube is inside a light-tight housing. A mechanically activated switch in the base of the housing turns off the high voltage to the photomultiplier tube when the cover is removed, protecting the sensitive tube from damage by ambient room light.

The final radioactive gas technique, condensation, is used to sample tritium in the form of water vapor, i.e., one or both of the hydrogens of the water molecule have been replaced by H-3. These "mutant" forms of water are referred to as HTO and T_2O.

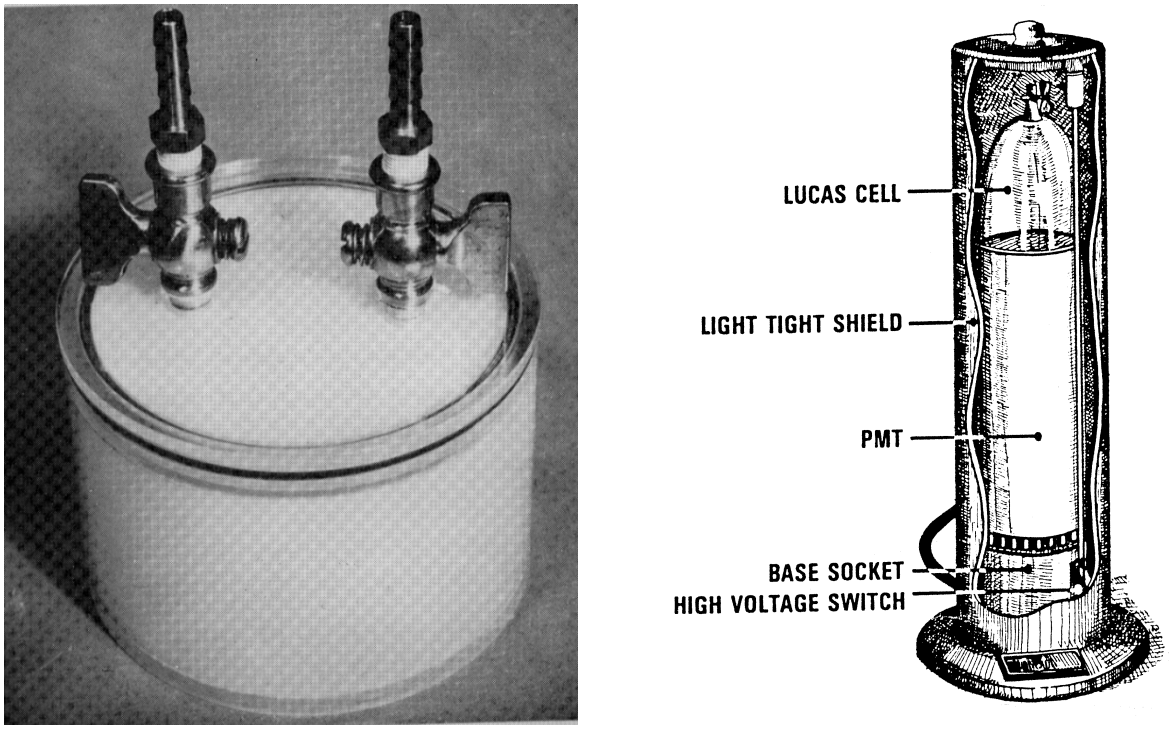


Fig. 30 - A Lucas Cell grab sampler and a commercial example

The condensation sampling method involves use of a container of liquid nitrogen (at minus 196° C.) with a “cold finger” consisting of a strip of copper extending up out of the liquid into the air. The cold metal surface will freeze out any water vapor in its vicinity. The ice that forms is then melted into a glass vial and the resulting liquid is counted for tritium activity in a liquid scintillation counter.

Air Sampling - Radon Gas In Buildings

It has become clear that a possible major public health problem involving radiation exists. This is the problem of radon gas emanating from soil and water supplies into homes and workplaces. It is a widespread problem with homes having been identified in all 50 states with levels above EPA standards. In 1986, EPA estimated that 6 to 12% of all U.S. homes have indoor radon levels exceeding the 4 pCi/liter recommended maximum. Several thousand cancer deaths annually are attributed to this exposure pathway. As a result, there is considerable interest in locating houses with potential problems. This can be done quite easily by performing gamma ray spectroscopy on charcoal canisters. The public health aspect of the radon problem is covered in more detail near the end of this chapter.

The charcoal canister is one method for inexpensively screening houses for radon level. A typical canister is 3" in diameter by 1" high, has a hole in the lid covered by a “diffusion barrier” to increase the average sampling time, and holds about 25 grams of activated charcoal (Figure 31). Over the course of several days’ exposure, radon gas diffuses into the canister and adsorbs onto the charcoal. At the end of the

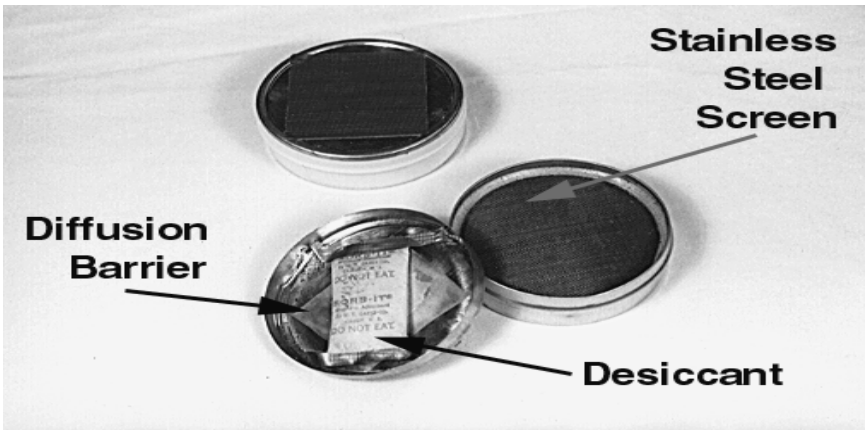


Fig. 31 - A charcoal canister for radon gas measurements

exposure period, the canister is resealed and mailed back to the analytical lab for counting. Figure 32 shows a radon canister counting setup.

The ^{222}Rn atoms decay with a 3.82 day half-life to a series of 4 short-lived daughters before encountering the long-lived Pb-210 daughter with a 22 year half-life which effectively stops the chain (review Figure 8 in this Chapter again). Fortunately, two of the four daughters are gamma emitters, Pb-214 with gamma energies of 242 keV (7.5%), 295 keV (19%) and 352 keV (37%) and Bi-214 with an energy of 609 keV (46%). The sum of the photo peak areas in the gamma pulse height spectrum is directly proportional to the average radon gas concentration when corrected for exposure time and for radioactive decay prior to counting. The "calibration factors" needed to evaluate a given canister count are usually obtained by exposing sample canisters to known radon concentrations in an environmental test chamber and then counting the canisters upon their removal.

A special dosimetry unit is often used when discussing radon exposures, the "Working Level," WL. Strictly speaking, the WL applies to the radon daughter products only. The formal definition of one WL is "any combination of short-lived radon daughters in one liter of air that will result in the emission of 1.3 X 10^5 MeV of potential alpha energy." If radon gas was present at a concentration of 100 pCi/liter, and if the daughter products were all in equilibrium, then the decay through the first 4 daughters would release exactly 1.3 X 10^5 MeV of alpha particle energy. Under the conditions described, 10% of the "alpha dose" is due to Radium A (Po-218), 52% to Radium B (Pb-214) and 38% to Radium C (Bi-214). The cumulative exposure to the radon daughters is measured in Working Level Months, WLM, the product of the air concentration in WL and the exposure time in months.

It is interesting to note that the radiation dose to the lungs from inhaled radon gas is not really a problem. The gas delivers only 1% of the dose. The other 99% comes from the decay of the radon daughter products! In a sense, the public hysteria concerning radon gas measurements is totally misplaced – the real culprit is the daughters. However, it is much easier to measure the gas concentration than the daughter concentration, i.e., the WL. Therefore, public health officials have taken the reasonable

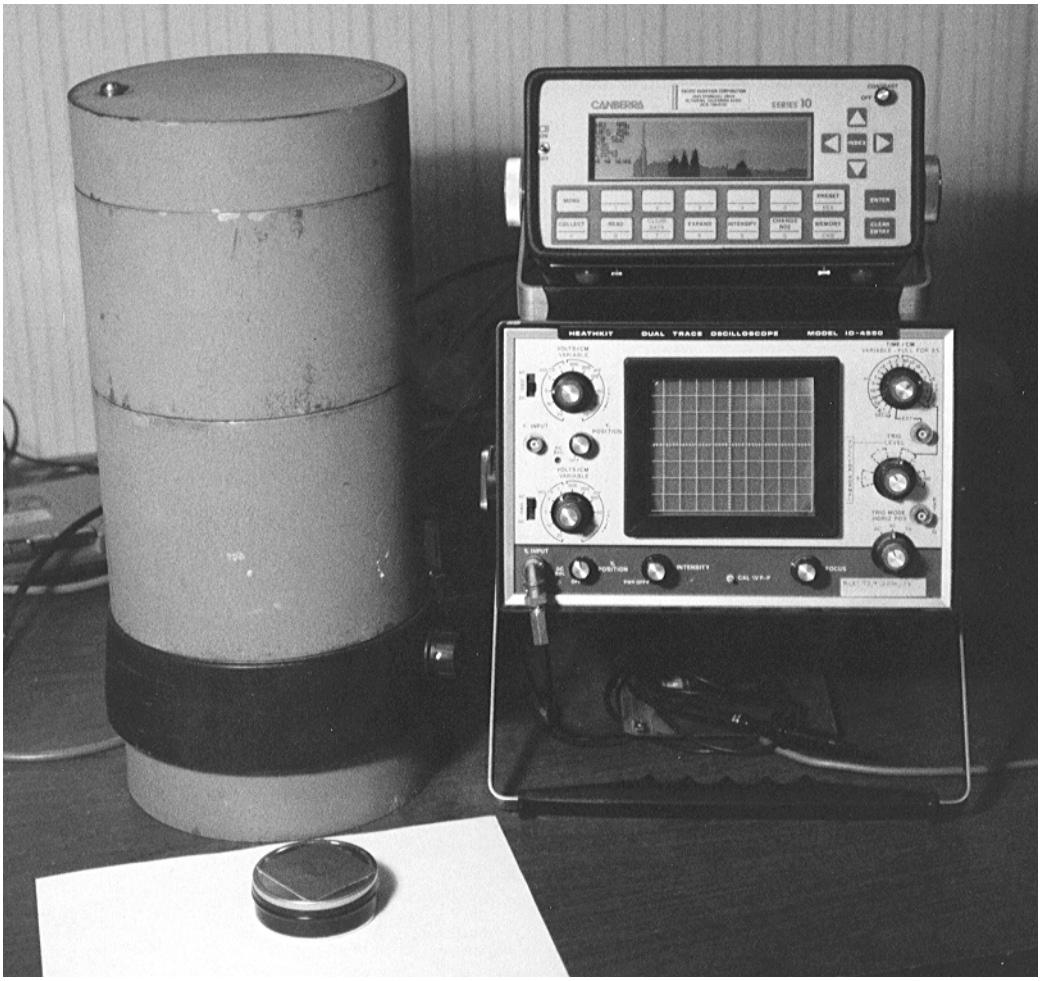


Fig. 32 - A radon canister counting setup

position that controlling the gas will keep exposures from the daughters at an acceptable level. The EPA guideline for residential radon gas levels, 4 pCi/l, is about 10 times lower than the occupational MPC for radon + daughters of 30 pCi/l.

Based on a 1984 NCRP report, the U.S. average outdoor radon concentration is about 0.1 pCi/liter. The radon daughter outdoor concentration averages 0.0006 WL. Typical indoor radon levels, excluding regions with high uranium concentrations, run 0.3 to 3 pCi/l while the indoor daughters are typically around 0.004 WL. Of course, in uranium mines the levels are much higher. Typical radon levels run 10 to 500 pCi/l and the daughters run 0.1 to 2 WL.

A novel method for radon measurements is the electret chamber device. An electret is a material that can hold a permanent electrostatic charge. One commercial supplier of electret radon monitors uses a teflon disk that holds a charge through both humidity and temperature changes. Radon diffuses into a chamber containing the electret and the ionization caused by the radon gas decays cancels some of the

electrostatic charge on the positively charged electret. The surface voltage of the electret is measured before and after radon exposure, and the voltage drop reveals the average radon concentration. Since the device is reusable, the measurement cost is low. The insensitivity to humidity and temperature changes is an advantage over charcoal adsorbers where humidity corrections must be made. Finally, the readout can be done in the field and the results obtained in a few seconds. See Figure 33 for a description of a commercially available electret monitor.

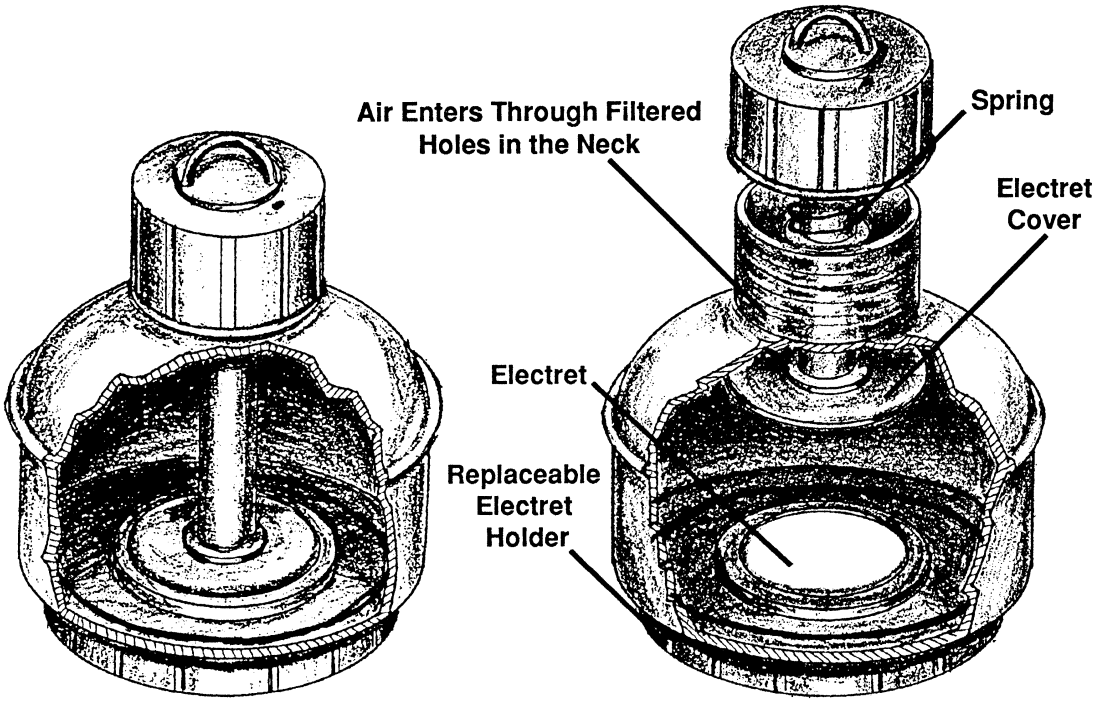
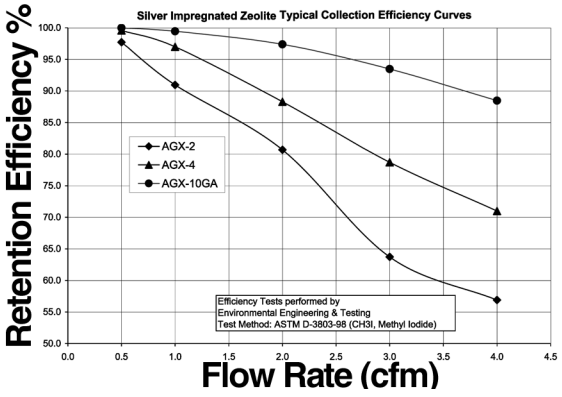


Fig. 33 - An electret radon monitor

Courtesy, Rad Elec Inc.

Air Sampling - Radioiodine

Sampling for radioactive isotopes of iodine is complicated by the chemistry of the element. In pure form, it is a dark colored solid and is a member of the halogen family of chemical elements. At room temperature, the solid sublimates, i.e., releases atoms of iodine as a gas. In this form, it tends to attach readily to the nearest surface or ambient dust particle. Iodine readily forms compounds. Thus, iodine might exist in the occupational environment as elemental iodine, as particulate (attached to dust) or as some chemical compound.



Courtesy Hi-Q Environmental Products Co., Inc.
7386 Trade St., San Diego, CA 92121

Fig. 34 - A silver zeolite cartridge and its efficiency curves

One of the more common locations requiring radioiodine sampling is at nuclear reactor sites. Federal law (specifically, NRC Regulatory Guide 1.97) requires three levels of monitoring. The in-plant air must be sampled during normal operations, the environs outside the plant must be sampled periodically, and the plant must have special high range instruments available for post-accident monitoring.

Routine monitoring usually means sucking air through a collection medium with a high volume or low volume sampler. The medium for radioiodine capture is available in two different types - activated charcoal or silver impregnated zeolite cartridges. Both types exhibit similar collection efficiencies. The unit cost of the charcoal cartridges is several times lower than the silver zeolite. Figure 34 shows a photo of a typical cartridge and a plot of the iodine collection efficiency vs. airflow rate.

Adsorption is the process whereby gas molecules will attach to the surface of some substance (the adsorbent). Good adsorbents are materials such as silica gel, activated charcoal (charcoal that has been heated to a high temperature to drive off trapped gases) and silver zeolite (AgZ). After removal from the sampler, the canister is counted by placing it next to a scintillation or germanium counter. The radioiodine gamma rays easily pass through the wall of the cartridge for counting.

For the highest collection in charcoal, it can be impregnated with triethylene diamine (TEDA) which chelates the iodine and reduces desorption from the cartridge. If an air stream is passed through, the cartridge will adsorb radioiodine and noble gases, e.g., xenon and krypton. The noble gases are a major hazard in a reactor accident involving any rupture of the fuel elements. Although the majority of the noble gas activity passes on through the charcoal, the less than 1% that is trapped complicates the analysis. A solution to the noble gas problem is to substitute silver zeolite cartridges. They retain less than 0.007% of the Xe and Kr that are captured by charcoal.

There is a potentially dangerous situation with regard to use of silver zeolite. The problem concerns absorption of water from the gas stream which causes a hydration reaction in the de-hydrated AgZ. This chemical reaction releases heat. If the AgZ reaches a temperature of 150° F, it can cause an explosion if <u>hydrogen gas is present</u> in the sampling gas stream in concentrations above 4%.

At the upper range limit specified in Reg Guide 1.97 for post-accident radioiodine monitors, 100 microcuries/cm^3, an air sampler cartridge will accumulate almost 3 curies per minute at an airflow rate of only 1 cfm! For a single minute of sampling, this would produce a dose rate of about 6,000 mrem per hour a foot away from the cartridge. Clearly, steps must be taken by the plant, before an accident, to prepare special handling facilities for analyzing post-accident air sampler cartridges.

At the other extreme, Reg Guide 1.97 establishes specifications for nuclear power plant instrumentation during normal, non-accident environmental monitoring of radioiodine. In this case, the equipment must be able to detect over the activity concentration range of 10^{-9} to 10^{-3} μCi/cm^3. The struggle here is to have enough sensitivity to reach the low end of that range. That goal is usually reached by greatly extending the sampling time, normally used in-plant, to maximize the sample volume. Technologists will not have to worry about environmental charcoal cartridges producing high radiation areas!

When monitoring the workplace air for radioiodines, the Derived Air Concentration, DAC, discussed earlier in this Chapter is the key number. Recall that a radiation worker can receive the entire annual allowed dose equivalent of 5 rems (under US NRC regulations) from inhaling the DAC for 2000 hours. Thus, occupational radioiodine air monitors need to have a detection sensitivity below the DAC level. Figure 35 lists the USA DACs for common radioactive isotopes of iodine.

Nuclide	DAC (μCi/ml)	Nuclide	DAC (μCi/ml)
I-123	3 E-6	I-129	4 E-9
I-125	3 E-8	I-131	2 E-8

Fig. 35 - DAC values for iodine radioisotopes

Air Sampling - Tritium

The hydrogen radioisotope ^3H or tritium is the final "special case" to be covered under air sampling. Tritium decays by beta emission, in common with thousands of other radioactive materials. What makes tritium a special case is that the energy of the emitted particle is the lowest of virtually any beta emitter a technician is likely to ever encounter in their career. Extraordinary steps have to be taken to reliably monitor tritium, particularly at low concentrations. Another complicating factor is that tritium, analogous to iodine, can be found in the environment in more than one chemical form. Common forms for tritium in air include HT, T_2, HTO and T_2O, where "T" = ^3H. Both HT and T_2 are, of course, just diatomic hydrogen gas where one or both of the H atoms are the tritium isotope. The other forms are just molecular water with either one or two ^3H atoms substituted for the normal hydrogen atoms.

In terms of tritium air sampler design, a key parameter, again, is the value of the DAC. Since tritium is commonly found in air as both a gas and as water vapor, it would seem that the radiation protection technologist would be concerned about two DAC values. This is not the case. Tritiated water vapor can be absorbed easily

through the skin while tritium gas produces no skin dose from the extremely weak betas emitted. In fact, the DAC for tritiated water is 25,000 times lower than for tritium gas. Therefore, the US NRC has taken the regulatory position that one should "use the [water DAC] value as HT and T_2 oxidize in air and in the body to HTO." The tritiated water DAC value is 2 E-5 μCi/ml in the USA.

Tritium air samplers, often called tritium sniffers, suck ambient air through an ionization chamber. The block diagram of the instrument is identical to Figure 26 earlier in this Chapter. Due to the very low energy of the tritium betas, a very sensitive electrometer circuit must be used to detect the tiny current produced inside the chamber by tritium decays. The sniffer cannot tell the difference between the water vapor form and gaseous form of tritium. This is not a problem since the DAC is conservatively based on the assumption that all occupational tritium eventually ends up as the water vapor form.

The femto-Tech, Inc. Model PTM-1812 monitor is pictured in Figure 36. It contains a 375 cubic centimeter ion chamber to obtain the necessary sensitivity. The minimum detectable concentration is 1 μCi/cubic meter for this unit. It is capable of reading up to 20,000 μCi/cubic meter of tritium in air with ±10% accuracy.

In general, tritium sniffers require some tender, loving care. Numerous ambient conditions such as the presence of smoke, moisture, loose ions, radon, other radioactive gases, and gamma ray sources all produce a background reading. Frequent re-zeroing may be necessary if the work environment continually changes.

Courtesy, femto-Tech, Inc.

Fig. 36 - The femto-Tech tritium air monitor

Water Sampling

In the United States, ground water usually has a concentration of less than 20 picocuries (10^6 pCi = 1 μCi) of alpha activity per liter and less than 30 pCi per liter of beta activity. There are a couple of practical problems in groundwater sampling. If the water has a high silt level (in other words, it's muddy), then the clay in the silt will act as an ion exchange medium and trap radioactivity when the water is passed through a filter to remove the silt. This is usually dealt with by reporting TWO RESULTS for a water sample - the activity concentration in the filtered water and the activity per unit mass of the silt on the filter paper.

The second problem in collecting a water sample is assuring that it is a "representative sample." This merely means that the concentration in the sample is the

Fig. 37 - A Marinelli beaker and scintillation counter

average that exists throughout a large body of water (a stream or lake, for example). Unfortunately, due to temperature variations, currents and other natural factors, the concentration may vary considerably from point-to-point and day-to-day. One answer to the problem of a representative sample is to collect a composite sample. This means that small samples are collected periodically and/or at different locations and then all the small samples are combined together to make the composite sample submitted for analysis. This sample has a better chance of being closer to the average representative sample than any of the small samples making it up.

Glass and polyethylene bottles are usually used to collect water samples. As previously mentioned, the glass is necessary for samples containing tritium to reduce the possibility of losses through diffusion. To obtain the gross gamma activity, a Marinelli beaker is often used. This special beaker has a re-entrant cavity which fits over the gamma ray detector [NaI(Tl) crystal or germanium counter, see Figure 37]. This counting geometry increases the gamma sensitivity compared to standing a bottle on top of the detector. After the gamma concentration is measured, the water can then be evaporated to dryness. The residue is then counted for alpha and beta activity, usually in a proportional counter. Self-absorption can again be a problem if the residue is too thick.

Under the Safe Drinking Water Act of 1974, the Environmental Protection Agency was ordered to establish regulations for radionuclides in drinking water by specifying a "maximum contaminant level" or MCL. The agency proposed interim regulations in 1975 which listed separate MCL values for alpha emitters and beta/gamma emitters. In July 1991, revised proposed regulations were published. Final values were expected to be

released in 1993. As a result of the long and continuing delay, legal action was taken against the EPA. A court mandated deadline was imposed - publish by November 2000 or else! Final standards have now been released and they became effective in 2003. The maximum contaminant level for uranium is 30 μg/l, ^{226}Ra and ^{228}Ra were both set at 5 pCi/l, and an adjusted gross alpha activity MCL was set at 15 pCi/l (excluding radon and U). The beta/gamma MCL was established by limiting the annual effective dose equivalent to 4 mrem/yr total body or any single organ.

Food Sampling

Very few food samples are routinely collected in normal monitoring of the environment. The exception would be a nuclear power reactor site where fish and vegetable species are routinely covered. However, milk is routinely collected on a large scale. In the 60s and 70s, the U.S. participated in an intercontinental sampling program that included the Canadian Milk Network, the Pan American Milk Sampling Program, and the Pasteurized Milk Network, PMN (sponsored by the Environmental Protection Agency and the Public Health Service). The PMN program in the U.S. operated 63 stations across the country (see the map in Figure 38). Recalling the earlier mention of grass as an excellent collection medium for fallout fission products, it can be seen that milk is an excellent sampling medium.

Around the 1980s, EPA combined numerous federal monitoring programs. Evolution led eventually to the RADNET program. As of 2010, RADNET operated some 318 stations sampling air, rain, drinking water and milk for radioactivity. The RADNET milk stations are shown as an overlay in Figure 38.

Typical levels found in milk are of the order of 10s of picocuries per liter. Radionuclides sought include Sr, I, Ba and Cs. In analyzing milk for fission products, it is frequently passed through ion exchange columns. Iodine is the only anion fission product present in significant amounts in milk so it can be removed by the anion resin. The resin can then be counted directly in a gamma ray detector. The milk that passes through the column is then processed further to separate out the remaining strontium and cesium isotopes.

Environmental Problem Areas

Radon and Public Health

The relationship between indoor radon exposure and risk of contracting lung cancer was firmly rooted in controversy for over a decade. Reputable scientists were found on both sides of the issue. Many epidemiological studies failed to demonstrate a link between low levels of radon and lung cancer, but others argued that these studies are not valid indicators.

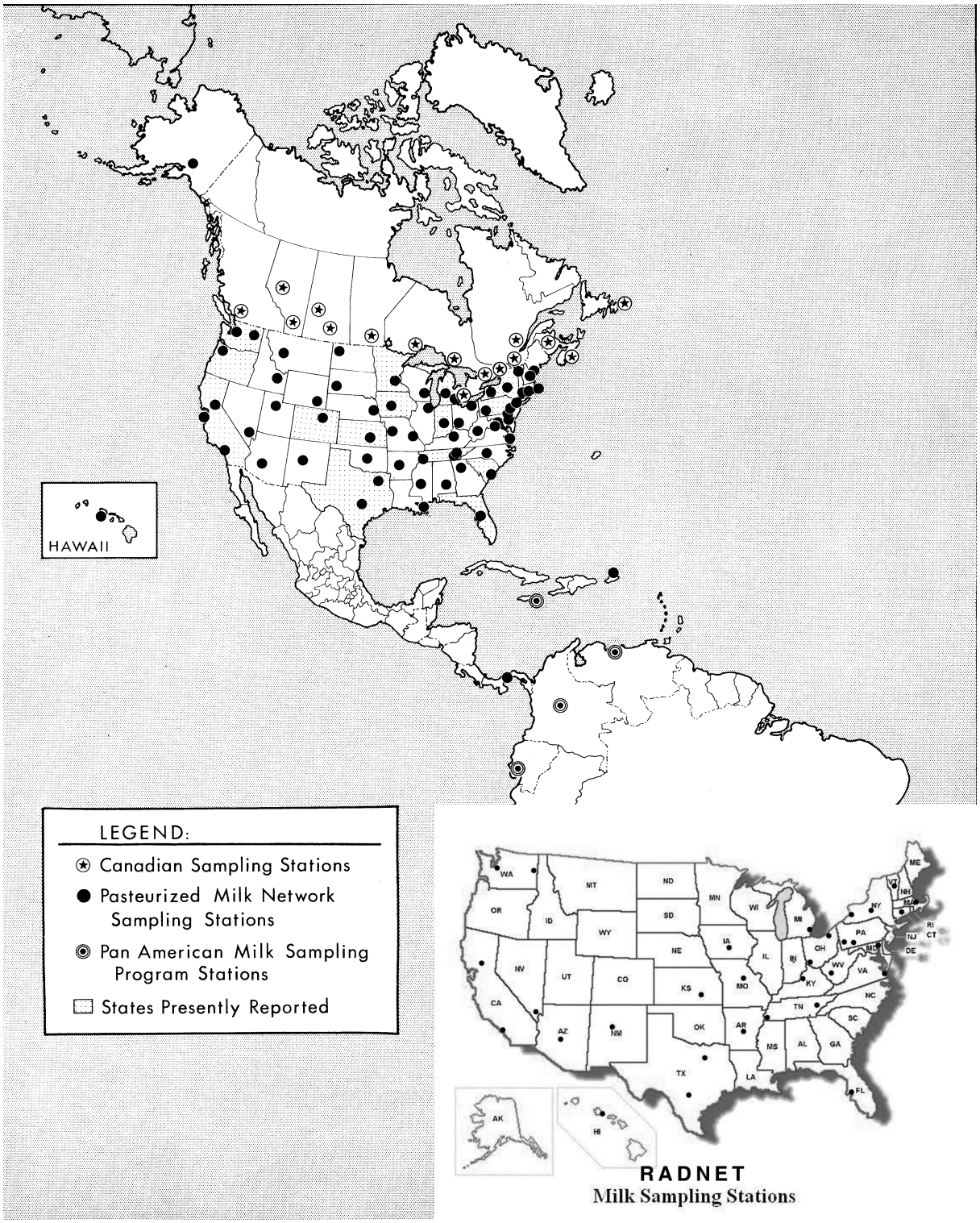


Fig. 38 - The Global Milk Network sampling station locations (1970s) and RADNET locations (2011)

In Chapter 4, under the topic of hormesis, large population radon studies in Finland, China, Pennsylvania and Sweden were mentioned. The results indicated that women living during the 50s and 60s in areas with higher than normal indoor radon levels showed lower than expected lung cancer rates. In the U.S. as a whole, a report by Bernard Cohen showed that there was an inversely proportional relation between lung cancer rate and the average radon level in a county. The higher the radon level, the lower the cancer rate. This study has been subjected to several criticisms.

One of the problems is that people living in the U.S. are quite mobile. On the average, families move every 4.8 years. Home owners are a bit more stable, moving on the average of every 11.7 years. But lung cancer shows a 5 to 30 year lag period between the time of exposure and disease onset. Thus, the county where a cancer is diagnosed in a subject may not have been the county of residence during most of the exposure period.

A more serious problem in evaluating population studies has to do with the problem of smokers versus non-smokers. In a 1990 "Position Paper," the Health Physics Society was critical of the direction the Environmental Protection Agency was moving regarding the indoor radon problem. The society's point was that cigarette smoking is the major cause of lung cancer. Even if all U.S. homes were modified to reduce levels below the 4 pCi/l guideline, the number of lung cancers would be reduced "only by a very small percentage of the total." A 1990 scientific study showed that 90% of U.S. lung cancer risk is due to smoking. By limiting the population studies to females in the 50s and 60s, the attempt was made to reduce the influence of smoking habits on the results. However, since smoking is such an overwhelming factor, it is difficult to eliminate the effects in a population study where the actual smoking habits of the individuals are not known.

Finally, a 2003 report demonstrated that some of the confusion may have been caused by the altitude at which the exposed persons resided. Some (but not all) of the negative correlation in the original Cohen radon study was cancelled out if the average elevation of the counties is considered. Apparently higher elevations result in lower lung cancer risk, possibly a result of lower oxygen concentration. Another 2003 review of the controversy over the Cohen radon study concluded that the debate appears to be over - there are plausible explanations for Cohen's inverse relationship between risk and radon concentration.

Instead of population studies, case-control studies are the preferred choice for radon risk evaluation. Unfortunately, these are much more time-consuming and expensive, as individuals must be identified to set up a case group (people with lung cancer) and a matched control group (persons without lung cancer). Then, an extensive interview is conducted to determine living habits, smoking history, etc. Measurements must be made of radon levels in the buildings occupied by the cases and the controls. Finally, some statistically significant conclusions may eventually be reached.

The results of a Swedish study were reported in the New England Journal of Medicine in 1994. It was the first large study completed. A total of 4207 cases were included, 1360 of them with lung cancer and the balance as controls. Radon levels in the homes involved had been measured for 23 years. Most of the lung cancers were verified by tissue sampling. In contrast to the Cohen study, it was reported that lung

cancer risk was higher in homes with higher radon concentrations. Compared to a 1.4 pCi/l reference level, the risk was 30% higher in homes between 4 and 11 pCi/l and 80% higher in homes above 11 pCi/l.

These results are consistent with the EPA risk estimates from indoor radon. Their most recent study was released in late 2003. It was triggered by the publication of the BEIR VI radon study. The EPA concludes that 13% of U.S. lung cancer deaths are caused by radon exposure. The absolute risk estimates are substantially higher than previously projected. The EPA estimates that 26% of lung cancers in persons who never smoked result from radon exposure. Further, if the population were all exposed at the 4 pCi/liter EPA action level, the cumulative lifetime risk of death is 0.73% for non-smokers, and 6.2% for persons currently smoking. These figures were arrived at using extrapolated values downward from the uranium miner risk estimated in the BEIR Committee report from the National Academy of Sciences.

Some Environmental Restoration Projects

With the ending of the "cold war," and the accompanying reductions in nuclear weapons production as well as the increasing availability of formerly classified documents, public demands have led to increasing emphasis on restoring nuclear facility sites to their preoperational levels of natural radioactivity. Projects along these lines are currently underway in several countries, not just the U.S. A few cases of contamination and planned restoration will be examined here briefly.

Between 1946 and 1958, 66 nuclear weapons tests were conducted on territory of the present Republic of the Marshall Islands on or near the Pacific atolls of Bikini and Enewetak. (See Figure 39.) The residents of Bikini, over the next 3 decades, were relocated 5 different times. Inhabitants of Enewetak and Rongelap were also resettled multiple times. During 1969, about ten years following the atmospheric nuclear test ban, Bikini underwent extensive cleanup and restoration procedures. The initial resettlers were moved back in 1972. Six years later, 140 Bikini residents exceeded the radionuclide body burden limits so the population was moved off the atoll again.

During 1980, at a cost of $218 million, the atoll of Enewetak underwent environmental restoration. It was resettled later that year. However, local sources of food were inadequate so 100 inhabitants decided to return to their previous island. The next year, the U.S. and Republic of Marshall Islands signed a formal agreement which finally became effective in 1986. This agreement set up a monetary fund to compensate islanders who could demonstrate personal loss or damage to property. To date, over 500 claims have been settled.

In addition to compensation for damages, the agreement also set up a radiological surveillance project. The goal is to survey all of the islands in the nation, not just those known to have received heavy fallout. An environmental lab was established in the capital city of Majuro, and the project was placed totally under the control of the Marshallese people. A main objective is to establish credibility. There is a long history of mistrust of DOE efforts and surveys on the part of the Marshallese. The main efforts of the survey project have been directed at ambient gamma ray

spectra and soil sampling of all of the 1200 islands making up the Republic. HPGe detectors are used for field gamma measurements. Although their superior energy resolution is not needed to quantify the rather simple fallout nuclide spectra, they were chosen to enhance the credibility of the project. In most locales, this equipment could be field operated quite simply. In the Republic of the Marshall Islands, the accessibility of liquid nitrogen, constant high humidity, frequent drenching rainstorms and corrosive salt air all combine to make the measurement a real challenge! During 1991 and 1992, 80% of the land area was surveyed. The levels of Cs-137 measured to date vary by a factor of 40,000, reflecting the wide variation in meteorological conditions and the large distances between the northern atolls (where tests occurred) and the southern atolls.

In terms of sheer size, the environmental restoration of the former Soviet Union dwarfs projects in all the rest of the world (see Figure 40 for a map of nuclear sites). Fallout from nuclear detonations was much more widespread than in the U.S. because all North American testing was done

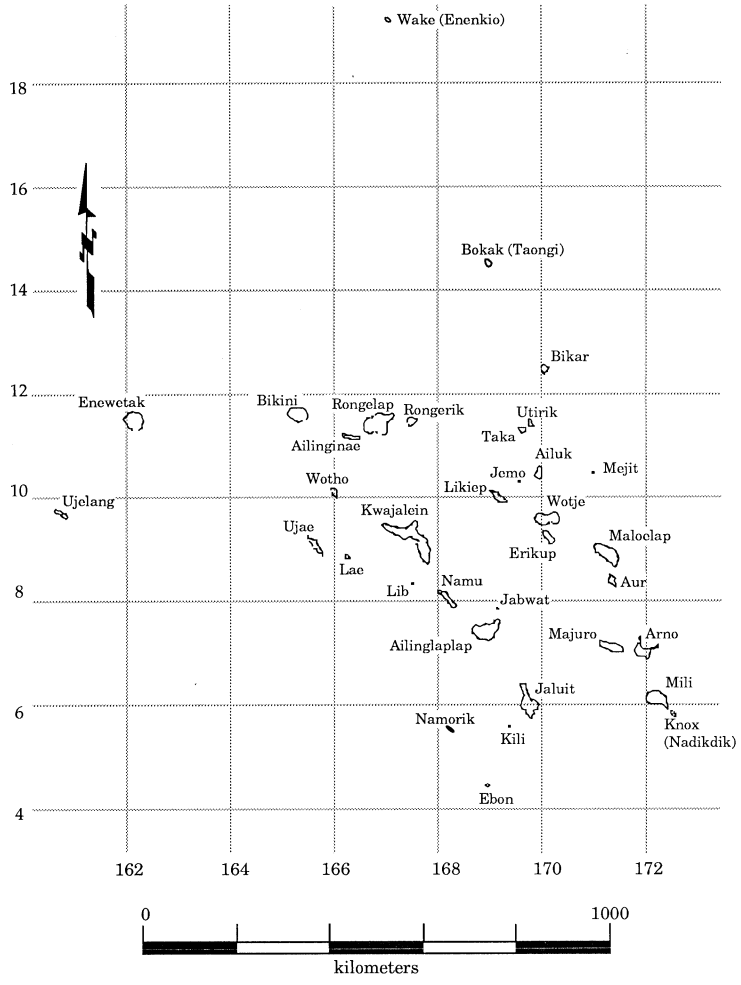


Fig. 39 - The Republic of the Marshall Islands

Courtesy, S.L. Simon *et. al.*, Proc. of the 26th Midyear Meeting, Health Physics Society, 1993 and RMI Historic Preservation Office

Fig. 40 - Nuclear sites in the former Soviet Union

at a single location in Nevada. The Soviets used two widely separated test sites – the Siberian area of Semipalatinsk (467 weapons tests, 122 of them in the atmosphere) and Novaya Zemlya in the Arctic Ocean. (There were 119 tests, total, at the Nevada Test Site.) In addition, the Soviets exploded 115 nuclear devices throughout most of the land mass of the Soviet Union for peaceful uses. These included dissipating methane gas in coal mines, oil exploration, production of vitrified underground cavities for oil and natural gas storage, to put out a fire at an oil well and to excavate a canal. Weapons production facilities were concentrated at two main locations. Near the Yenisey River in Siberia, at Krasnoyarsk, three large plutonium production reactors were built inside a huge cavern excavated from solid rock 180 meters below the surface. River water passed through the reactors for cooling and was recirculated back into the river with no treatment.

The fuel reprocessing and plutonium extraction took place at the super-secret Mayak Chemical Combine near Chelyabinsk at the southern end of the Ural Mountains. This facility released its liquid wastes into the Techa River. Between 1949-56, some 2.75 million curies, primarily ^{90}Sr and ^{137}Cs, were sent downriver. The river dumping was stopped when elevated levels of radioactivity were picked up where the river meets the Arctic Ocean, about 1600 km from the point of release. Downriver villages

were evacuated between 1953-61. The Chelyabinsk Institute of Physics and Biology estimates doses as high as 3.5 Sv/yr (350 rem/yr) to the maximally exposed villagers.

In 1989, Soviet officials finally admitted the occurrence of a huge explosion at the Mayak complex on September 29, 1957. A malfunction in the cooling system for a high level nitrate-acetate radioactive waste storage tank led to overheating and a chemical explosion. Fallout from an estimated 2 million curie release was spread over 600 miles downwind, involving three provinces. A population of over 10,000 people had to be relocated. No fatalities resulted from the accident. Most of the affected farm land was decontaminated and back in production by 1989.

In addition to accidental releases, the former Soviet Union had a long-standing policy of disposal of worn out reactors by sea burial. Many of the cores sunk in the Kara Sea east of Novaya Zemlya still contained their loading of uranium fuel. A related problem is Lake Karachai. Critics call it "the single most polluted spot on the face of the planet." Shoreline dose rates as high as 600 rem/hour have been measured! When waste releases into the Techa River were halted, the Chelyabinsk complex chose the lake instead. It now holds 120 million curies of waste open to the environment. One of the problems with open waste containers was demonstrated when, during a 1967 drought, a tornado picked up bottom sediment exposed when the water level dropped, and spread contamination over a large area. To reduce the chances of this happening again, the lake bottom is slowly being covered with stones and concrete blocks to stabilize the sediment.

Future plans for environmental restoration are proceeding slowly, due chiefly to the economic collapse caused by conversion to a free-market economy. A complete, detailed survey of the contaminated territory within the Russian Republic was ordered by Russian President Yeltsin. This survey was to be completed by the beginning of 1993. In addition, the Russians were trying to set up a data base of all radiation-exposed victims, both workers and members of the public. As of 2010, very little information appears in the published literature to verify an active environmental restoration program at any of these sites.

On a more optimistic note, some of these complex projects do reach a successful conclusion. At the end of 2005, the US DOE announced that the restoration of the Rocky Flats Nuclear Weapons Plant near Denver, CO had just been certified as complete. For close to 4 decades, the plant manufactured the plutonium triggers for the US nuclear weapons arsenal. When site cleanup work began in the mid 1990s, the costs were estimated by the DOE to be upwards of $37 billion and the cleanup time was projected to be 70 years! The actual project cost $7 billion and was concluded in about 9 years. This amazing performance was due to the approach taken early on in the cleanup. Plutonium and americium were thought to migrate through the air, water and soil of the site due to their formation of water soluble compounds. It was eventually proven by Los Alamos National Lab personnel (using "x-ray absorption fine structure spectroscopy" at the Stanford Synchrotron Radiation Lab) that the contaminants were in a low solubility hydrous oxide form which would be spread mostly as particulates, confined to a thin surface layer. By performing decontamination inside large tents, wind blown and rainwater

carried particles were controlled so well that cleanup proceeded much more rapidly than originally planned.

This concludes the Radiation Protection Instrumentation Unit of the text. The third and final Radiation Protection Operations Unit will deal with various practical aspects of the profession of radiation protection technology.

Problem Set

1. Briefly describe how environmental monitoring results are used to estimate population radiation doses near a nuclear facility.

2. What is meant by the "radiation impact" of a nuclear licensee? How is it measured?

3. Distinguish the terms "pathway," "critical pathway" and "critical nuclide." Make a simple sketch to illustrate a common critical pathway for fission product fallout.

4. Why must a radionuclide have a reasonably long half-life in order to be measured in an environmental monitoring program? Give two reasons why short-lived nuclides probably won't give a large contribution to the population dose near a nuclear facility.

5. Describe some of the major differences between a preoperational and a postoperational monitoring program.

6. Name some practical problems which make it more difficult to achieve "sensitivity and selectivity" in a postoperational monitoring program. How might these problems be solved?

7. What characteristics are needed in an environmental dosimeter for "direct radiation?" Name one practical device that meets these characteristics.

8. Name two methods used to determine the amount of surface-deposited radioactivity at a site.

9. What two factors need to be taken into account in interpreting data from a fixed filter sampler? How can their effects be minimized?

10. Based on measurements in a semiconductor leak-test facility, workers in the area are continuously exposed to an average concentration of 2 microcuries of Kr-85 per cubic meter of air. What fraction of DAC does this represent? Estimate their annual radiation dose.

11. Calculate the DAC for americium-241.

12. Why does a stack monitor usually have a logarithmic readout?

13. A radon charcoal canister counter has a 1 pCi/l detection limit. A canister is exposed to 12 pCi/l of radon. How long does the counting lab have after the canister is resealed to count it and still detect radon?

14. What is meant by a "representative" water sample? Why is a composite sample usually more representative than a single grab sample?

15. How is the problem of silt in a water sample handled?

16. Why are water samples containing tritium collected in glass containers?

17. Why is milk commonly sampled for radioactivity? How is it processed at the analysis lab?

18. What is the chief problem in evaluating population studies attempting to correlate radon gas exposure in the environment with the probability of contracting lung cancer?

19. Why is it felt necessary to "reconstruct doses" to populations who were exposed to radionuclide releases near nuclear facilities?

S-1. Why is a nuclear power station that uses a river for cooling water required to collect environmental samples UPSTREAM from the discharge point?

S-2. How was the "reporting level" determined by the NRC? How is it used in a postoperational monitoring program?

S-3. What unique problems have been encountered in the national radiation survey of the Republic of the Marshall Islands?

Other Resources

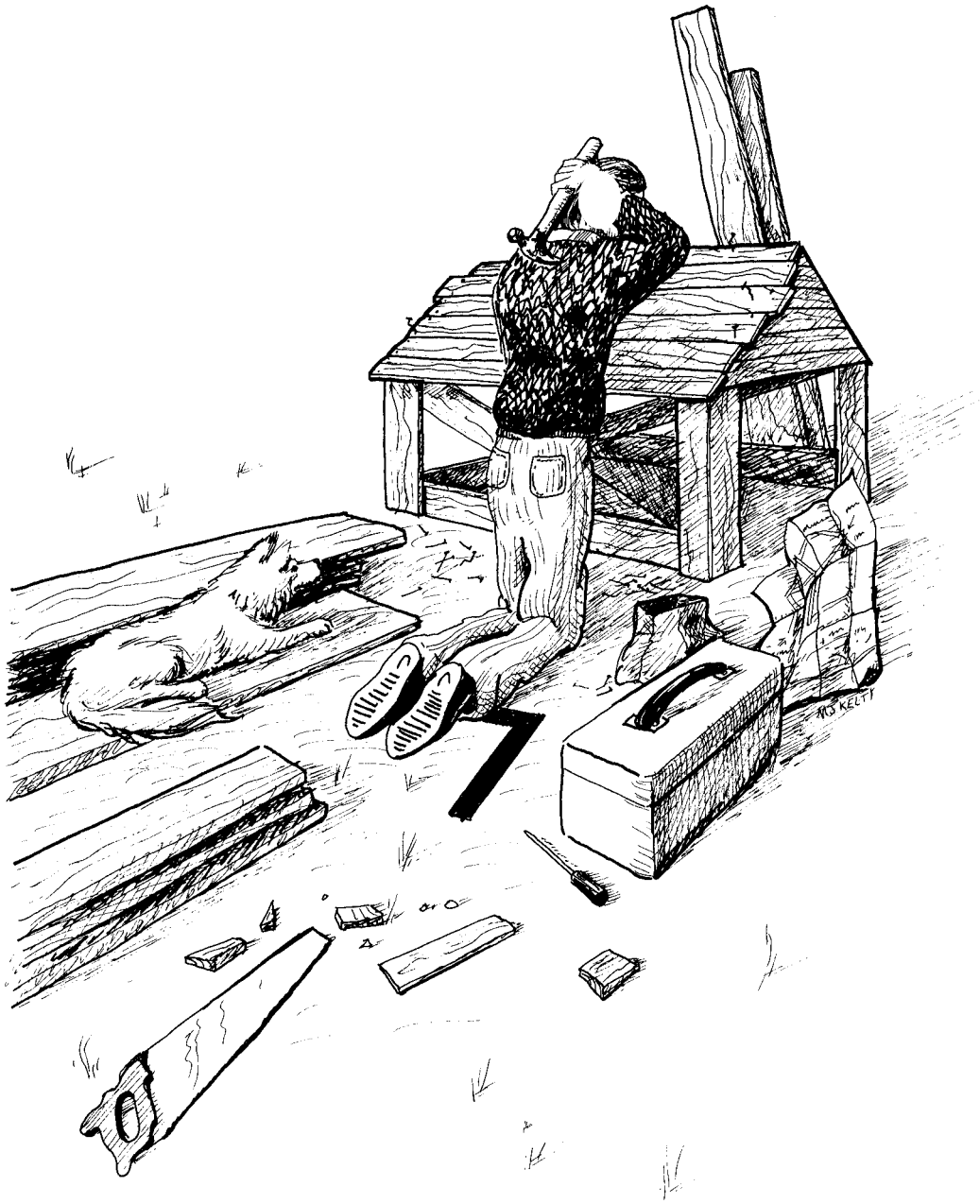
1. M. Eisenbud and T. Gesell, "Environmental Radioactivity from Natural, Industrial & Military Sources," 4th Edition, Academic Press, Orlando, FL, 1997.

2. "A Citizen's Guide to Radon," U.S. Environmental Protection Agency, EPA Document 402/K-09/001, Washington, D.C., 2009. Available free at www.epa.gov/radon/pdfs/citizensguide.pdf.

3. "Air Sampling in the Workplace," US NRC Regulatory Guide 8.25, Revision 1, Nuclear Regulatory Commission, Washington, DC, 1992.

Unit 3

Radiation Protection OPERATIONS



11

Protection Principles, Shielding and Transport

Outline of this Chapter

Chapter Summary

This is the first chapter of the final unit, Radiation Protection Operations. The focus shifts to those day-to-day tasks that make up much of the workload of the radiation protection technologist - conducting radiological surveys, managing rad waste, verifying compliance with dose limits, etc.

Chapter 11 is concerned with external and internal protection of persons from harmful radiation effects. This involves the control of exposure time, use of distance, shielding, protective clothing and respiratory protection.

Under U.S. federal regulations, establishment of explicit ALARA programs is mandatory. Various Regulatory Guides and expert reports are available to assist in this task. Time and distance control are proven techniques to reduce external exposures. Shielding is the third of the "big three" external control measures.

Gamma ray shielding thicknesses can be estimated by technologists by using the half- and tenth-value layer concept. For thicker shields, build-up of residual gamma ray energy must also be accounted for. Beta shielding is a two-step process. Consideration of the range is sufficient to stop the betas. However, since betas are charged, when stopping, they produce bremsstrahlung radiation. The second design step is to add material to shield the bremsstrahlung. Neutron shielding design is similar to gamma shielding. The HVL and TVL approach will produce the approximate thicknesses required. Shielding principles are illustrated by several applied examples.

Transport of radioactive packages involves a complex set of regulations. Decisions must be made as to form, quantity and package type needed. Then, appropriate labels and shipping names are chosen. The shipping papers must be completed correctly and finally, the package is tested for contamination before being turned over to the carrier.

Respirators are an effective technique for controlling internal exposures. Regulations impose maximum limits on the protection factor that can be claimed for various types of respirators. Availability of sufficient O_2 in the work environment is also a major factor in choice of equipment. Particular care must be taken in assuring a proper fit for any respiratory protective equipment. In addition to respirators, airborne contamination can also be handled by engineering controls such as a chemical fume hood or glove box.

Basic Principles

The ALARA Philosophy

The principle of "As Low As Reasonably Achievable" underlies all aspects of the various tasks which make up an external radiation protection program. The philosophy was first stated explicitly by the Nuclear Regulatory Commission (NRC) in 1971. The first Regulatory Guide to use the term was number 8.8 which was released in 1973. The statutory authority for the principle is contained in Part 20.1101 of Title 10 of the Code of Federal Regulations. This current version of the federal regulations requires that, "Each licensee shall develop, document, and implement a radiation protection program... to ensure that the occupational doses and doses to members of the public are as low as is reasonably achievable (ALARA)."

The underlying need for the ALARA principle is a result of the lack of a comprehensive biological model that could accurately predict the results of a dose of radiation to a person. In addition, more information is needed to make the "value judgement" as to the "cost to society" of the various forms of radiation injury. This factor would have to take into account the somewhat unique role of radiation as a causative agent in that the actual injury may be separated by many years from the "initiating dose" of radiation.

In theory, the radiobiological problems should be able to be put on a quantitative basis. At some future time when a comprehensive theory of radiation injury is accepted, the new theory will be able to give numerical answers as to the scope of those radiation induced injuries. However, in actual practice, the answer may not turn out to be that simple. As is discussed in Chapter 15, the present radiation protection standards and regulations are based on experiments conducted with high doses of radiation delivered at dose rates which are very high compared to virtually all occupational exposure conditions currently encountered in the practice of radiation protection technology. The effects which are seen under these "abnormal" conditions are then scaled down (extrapolated) to doses and rates which might reasonably be encountered in applied practice. In its Report Number 43, **Review of the Current State of Radiation Protection Philosophy,** the National Council on Radiation Protection and Measurements (NCRP) states, "All national and international groups which have studied the problems of quantitative carcinogenic risk estimates have regarded the practice of linear extrapolation as overestimating the risk .... The BEIR [Biological Effects of Ionizing Radiation Committee of the National Academy of Sciences] Report extrapolates by a factor greater than 1,000 in dose and by factors from 100 million to a billion in dose rate, from the level of observed effects to the levels encountered by the general population." An additional "complication" that calls straight-line extrapolations into question is the increasing acceptance of the concept of radiation hormesis, the idea discussed in Chapter 4 that, at certain doses, low level radiation may be beneficial to humans. As a result of all of this uncertainty, it is certainly prudent to keep radiation exposures ALARA until the final evidence is all in.

It should be noted that the ALARA principle applies BOTH TO INDIVIDUAL DOSES AND TO COLLECTIVE DOSES at a licensed facility. Soon

after the introduction of ALARA, some facilities realized that they could reduce the AVERAGE dose to their workers BY USING MORE WORKERS TO SPREAD THE DOSE AROUND. Carried to the extreme, radiation badges were issued to office workers, gate guards, cleaning personnel, etc. The net effect was that the total dose received by workers, measured in person-rem, could now be divided by a huge number of "workers" so the "average" dropped dramatically. The NRC subsequently clarified the fact that ALARA applies to the collective as well as individual doses at a facility. The collective dose is the sum of all doses received by workers and is reported in person-rem. If three workers stood in a one R/hr gamma ray field for one hour, the collective dose would be 3 person-rem. The average dose would be 1 rem for each.

NRC Regulatory Guide 8.10, **Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable,** states that "Two basic conditions are considered necessary in any program for keeping occupational exposures as far below the specified limits as is reasonably achievable. The management of the licensed facility should be committed to maintaining exposures as low as is reasonably achievable, and the personnel responsible for radiation protection should be continually vigilant for means to reduce exposures." The guide then discusses the specific responsibilities of management and the radiation protection staff. A summary of these areas of responsibility is given in Figure 1.

The management "commitment" must be in the form of a written policy statement. This normally is included in literature given to new employees. Item B, a formal exposure audit, is designed to locate the groups of workers or workplace locations with the highest radiation levels. Then, attention can be focused on reducing levels where "it does the most good." Formal training must be provided to each worker sufficient to meet their needs to safely deal with any radiation problems which they have

Management Commitments:

A. Plant personnel should be made aware of management's commitment to keep exposures ALARA

B. Management should periodically perform a formal audit to determine how exposures might be lowered

C. There must be a well supervised radiation protection capability with well defined responsibilities

D. Plant workers must receive sufficient training

E. The RSO should be given authority to enforce safe plant operation

F. Changes must be made where they will substantially reduce exposures at a reasonable cost

RSO and Staff Tasks:

A. They should know the origins of radiation exposures at the facility

B. They should continually seek ways to reduce exposures

C. They must see that equipment and supplies for radiation protection are provided to workers

Fig. 1 - NRC mandated ALARA responsibilities in the USA

the potential to encounter. This means that radiation safety training must be conducted at different levels for different workers. The mailroom clerk would receive much less radiation protection training than a technician on a process line handling hazardous unsealed sources. In addition to being required to appoint a "well qualified" Radiation Safety Officer, this RSO must also be given sufficient authority to prevent unsafe practices from occurring. If the RSO discovers a worker performing a task unsafely in the interest of production speed, it would be improper to require that the RSO draft a memo to the next radiation safety committee meeting for consideration of action. The RSO must have authority to shut down such procedures on the spot.

The last management item, Item F, raises an interesting question. What is the dollar cost of one person-rem of exposure? In other words, how much effort should be reasonably expended to reduce the collective dose at a licensee by one person-rem? This question has already been debated for years. It is clear that many members of the general public would place a much higher value for the cost of 1 person-rem than could be supported by the biological extrapolations of radiation risk. In international discussions of the issue, it has become clear that the "value of a human life" is quite different from country to country. This has resulted in the position by the ICRP that the value of a person-rem is a "national decision."

In the United States, the first guidance as to the cost of a person-rem came in 1976 in the form of an NRC Regulatory Guide for power reactors. The guide used a value of $1,000 per person-rem as a suggested figure and specified it was "measured" in "1975 dollars." As of 2011, the U.S. NRC accepts a value of $2,000 per person-rem. This value was published in their NUREG/BR-0058 document. A 1980 Department of Energy (DOE) report, **A Guide to Reducing Radiation Exposure to As Low As Reasonably Achievable (ALARA)** suggested, "If a dose reduction can be achieved at a cost of less than or equal to $2,000 per person-rem, then it is cost beneficial and should always be done." This figure was arrived at using an inflation factor on the $1,000 mentioned above. (The DOE figure is expressed in 1980 dollars). As of 2000, the applicable guide is DOE PNL-6577 which uses a figure of $10,000. In the nuclear power industry, an informal survey in 2000 showed a range of $2,500 to $25,000 per person-rem in use, with an average of $13,000. A cost-benefit analysis is done by calculating the collective dose reduction for one year at a facility if some proposed action (e.g., adding 5 cm of lead to a source vault) is taken. Next, the cost of making the change is calculated. The costs should include the capital cost of the construction or the new equipment (divided by the number of years of life of the improvement), annual maintenance costs and costs associated with the "downtime" at the facility during the construction phase. If the change costs less than the $/person-rem value used, the change must be made. See Sample Problem 1.

ALARA Program Planning

In planning an ALARA program at a facility, it is useful to follow the recommendations of the ICRP and the NCRP concerning reference ranges. The basic idea is that certain levels of exposure should trigger an investigation with the objective of reducing doses whenever reasonable. The Individual Reference Range, or IRR, is a numerical range of TEDE values that automatically results in an investigation if exceeded. To prevent excessive investigations, it should be set specifically for a given facility using

Sample Problem 1

GIVEN:

A ventilation system would lower the TEDE by 20 mSv annually per worker at a power reactor radioactive gas handling facility with 7 radiation workers.

FIND:

What annual cost over the life of the ventilation system would be reasonable?

SOLUTION:

ALARA considerations for this nuclear plant dictate $5,000 per person-rem annually. The reduced collective dose = Annual Dose x # of Workers = 20 mSv x 7 persons = 140 person-mSv x 0.1 rem/mSv = 14 person-rem. Using the above cost guideline, the ventilation system should be installed if it can be built for $5,000/person-rem x 14 person-rem = $70,000 annualized cost.

historical dose records for guidance. It is given both an upper and lower limit to allow a facility some latitude in judgement. Individual doses falling within the IRR (i.e., above the lower limit but less than the upper limit) are left to the discretion of the radiation safety authorities at the facility in terms of investigating or not.

Since ALARA programs must deal not only with individual doses but with the collective dose as well, the second reference range is the Collective Reference Range, or CRR. When the facility collective dose for a review period exceeds the CRR, then steps must be taken to assure that this dose is reasonable. The recommended review period for most facilities is quarterly. If individual or collective doses are compared to the reference ranges for periods less than 3 months at a time, the normal variations associated with radiation doses in the real workplace would trigger too many investigations. Similarly, if analysis is not made at least every 3 months, a problem with excessive doses to an individual will not be caught soon enough that corrective action can be taken.

The NCRP recommends that the upper limit for the IRR be above the dose of 99% of potentially exposed workers and above 95% of all doses reported above the detection limit by the badge processor The lower limit of the range for the IRR should be the dose which is exceeded only by 20% of the workers who actually receive a measured dose. In setting the range for the CRR, the average collective dose for recent review periods should be calculated. Then, the lower limit for the CRR is usually one standard deviation above the average, and the upper limit of the CRR is set at two standard deviations above the average.

Control of Exposure Time

Time is the first of the "big three" exposure-reducing tools available to the practicing radiation protection technologist. When time, distance and shielding are used effectively, adequate external protection can always be provided to radiation workers and members of the general public. The technologist should be aware that radiation doses are directly proportional to the time spent in the field. If the time spent in a

given radiation field is doubled, the worker's dose is doubled. Therefore, to limit doses, the time spent in the field must be limited.

It should be emphasized again that, although the concept is simple, IT WORKS. The control of time is the principle behind the radiation work permit which is commonly used at nuclear power stations. Figure 2, shown below, is a sample of such a permit. After a description of the job to be performed, and a listing of measured radiation levels at the job site, item K specifies the person-rem collective dose

San Onofre Nuclear Generating Station

Radiation Protection Procedure S-VII-9.9
CN05CA-275-RA

№ 40477

RADIATION EXPOSURE PERMIT

OFFICIAL USE ONLY
REPCODE
JLC
JDC
WORK
SYST
COMP

A Initiating Supervisor _____ Date _____

B Expiration Time (24 Hour Time) ____ Expiration Date MO DA YR

C Job Location _____

D Job Description _____

E System _____ Component _____

F Personnel to Perform Work:

	NAME CODE	NAME	INITIALS	PARTICIPANT NUMBER	QUARTERLY DOSE TO DATE	QUARTERLY DOSE REMAINING
1						
2						
3						
4						
5						
6						
7						
8						

ENTERED ON LINE BY

G Escort(s) (Line Numbers Above) _____

To Be Completed By Health Physics Personnel

H Protective Clothing Dosimetry and Respiratory Equipment Personnel

PROTECTIVE CLOTHING		DOSIMETERS		RESPIRATORY EQUIPMENT
☐ *Dry Work Clothes	☐ Plastic Rainsuit	☐ Neutron ☐ 200 mR		☐ Not Required
☐ Coveralls over Street Clothes	☐ Rubber Boots	☐ Wrist ☐ 500 mR		☐ Air Sample Required
☐ Lab Coat	☐ Face Shield/Beta Goggles	☐ Finger ☐ 1000 mR		☐ Full Face Particulate
☐ Gloves	☐ Steam Generator Entry Kit	☐ Head ☐ 5000 mR		☐ Powered Air Purifying
☐ Plastic Booties, Rubber Shoe Covers	☐ Head Covering	☒ Chest Badge		☐ Supplied Air
☐ Paper Coveralls	☐ Modesty Garment	☐ Steam Generator Packet		☐ Self-Contained Air
*Dry Work Clothes: Red underwear and socks, head gear, coveralls, cotton and plastic gloves, plastic booties, and rubber shoe covers.	☐ Canvas Shoe Covers	☐ Additional Dosimetry—Specify:		

SPECIAL INSTRUCTIONS

I Health Physics Coverage ☐ INITIAL ☐ ON CALL ☐ CONTINUOUS ☐ SELF

J

GENERAL AREA RADIATION LEVEL	GENERAL CONTAMINATION LEVEL	AIRBORNE MPC'S
mrem/hr	DPM/100 cm^2	
HOT SPOT RADIATION LEVEL	MAXIMUM CONTAMINATION LEVEL	
mrem/hr	DPM/100 cm^2	

K Person-Rem Limit _____

REMARKS

DISTRIBUTION:
Original – Health Physics
Canary – Watch Engineer (Control Point Monitor)
Pink – Exclusion

HEALTH PHYSICS SUPERVISOR

UNIT SUPERINTENDENT (FOR ENTRY WITHIN CONTAINMENT SECONDARY WHILE REACTOR CRITICAL)

SCE SO(1) 21 REV. 2 01/22/81

Fig. 2 - A sample radiation work permit

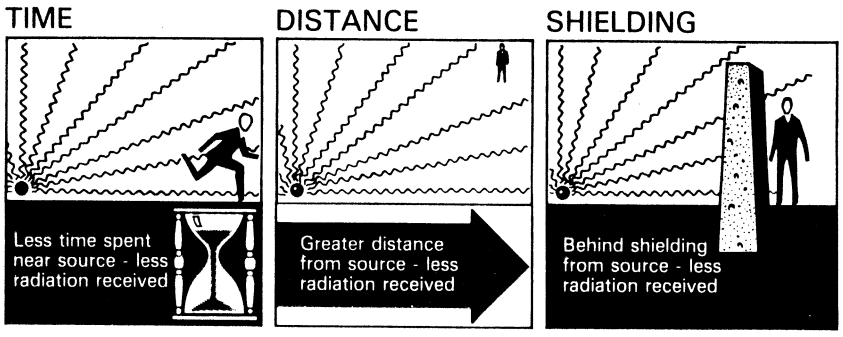
Courtesy of Southern California Edison Co.

limit for the task. The escort radiation protection technicians can then estimate the maximum stay time at the job location so as not to exceed the collective dose limit. Stopwatches and walkie-talkie radios are frequently used in such situations. One company even markets a set of radio transmitting dosimeters that broadcast the cumulative dose reading for each individual of a work party to a central control console which alarms as each person reaches a pre-set dose value.

One other comment is in order. As we will see in Chapter 15, the dose limits for members of the public include a 2 millirem "in any one hour" restriction. This does NOT mean that a survey meter reading of, say, 50 millirem/hr in a public area is necessarily illegal! Note carefully the regulation wording "in any one hour." The idea here is that you should look at the average dose rate, over a one hour period, not the instantaneous dose rate. For example, the 50 mrem/hr may have been the maximum measured during a medical procedure which takes 1 minute to perform but 30 minutes to get set up. Thus, the "excessive" dose rate could be present only for 2 minutes out of "any one hour" so, the average rate would be 1.7 mrem/hr which does meet the regulatory limit.

Exposure Control Through Distance

Distance is the second of the "big three." In many cases, it is more important to control distance than time in a radiation field. For sources which can be treated as approximating a point (a radiation source in which you are at a distance at least three times the longest dimension from it), the radiation dose received is inversely proportional to the square of the distance of separation. The effects of time and distance on dose are compared in Figure 3. Notice that the time dependence follows a first power law while the distance dependence follows an inverse square (second power) law. Thus, the distance of separation between the person and the "point" source has a



$$\text{Radiation Dose} \propto (\text{Time in Field})^1$$

$$\text{Radiation Dose} \propto \frac{1}{(\text{Distance of Separation})^2}$$

Fig. 3 - Use of time, distance and shielding to control dose

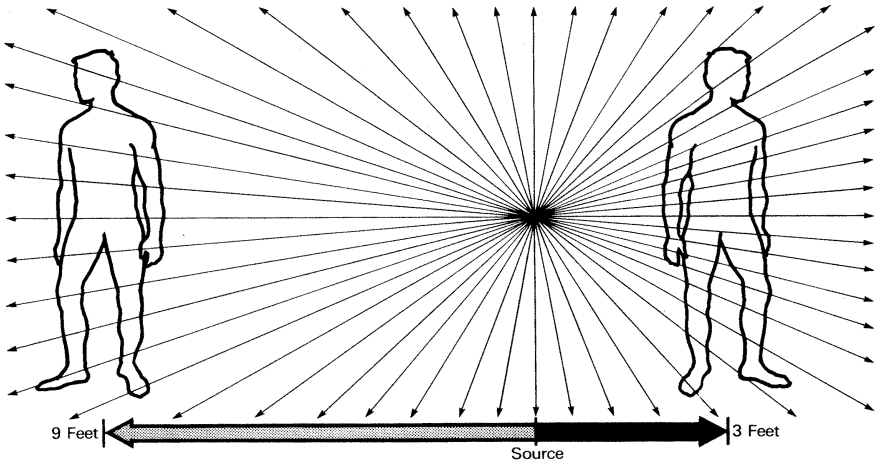


Fig. 4 - The effect of distance on dose

relatively greater influence on dose than the time factor. (See Figure 4). The mathematical relationship is illustrated in the example in Sample Problem 2. Assume that a replacement gamma ray source has arrived for a cobalt-60 irradiator. Following good ALARA practice, the radiation protection staff conducts some simple measurements with a stopwatch and a non-radioactive mockup of the source to find the best way to transfer the new source to the irradiator. It is determined that a technologist using a 1 meter handling tool is exposed to the unshielded source for about 5 seconds during transfer. Similarly, use of a 2 meter long handling tool extends the exposure time to 10 seconds. Using the two techniques, the relative doses are as shown in the Problem.

Note that the dose received by the technologist is half as much using the long tool rather than the short one, even though it takes twice as long to complete the job. Because of the exponent "2" in the distance formula compared to the exponent of only "1" in the time formula, our "common sense" tends to be fooled into giving more equal weight to the two factors instead of correctly realizing that distance is much more important than time in exposure control situations.

Before leaving the distance factor, it should be recognized (as first mentioned in Chapter 5) that there are frequently monitoring tasks in which the source clearly cannot be treated as a "point." Measuring the surface dose rate in contact with a 55 gallon rad waste shipping barrel would be one example where the large volume,

Sample Problem 2

GIVEN:

A source is exposed for 5 seconds at 1 meter or 10 seconds at 2 meters.

FIND:

Which condition gives the lowest dose?

SOLUTION:

Dose $\propto$ (Time)^1 ÷ (Distance)^2. Thus, for the 1 meter tool, Dose $\propto$ (5 sec)^1 ÷ (1 meter)^2 = 5/1^2 = 5 "units." For the 2 meter tool, Dose $\propto$ (10 sec)^1 ÷ (2 meters)^2 = 10/2^2 = 10/4 = 2.5 "units." Clearly, the 2 meter tool is the one of choice.

compared to the measurement distance, would have to be taken into account. A second example would be measuring the dose rate in a small compartment in a nuclear submarine where the source is a reactor coolant pipe that runs through the compartment. In this situation, it is impossible to get "three times the longest dimension of the source" away from the pipe. By use of the calculus, exact answers are available by which the dose rate at a point in space can be computed for virtually any size and shape of a radioactive source. Personal computer programs are available to do these calculations. In practical radiation protection technology exact answers seldom are needed (after all, how good is the calibration on your survey instrument?).

The following rules of thumb can be used to make an educated guess as to the dose rate behavior versus distance:

LINE SOURCE - The dose rate falls off with distance approximately as 1/R (doubling the distance halves the dose rate).

DISK SOURCE AND CYLINDRICAL SOURCE - The dose rate falls off a little faster than 1/R but not as fast as $1/R^2$ (doubling the distance drops the dose rate below a half but not down to a fourth).

One final example on the use of distance is related to medical uses of radioisotopes. The allowed annual dose, under 10 CFR 20 regulations, for persons not occupationally exposed is 100 mrem. Visitors to radioactive patients who are undergoing therapeutic treatment with internally administered radioisotopes or sealed sources in applicators (as described in Chapter 6), might potentially exceed the annual limit. To deal with such cases, NCRP Report 155, **Management of Radionuclide Therapy Patients,** was issued in 2006. Relative to the question of distance, the NCRP suggests, "As far as visitors are concerned, there is little likelihood of their exceeding this dose … if they remain about 6 feet or more from the patient, except for a brief period to shake hands, deliver mail, etc. In general, pregnant women and children should not be allowed to visit patients having an appreciable radioactive burden."

Shielding Design

Gamma Ray Shielding

The third factor of the "big three" for controlling external exposure is shielding. In this section, design principles for shielding gamma ray sources will be covered. The next two sections deal with beta and neutron shielding respectively.

A simple, yet effective way for calculating the needed thickness of some shielding material is to use the half-value layer and tenth-value layer concepts. These two terms are defined in Figure 5. Note the use of the preposition TO instead of BY. The half- and tenth-value layers reduce TO one half and TO one tenth respectively.

Half-Value Layer = HVL = Thickness to reduce exposure rate TO one half

Tenth-Value Layer = TVL = Thickness to reduce exposure rate TO one tenth

Fig. 5 - Definitions of the HVL and TVL

Sample Problem 3

GIVEN:
A HVL reduces to 1/2 and a TVL to 1/10
FIND:
How many HVL = one TVL?
SOLUTION:
Stated in equation form, the problem becomes: $(1/2)^x = (1/10)$, find x. Taking the log of both sides, $\log (0.5^x) = x \log 0.5 = \log 0.1$ or $x = (\log 0.1) / (\log 0.5)$. So, finally, $x = (-1) / (-0.301) = 3.32$ HVL in a TVL.

(Reduction BY one tenth leaves 90% of the exposure rate present.) Thus, a TVL is thicker than an HVL. Under ideal conditions it is possible to calculate the exact relationship. Two HVL will reduce to 1/4 (1/2 x 1/2) and three to 1/8 (1/2 x 1/2 x 1/2) but 4 would reduce below a tenth. So, there are between three and four HVLs in a TVL. Mathematically, the number of HVL needed to equal a TVL is the power to which 2 must be raised to equal 10. The correct solution is as shown in Sample Problem 3. Therefore, it takes about 3 1/3 HVL to equal a TVL.

Note, however, that this mathematical truth is not necessarily workable in operational radiation protection. If the gamma ray source has more than one energy present, MORE THAN 3 1/3 HVL will be needed to reduce the exposure rate to 1/10 as the lower energies will be more effectively removed than the higher due to the strong energy dependence of the photoelectric cross section (thus the gamma ray beam gets progressively "harder" [has a higher AVERAGE beam energy] as the beam penetrates through the shielding). In addition, this simple calculation ignores the fact that Compton scattered gamma rays are still present in the shield in the form of lowered energy photons. Still, as a first approximation, it is useful to consider 3 1/3 HVL = 1 TVL.

A table of some measured HVL and TVL values for various shield materials is shown in Figure 6. The values for x-ray spectra given here are for situations in which virtually all the lower energies have been filtered out already.

Radiation	Half-Value Layer (cm)			Tenth-Value Layer (cm)		
	Lead	Iron	Concrete	Lead	Iron	Concrete
125 kVp x-rays	0.027	0.09	1.6	0.088	0.3	5.3
6 MeV x-rays	1.7	3.0	10.4	5.6	9.9	34.5
Cesium-137	0.65	1.6	4.8	2.1	5.3	15.7
Radium-226	1.66	2.2	6.9	5.5	7.4	23.4
Cobalt-60	1.2	2.1	6.2	4.0	6.9	20.6

Fig. 6 - A table of HVL and TVL values for THIN shields

The basic calculational approach to gamma shielding is to determine (by measuring it) the existing exposure rate, decide on the desired exposure rate after shielding, and then calculate how many HVL or TVL will be needed. The basic working equation which relates the desired and measured exposure rates (shielded and unshielded) is given below. All this equation does is to express mathematically the fact that each HVL reduces the exposure rate to 1/2 the unshielded value and each TVL reduces it to 1/10 the unshielded value.

$$X/t_{shielded} = R_{shielded} = (R_{bare}) \times (1/2)^{\#HVL} \times (1/10)^{\#TVL} \qquad \text{[Eqn. 1]}$$

where #HVL = Shield thickness (cm) / HVL (cm) and

#TVL = Shield thickness (cm) / TVL (cm).

Sample Problem 4 shows how the equation is used in practice. It is necessary to know how many hours per week a radiation worker could stand about one meter in front of a workbench holding a 2 Ci Cs-137 source and a 700 mCi Co-60 source behind a "wall" of 2 inch lead bricks.

Remember that anything raised to the "0th" power = 1. This problem could also have been worked by calculating the number of TVLs needed. The power function calculation is done using the (Y^X) button on a scientific calculator. Any remaining confusion over these calculations should be remedied by a review of Chapter 5.

The HVL and TVL approach works well for routine operational questions like "How many bricks are needed here?" However, if the sources are of high activity and the shielding is thick, then problems occur. This is a direct result of the fact, which

Sample Problem 4

GIVEN:

A work station is located 1 meter from a 2 Ci source of ^{137}Cs and a 0.7 Ci source of ^{60}Co shielded by 2 inches of lead.

FIND:

How many hours per week would deliver 5 rem annual exposure to the worker?

SOLUTION:

From Chap. 5, Fig. 15, for Cs-137, R_{bare} (R/hr) = 0.5 A E/r^2 = (0.5 x 2 x 0.662 x 85%) / 1^2 = 0.56 R/hr. The 0.662 and 85% come from Appendix A-1. Next calculate the 2" lead wall equivalence in HVL:

#HVL = (2" x 2.54 cm/") / 0.65 cm/HVL = 7.8 HVL, using Fig. 6 HVL data. Equation 1 now gives $R_{shielded}$ = 0.56 R/hr x $(1/2)^{7.8}$ x $(1/10)^0$ = 0.56 R/hr x 0.0045 x 1 = 0.0025 R/hr.

Similarly, for Co-60, R_{bare} (R/hr) = 0.5 A E/r^2 = (0.5 x 0.7 x 2.5) / 1^2 = 0.875 R/hr, and #HVL = (2" x 2.54 cm/") / 1.2 cm/HVL = 4.2 HVL, using Fig. 7 HVL data. So, $R_{shielded}$ = 0.875 R/hr x $(1/2)^{4.2}$ x $(1/10)^0$ = 0.875 R/hr x 0.054 x 1 = 0.048 R/hr. Since both sources are present behind the lead bricks, the exposure rate at the work location is the sum, 0.0025 + 0.048 R/hr = 0.0505 R/hr.

Assuming 50 work weeks per year, the average rate to deliver 5 rem is 5 rem/50 wks = 0.1 rem/wk. The worker could thus stand there for 0.1 R/wk / 0.0505 R/hr = 2 hours/week.

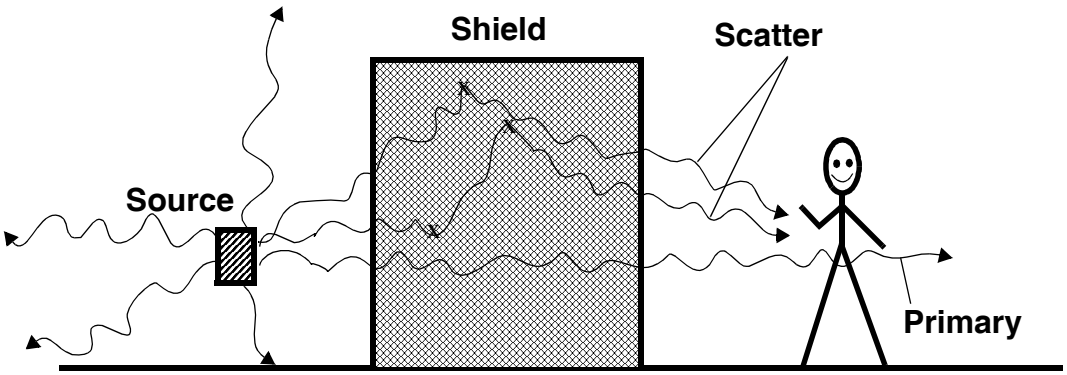


Fig. 7 - Build-up of scattered radiation in a thick shield

was emphasized in Chapter 3, that Compton scattering and pair production do not remove all the photon energy from the beam. The residual, lower energy Compton photons and the annihilation gamma rays are still transporting energy through the shield. In the case of a thick shield, these stray photons can interact a second time and scatter in a different direction to produce a higher exposure rate outside the shield than that produced by the primary transmitted beam. (See Figure 7.) The thicker and taller the shield, the larger the build-up of this scatter component. In addition, since Compton scatter and pair production are only likely for medium and high energy gamma rays, respectively, the energy of the photons from the source will affect the amount of the scatter contribution to the exposure rate. Finally, the interaction coefficients for the three photon processes are dependent on the absorber Z. Thus, the shield material influences the amount of scatter which builds up. This situation is solved by introduction of a "build-up factor" into the gamma ray attenuation equation.

For a thin shield, the intensity of the gamma ray beam that penetrates a given shield is given by Equation 2:

$$R_x = R_0 \, e^{-\mu x} \qquad \textbf{[Eqn. 2]}$$

where R_x = exposure rate behind shield of thickness "x"
R_0 = bare exposure rate, i.e., no shield
x = shield thickness and
μ = total linear attenuation coefficient.

This equation just states that gamma rays are exponentially attenuated by the shield. However, the derivation of this equation was based on the assumption that any gamma ray which interacted in the shield was removed by that interaction. From Chapter 3 we know that, strictly speaking, this works only for photons that interact by the Photoelectric Effect. There is residual photon energy following both a Compton and a pair production interaction. If the shield is thin enough, we can ignore these residual photons. In the case of a thick shield, of relatively high Z and density, with a gamma ray source of medium to high energy, the simple exponential law of Equation 2 greatly underestimates the actual exposure rate outside the shield. As previously mentioned, we insert a build-up factor, B, as shown by Equation 3:

$$R_x = B \, R_0 \, e^{-\mu x} \qquad \textbf{[Eqn. 3]}$$

where B = Build-up Factor for shield.

Graphs of B are given in various standard reference books for various shield materials. These graphs are consulted to find B for a specified configuration and then Equation 3 is used to predict the exposure rate outside the shield. To cite a couple of numerical examples, a point cobalt-60 source inside 30 cm (one foot) of lead shielding would have a build-up factor of about 7. (The linear attenuation coefficient, μ, is 0.7/cm). A point cesium-137 gamma ray source shielded by 10 cm of water would have a B of about 2 (the linear attenuation coefficient for Cs-137 gamma rays in water is 0.085 per cm.) See Sample Problem 5.

Sample Problem 5

GIVEN:

A Co-60 source is shielded by 30 cm of lead. The expected exposure rate in contact with the shield is 0.027 mSv/hr based on pure exponential absorption.

FIND:

What actual exposure rate would be expected in this case?

SOLUTION:

To include the effect of build-up, the exposure rate should have been calculated with Eqn. 3 instead of Eqn. 2. Comparing the two equations, the only difference with a "thick" shield is the multiplicative factor B. Thus, the actual exposure rate would be expected to be "B" times higher than predicted by Eqn. 2 or in other words, 7 x 0.027 mSv = 0.19 mSv. The build-up of "7" came from the text just above.

One final problem, in gamma ray shielding design, should be pointed out. Although room air isn't normally thought of as providing significant shielding for

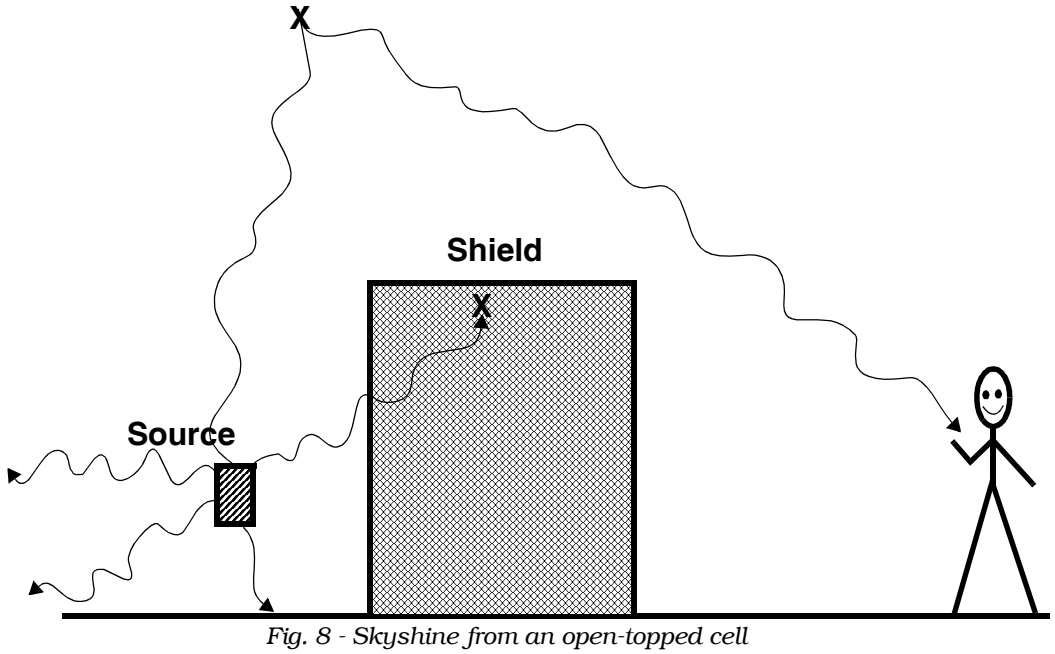


Fig. 8 - Skyshine from an open-topped cell

gamma ray sources, the presence of air does provide atoms which can Compton scatter gamma rays. Thus, the gamma rays appear to "turn corners." Figure 8 illustrates the particular problem of "skyshine." The name was coined to reflect the fact that gamma rays appear to shine down from the sky if adequate shielding is not placed above the source. For obvious practical reasons, the open-topped cell is a common design. However, care must be taken if high activity sources are used inside, because skyshine can produce a significant radiation field outside the cell wall. To illustrate the magnitude of the problem, a 50 Ci cobalt-60 source behind a 12 foot high concrete wall with no top will produce a skyshine level of about 17 mR/hr three feet outside the wall.

Beta Ray Shielding

Since beta particles are charged particles, they have a definite, predictable range beyond which they will not travel. Then, in principle, if a thickness of shielding greater than or equal to their range is placed in their path, 100% of the betas will be stopped. In practice, the betas will be stopped, but they will produce bremsstrahlung radiation as they slow down and come to rest. Thus, the relatively non-penetrating beta rays "turn into" a penetrating photon source.

As mentioned in Chapter 3, the intensity of the bremsstrahlung radiation is proportional to the number of betas, their energy and the atomic number, Z, of the absorber (shield). Equation 4 is often used for design of beta shielding to take into account the fraction f of beta energy that is converted to bremsstrahlung:

$$f = k \times Z \times E_{max} \qquad \textbf{[Eqn. 4]}$$

where k = 3.5×10^{-4} (low Z shield) or 5×10^{-4} (high Z)

Z = Atomic number of the shield material

E_{max} = Maximum energy of isotopic beta emitter, in MeV

The proportionality constant, k, turns out to depend somewhat on both the beta energy and the absorber Z number. Two values for "k" are recommended here to give a conservative answer for isotopic beta sources interacting in low Z (e.g., tissue) and high Z (e.g., lead) absorbers.

In designing a shield for a beta source, a material is selected that will minimize bremsstrahlung production. From Equation 4, this is seen to require a material with a low Z. Common beta shields use plastics which have rather low effective Z due to the high carbon and hydrogen content. Next, the range of the highest energy beta is determined from a standard beta Range–Energy curve (see Figure 9), and this value becomes the desired shield thickness. (Note: The concept of measuring range in density thickness was introduced in Chapter 5.) Then Equation 4 is used to estimate the bremsstrahlung production in this first layer of the shield. Since the "f" represents the fraction of emitted beta energy that appears as photon energy, we can "replace" the beta source with a hypothetical gamma ray source. The activity of this "effective" source is "f" times the activity of the beta source ($A_\gamma = A_\beta \times f$). A second shield layer is then designed to reduce the exposure rate to desired levels. A theoretical examination of bremsstrahlung produced from isotopic beta sources shows that the average energy of the bremsstrahlung is about $1/4 \times E_{max}$. This whole process is illustrated

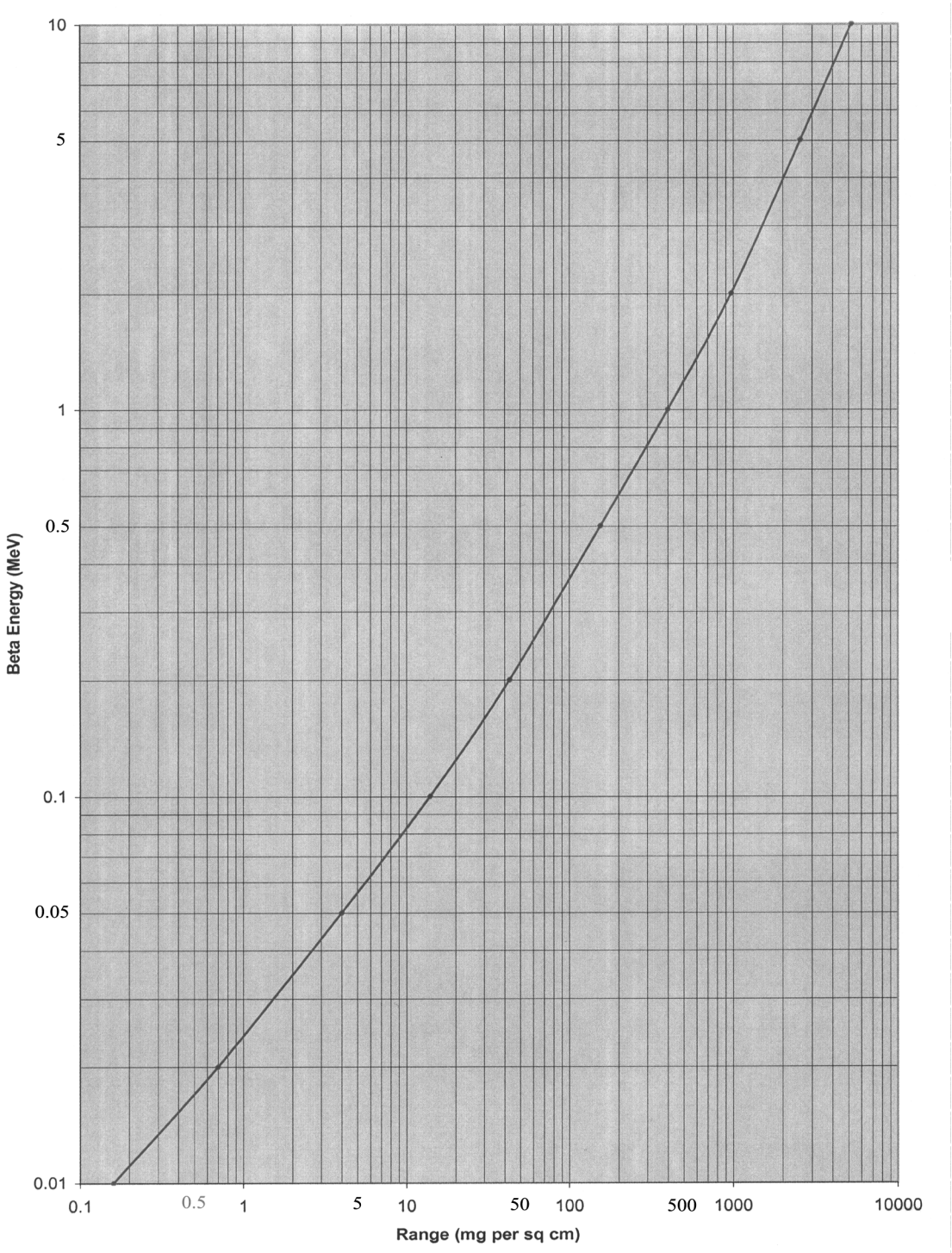


Fig. 9 - A beta ray Range-Energy curve

Sample Problem 6

GIVEN:
A 90 GBq beta source with E_{max} = 1.2 MeV requires shielding.
FIND:
What shield will reduce the external dose rate to 0.02 mSv/hr @ 50 cm?
SOLUTION:
Choose acrylic plastic (Z_{eff} = 5.85) as the inner layer. Density = 1180 mg/cm^3. From Fig. 9, the range of a 1.2 MeV beta is 500 mg/cm^2. The required thickness for the inner first layer is thus t (cm) = Range (mg/cm^2) / Density (mg/cm^3) = 500 mg/cm^2 / 1180 mg/cm^3 = 0.42 cm of acrylic plastic.
Next, the fraction of energy appearing as bremsstrahlung is calculated from Eqn. 4: f = 3.5 x 10^{-4} Z E_{max} = 3.5 x 10^{-4} x 5.85 x 1.2 = 2.5 x 10^{-3} or 0.25%.
The effective gamma ray source activity is A_{gamma} (GBq) = A_{beta} (GBq) x f = 90 GBq x 2.5 x 10^{-3} = 0.22 GBq x 1 TBq/1000 GBq = 2.2 x 10^{-4} TBq. From Chap. 5, Fig. 15, the bare dose rate from this activity gamma ray source is H/t_{bare} = 0.15 A E / r^2 = (0.15 x 2.2 x 10^{-4} TBq x 1/4 x 1.2 MeV) / (0.5 m)^2 = 4.0 x 10^{-5} Sv/hr = 0.040 mSv/hr. To reduce this to 0.02 mSv/hr (2 mrem/hr) use one HVL worth of gamma ray shielding.

by Sample Problem 6. The exposure rate for the "effective" gamma source is calculated from the "0.5 A E rule" or the "0.15 A E rule" from Chapter 5 by substituting 0.25 x E_{max} of the beta source for the "E."

Note that the low Z shield MUST be inside (next to the source) and the high Z shield must be on the outside. If these layers are reversed so that the betas are stopped in the high Z layer, the bremsstrahlung problem is much worse. To see this effect, the exposure rate for the Sample Problem 6 source is recalculated in Sample Prob. 7 for the layers reversed. Note that the high Z value for "f" is used this time for the lead rather than the low Z value that was used in the previous sample problem.

A practical application of this principle of use of low Z beta shields is the use of polyethylene bottles (effective Z of about 5.4) in place of glass (effective Z of 11.6) when storing solutions of beta emitters. This simple rule will reduce the bremsstrahlung production by the ratio of the atomic numbers, i.e., by over 2 times.

Neutron Shielding

For the purposes of radiation protection technology, we do not need to deal with neutron shielding at great depth. Still, the technologist should have an idea of the types of materials and thicknesses required to reduce neutron radiation fields.

An appreciation of the amount of shielding needed for fast neutrons is gained by use of the half-value layer and tenth-value layer concepts. Figure 10 lists values for the HVL and TVL of ordinary concrete used to reduce the dose equivalent rate of

Sample Problem 7

GIVEN:
The inner plastic shield in Sample Problem 6 is replaced with lead.
FIND:
What effect will this have on the unshielded gamma ray dose rate?
SOLUTION:
If the betas strike lead, the bremsstrahlung fractional production will be $f = 5 \times 10^{-4} \ Z \ E_{max} = 5 \times 10^{-4} \times 82 \times 1.2 = 5 \times 10^{-2}$ or 5%.
The effective gamma ray activity in this case would be $A_{gamma} = A_{beta} \times f = 90$ GBq $\times 5 \times 10^{-2} = 4.4$ GBq X 1 TBq/1,000 GBq $= 4.4 \times 10^{-3}$ TBq.
The bare, unshielded dose rate with a lead inner shield would be $H/t_{bare} = (0.15 \times 0.0044$ TBq $\times 0.25 \times 1.2$ MeV) / (0.5 meter)$^2 = 8 \times 10^{-4}$ Sv/hr or 0.8 mSv/hr (80 millirem/hr)!! I t would now take a TVL plus 2 HVLs to reduce the gamma ray level to the 0.02 mSv/hr design specification.

fast neutrons. About 1 meter of concrete would be needed to reduce a 14 MeV neutron field by a factor of 100 (2 TVLs). As an additional aid, Figure 11 shows the concrete equivalence of three other common shield materials. These values hold for neutrons from about 0.5 to 1 MeV only.

Neutron Energy (MeV)	HVL (cm)	TVL (cm)
1	6.8	22.5
5	11	38
10	14	38
15	16	53

Fig. 10 - HVL and TVL values for neutrons in concrete

One final point of caution. In the capture of a neutron by a hydrogen nucleus, a capture gamma ray of 2.23 MeV is released by each interaction. The resulting gamma field may require additional shielding to reach an acceptable dose rate.

The highest energy neutrons, sometimes called relativistic neutrons, are shielded using a three-layer configuration. A material such as iron is chosen for its high inelastic scattering probability (see Chap. 3, Fig. 2). This material is positioned to be the first layer. The inelastic scattering

1 cm of concrete = 0.55 cm polyethylene
1 cm of concrete = 0.63 cm water
1 cm of concrete = 1.3 cm water saturated soil

Fig. 11 - Concrete equivalence for fast neutrons

interactions remove energy rapidly from the neutrons and cause their energy to drop into the fast neutron classification. The next layer is chosen to moderate and thermalize these fast neutrons. As discussed in Chapter 3, materials containing hydrogen are best for this purpose as a maximum energy transfer occurs during elastic collisions between objects of the same mass. Finally, a third shield layer is added to reduce the flux of the resulting thermal neutrons to a reasonable level. Several elements in the periodic table have high capture cross sections for slow neutrons. Cadmium, boron and lithium have commonly been used. Relatively small thicknesses are required to cause a significant flux reduction. Two millimeters of boron will reduce the thermal neutron flux by over 4,000 times.

Applied Shielding Examples

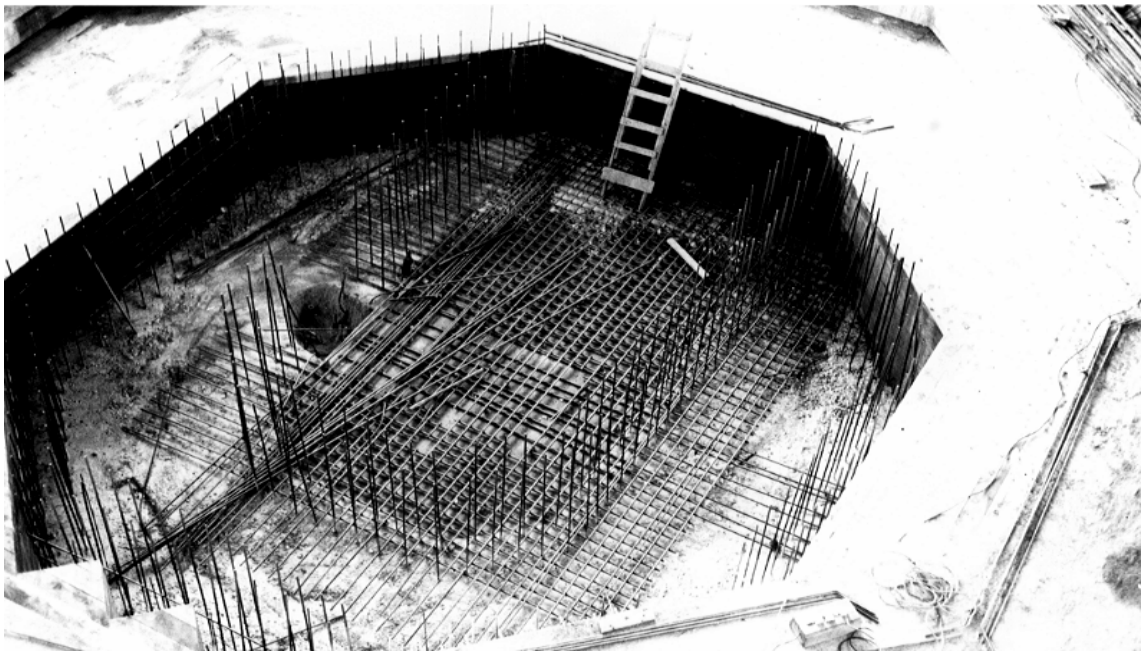
This section of the chapter will discuss representative shielding problems encountered in a sampling of different radiation facilities. The examples chosen illustrate only some of the principles discussed above and are not intended to be the "last word" in shielding theory applied to these installations.

Nuclear Reactors

Both research reactors and power reactors are intense sources of neutron and gamma ray fields during their operation. In addition, they contain a large inventory of beta-gamma emitters within the fuel assembly which must be considered when the reactor is shut down.

The neutrons emitted during nuclear fission range in energy from almost zero up to about 10 MeV. The energy which occurs most often is around 0.8 MeV while the AVERAGE for the fission spectrum is 2.5 MeV. Thus, fast neutrons are shielded most effectively by materials containing hydrogen. (Remember that this average fission neutron energy is for a BARE unshielded reactor. Shielded power and research reactor neutrons have average energies between 50 keV and 250 keV.) Commonly used materials for the main biological shield at a reactor include concrete, heavy concrete (which is loaded with iron pellets or metal ores to increase the density), water (e.g., the swimming pool reactor) and earth. Ordinary concrete is usually the cheapest to install. Heavy concrete requires additional care at the time of installation to insure that the heavy additives are properly mixed in and do not settle during curing. Also, the forms require additional structural support to be able to hold the extra weight without sagging. The advantage is reduced thickness. This is an important consideration when space limitations are present. Approximately 25% less thickness is required for heavy concrete (235 pounds/cubic foot) to give the same neutron attenuation as ordinary concrete (147 pounds/cubic foot). Figures 12 through 14 show preparations to pour heavy concrete shielding for a 0.5 MW research reactor.

Concrete is also effective against the gamma rays emitted by the fission products and the "prompt" gamma rays given off in the fission decay. About 70% of the fission product gamma rays are emitted within



Gollnick

Fig. 12 - Cavity that will hold the pressure vessel of a 0.5 MW reactor

the first 15 minutes after shutdown of the reactor. The prompt gamma rays range in energy from 10 keV to 10 MeV. These gammas have an average of 0.9 MeV. About 8 photons are emitted per U-235 fission. For this

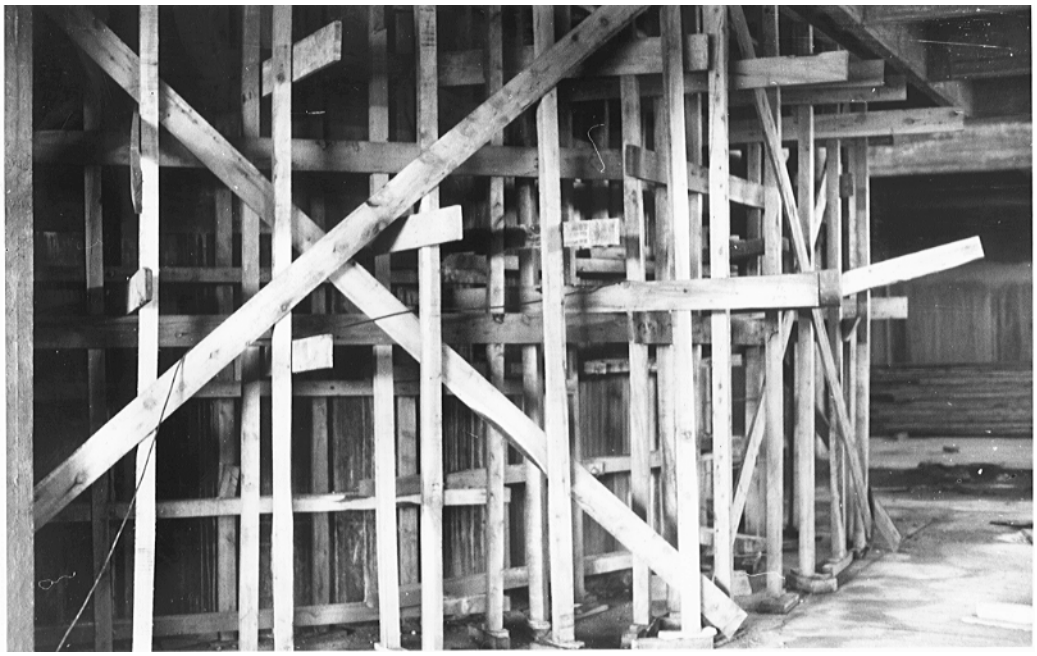


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Fig. 13 - Shoring needed to contain the high density concrete shielding

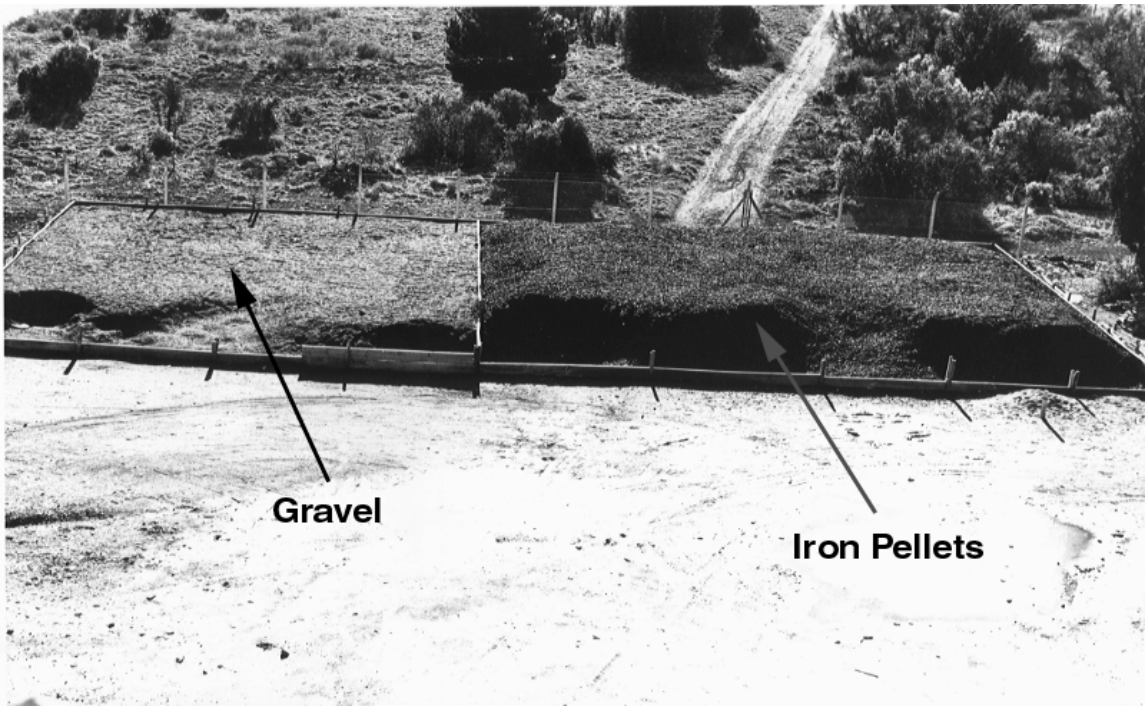


Fig. 14 - Construction materials for the heavy concrete

energy distribution, an HVL is equal to 5 to 7 cm of ordinary concrete or 3 to 5 cm of heavy concrete.

In the case of power reactors, the main shielding is poured permanently in place. It is frequent practice, at research reactors, to have much of the biological shielding movable in the form of large cast blocks. This allows some flexibility in experimental setups and eases the maintenance tasks. However, care must be taken to assure that all the shield blocks are back in place before start-up!

In addition to shielding the core, coolant pipes must also be heavily shielded. The coolant becomes activated by neutron bombardment (in the core) of corrosion products and additives in the circulating fluid. Pipes and valves are often shielded with locally applied sheet lead, lead wool or encased lead "blankets." Lead shot is sometimes used to fill irregular voids in shielding. During maintenance on these components, protective clothing is worn to shield against betas emitted by spilled coolant.

During refueling operations, the fuel elements must be handled. In a power reactor, these fuel elements contain megacuries of beta-gamma emitters. This problem is dealt with by use of water shielding. The individual fuel elements are stored in a "spent fuel pool" which provides gamma shielding and cooling for the decay heat. Typically, the pool provides about 8 meters of water above the top of the fuel elements.

Medical Facilities

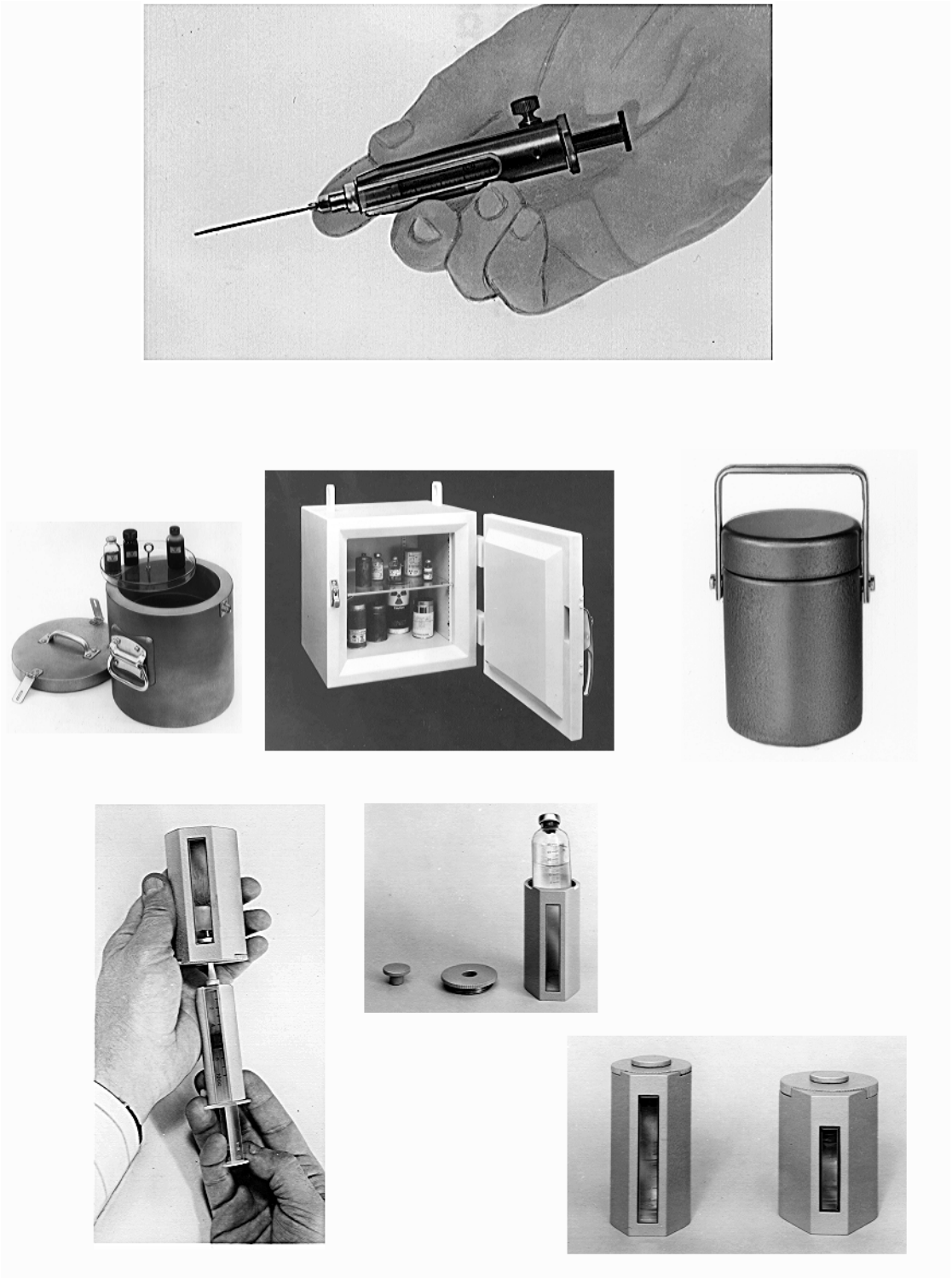
In the nuclear medicine laboratory, the typical problem from a shielding standpoint is beta-gamma sources used for diagnosis or treatment. The radioactive cows discussed in Chapter 6 are the largest activity sources. A ^{99m}Tc generator may contain from tens of gigabecquerels (hundreds of mCi) to a tenth of a terabecquerel (several curies) of Mo-99 while the ^{113m}In generators usually are under 4 GBq. Lots of lead bricks are used along with portable shields and syringe shields to protect the hands of the nuclear medicine technician during injections. A sampling of commercially available shields is shown in Figure 15.

In the diagnostic x-ray department, shielding must be provided in the walls which surround the various types of x-ray producing equipment (radiographic, fluoroscopic, CAT scanner, etc.). Lead is used almost universally for this purpose. Due to the Z^3 dependence of the photoelectric cross section, the lead is unusually effective. The required thicknesses are calculated using a method discussed in detail in National Council on Radiation Protection and Measurements Report 147, "Structural Shielding Design for Medical X-Ray Imaging Facilities."

The basic approach in NCRP 147 is to consider that the radiation exposure rate at a point is due to two components – the primary radiation from the direct beam and secondary radiation from leakage through the machine housing and scattered radiation from the patient. Any structural barrier (wall, floor, door or ceiling) that receives exposure from the primary radiation is termed a "primary barrier" while all other barriers are "secondary." The design goal is to reduce the exposure rate at a point outside the room to either 10 mR/week (0.1 mSv/week) for controlled areas occupied only by radiation workers or 2 mR/week (0.02 mSv/week) for non-controlled areas frequented by the general public. The radiation level present without shielding is determined at a distance of one meter from the photon source as the product W U T, where:

W = Workload in mA min/week
U = Use Factor and
T = Occupancy Factor.

The Workload includes the number of exposures made per week and the intensity settings of the equipment. The Use Factor accounts for the fact that the machine usually points in different directions for different types of exposures. Finally, the Occupancy Factor takes into account the fact that a given location outside the room wall is not necessarily occupied 100% of the work week by the same person. When W U T is divided by r^2 to account for the inverse square fall off of exposure rate with distance to the barrier, the weighted weekly exposure rate at that location is obtained. If this value is higher than the 2 or 10 mR/week design level, the ratio of the desired rate to the calculated actual rate gives the attenuation factor needed for the barrier. Report 147 then contains a section of graphs of the attenuation factor vs. shield thickness that are read to find the amount of lead that must be placed on the barrier. Sheet lead is readily available already bonded to gypsum wallboard or to various types of wood paneling. Typical medical x-ray installations require between 1/32 inch and 1/16 inch of lead for adequate protection.



Courtesy, Nuclear Associates, Carle Place, NY

Fig. 15 - Examples of nuclear medicine shielding

In the medical radiation oncology department, NCRP Report 151 is used for shielding calculations. It is titled "Structural Shielding Design and Evaluation for Megavoltage X- and Gamma-Ray Radiotherapy Facilities." The Co-60 equipment or medical accelerators used for treatment provide higher energy radiations than is covered by NCRP Report 147. Report 151 is intended for use only by a "qualified expert," and so is not discussed further here. Generally about 1 to 1.5 meters of concrete is needed for a primary barrier around a Co-60 unit or lower energy linac while 0.3 to 0.6 meters usually provides an acceptable secondary barrier against leakage and scatter radiation.

Industrial Radiography

Nondestructive testing is one of the largest uses of industrial radiation machines. A "shadow picture" is made by the industrial radiographer of some object with x- or gamma rays for purposes of inspection or quality assurance. X-ray machines have been in use since 1954 for performing structural examinations of aircraft. Clearly, the ideal of a shielded room enclosure is not practical for taking radiographs of an intact 747 aircraft. In addition to the sheer size of the "object," other economic factors dictate that time cannot be taken to isolate the aircraft for picture taking sessions. (It's only earning when it's flying). Thus, radiographic examination of aircraft takes place right in the maintenance building. Film packs in light-tight wrappers are attached to the outside surface during the normal work shift while other maintenance personnel are performing their tasks. Then, while other persons are on break, the radiographer makes the film exposures.

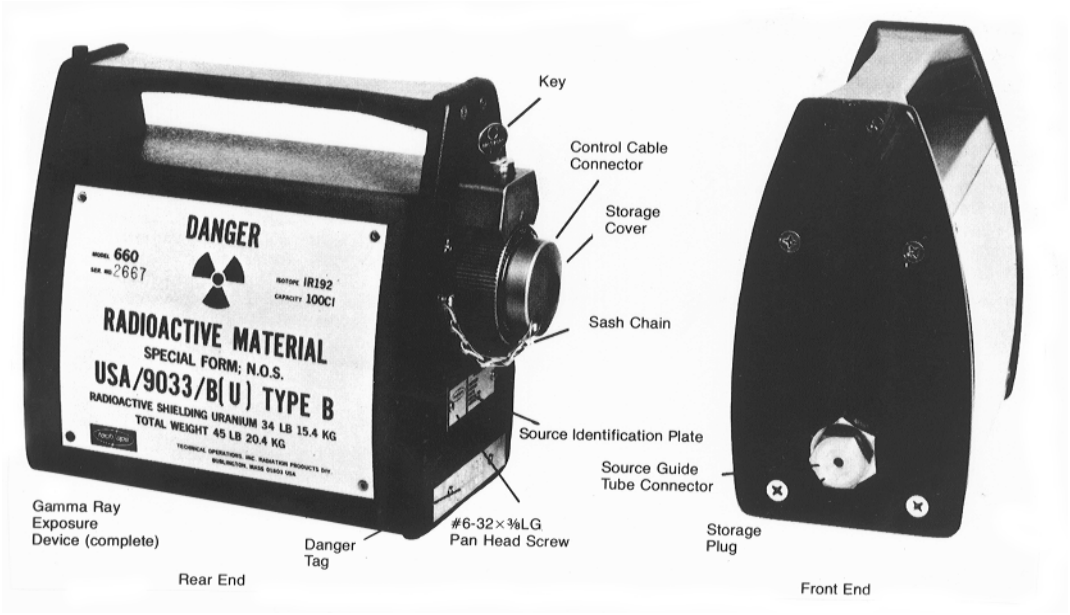


Fig. 16 - A gamma radiography exposure device (camera)

Courtesy, Amersham Corporation

Another area where industrial radiographers are involved is in the field inspection of welds on cross-country pipelines. Obviously, these welds must be inspected in the trench just before pipe burial. For pipes larger than 16 inches in inside diameter, the x-ray machine is frequently mounted on a motor driven crawler at the end of a 1500 foot cable. It is remotely advanced inside the pipe to the weld location. Photographic film is attached around the outside circumference of the pipe and a 360° "panoramic" exposure is made. The film is usually developed in a field darkroom. On small diameter pipelines, 4 separate shots are taken with an external source pointing across the pipe toward the film pack on the opposite side. Figure 16 shows a typical gamma radiography "camera" and Figure 17 shows the associated control cable, hand crank and source guide tubing.

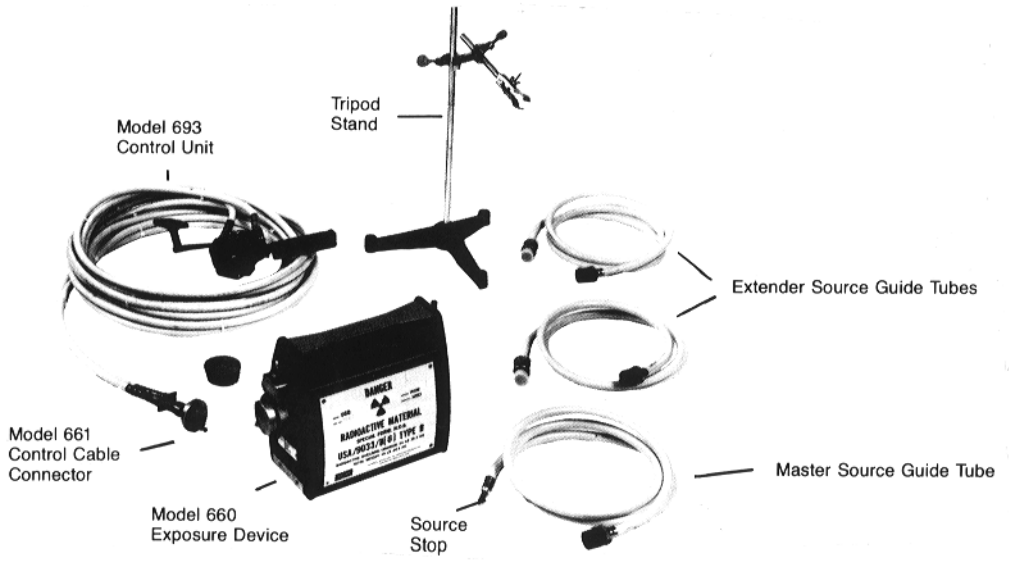


Fig. 17 - A complete industrial radiography setup

Courtesy, Amersham Corporation

Radiation protection in this application is somewhat tricky. The x-ray machine on the crawler is invisible to bystanders. If the pipeline is in a ditch, the earth sides provide some degree of protection for other workers. Rope barriers and signs can also be used effectively (see Figure 18).

In-plant industrial radiography is usually done with shielded rooms of concrete or lead. The doors must be interlocked to prevent personnel from entering while the source is on. Wall thicknesses are designed so that radiation levels are low enough that no person outside the room is likely to receive more than 1 millisieverts (100 mrem) in any one year. The actual design calculations can be performed using NCRP Report 147 as described previously. Figure 19 shows the design of a typical shielded room for industrial radiography.

There are a number of specific legal requirements for radiation safety when working with industrial radiography sources. Part 34 of Title 10 of the Code of Federal Regulations, 10 CFR 34, spells these out. Some of the more important rules are summarized as follows:

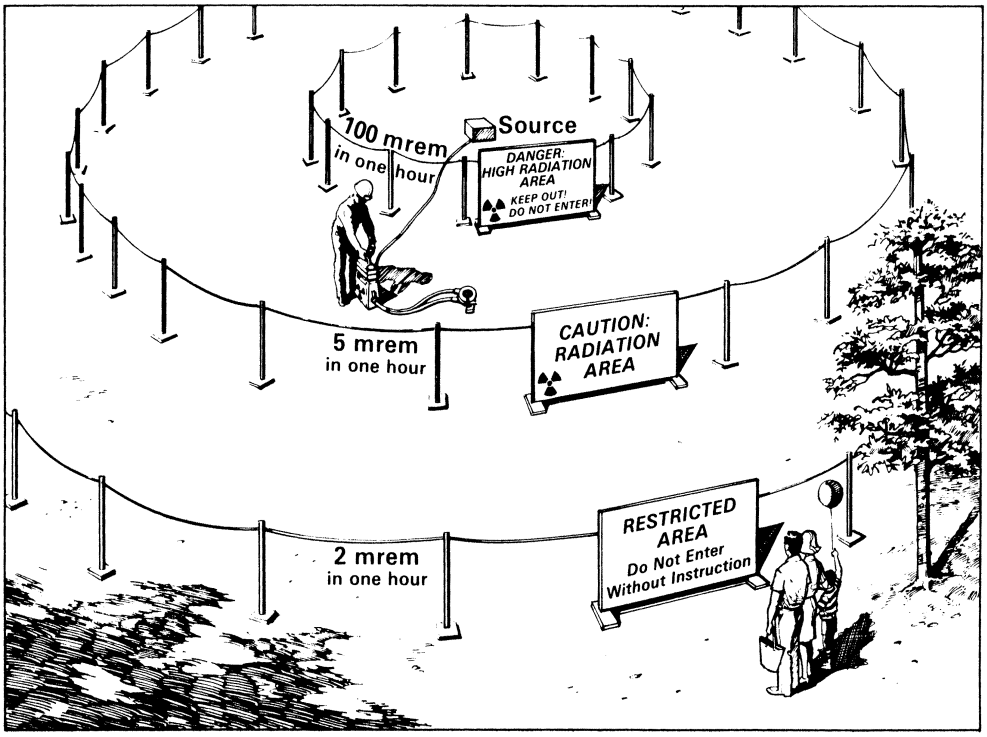


Fig. 18 - Use of barriers for exposure control

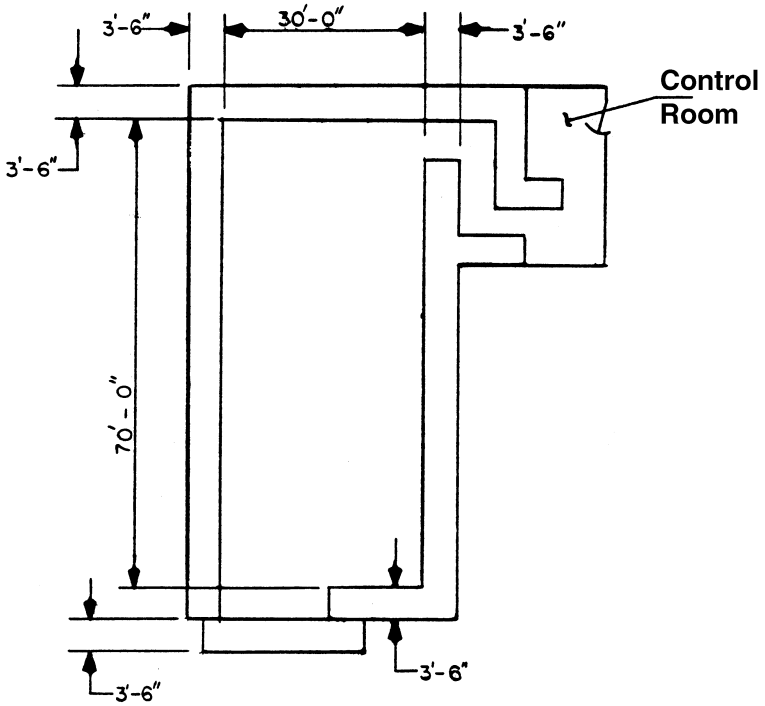


Fig. 19 - Example of a floor plan for a shielded radiography room

• **Surveys** - An instrument survey must be made after each exposure to prove that the source has retracted back into the shield. This is of critical importance. Virtually all of the fatal radiography accidents involving lost sources could have been prevented if this rule had been followed. (See Chapter 14 for a discussion of some of these.) Also a survey of the restricted area boundary must be made for each setup.

• **Signs** - Radiation warning signs must be posted conspicuously at boundaries of the high radiation area and the radiation area.

• **Surveillance** - If it is possible that someone could wander into a high radiation area (100 mrem in one hour) then direct visual surveillance must be maintained unless some means is used to prevent entry (e.g., a locked door).

• **Personnel Dosimetry** - The radiographer and assistants must wear a pocket dosimeter plus an alarm ratemeter set to 500 mR/hr and either a film or TLD badge. The pocket dosimeter must be read and recharged each day.

• **Survey meter** - Radiography meters must be calibrated every three months and be able to cover an exposure range from 2 mR/hr to 1,000 mR/hr.

• **Inventory** - A physical inventory of sources must be conducted every three months to prove they are not missing.

• **Utilization Log** - Each radiography source needs a log sheet showing when and where it was used.

• **Training** - The radiographer and assistants must receive training in radiation basics, bioeffects of radiation, safety procedures and must study case histories of former radiography accidents. Written documentation is required.

Nuclear Particle Accelerators

The shielding of nuclear accelerators depends on the machine type and energy. Negative ion machines usually are operated as electron accelerators. Positive ion machines accelerate nuclei which have had some or all of their atomic electrons stripped off. In terms of energy, accelerators are classed as shown in Figure 20.

Accelerator shield design involves two steps. The largest thicknesses are required for the prompt radiation field. The induced field dominates after accelerator shutdown. In electron accelerators, the bremsstrahlung field is the major component of the prompt radiation for virtually all energies. In the energy range from 15 to 50 MeV, some electron accelerators will also produce a significant neutron hazard through photoneutron production (the bremsstrahlung photon kicks out a neutron from the target or collimator assembly). Concrete rooms are the usual choice for these machines. From published data, the photon intensity in the various directions can be estimated. The HVL concept is then useful for calculating the needed thicknesses. It should be noted that in electron linacs, the shielding must usually be considerably thicker in the direction

Energy Range	Class
0 to 50 MeV	Low Energy
50 to 1,000 MeV (1 GeV)	Medium Energy
Over 1 GeV	High Energy

Fig. 20 - Accelerator classification by energy

that the beam is pointing compared to the transverse direction (at right angles to the beam direction).

In the case of positive ion accelerators, neutrons are the dominant contributor to the radiation field for low and medium energy installations. Hydrogenous materials are used as previously described. At high energies, the positive ion machines also produce muons in addition to relativistic neutrons. It has been found that a layer of iron is quite effective for the initial shielding. Through inelastic scattering, the neutrons are reduced in energy to where hydrogenous materials can be used to moderate and thermalize them.

Finally, skyshine is always potentially present with accelerators just as with any large radiation source. The neutrons which travel upward are scattered back downward toward the ground to cause a radiation field outside the shielding walls. This problem can be prevented by placing adequate neutron shielding in the ceiling of the accelerator vault. As a practical point, the presence of a large skyshine component can be shown by careful measurement of the dose equivalent rate as a function of the distance from the shield wall. Since it is not geometrically possible for the scattered radiation to reach points close to the wall without passing through a large thickness of shielding, the dose equivalent rate will first rise with increasing distance from the wall and then will flatten out and finally fall as expected from the inverse square law (see Figure 8 in this chapter). Close to the base of the wall, the upper wall section produces an "umbrella" which provides a "shadow" region without skyshine.

Transporting Radioactive Packages

Introduction

One of the more practical applications of the shielding principles is in the design of containers for transporting radioactive materials. To date, there have been no known deaths or injuries to transport workers, emergency services personnel, or the general public as a result of the radioactive nature of materials in transport. This safety record can be attributed to the proper packaging of radioactive shipments. There is an average of about three million radioactive materials shipments per year in the United States.

Currently, the DOT, the US Postal Service, The Transportation Safety Administration, and the NRC have responsibility for the safety of radioactive shipments. The DOT regulations are published in Title 49 of the Code of Federal Regulations, (CFR) Parts 100-185 (collectively these are referred to as the Hazardous Materials Regulations or HMR). The latest revision at press time was October, 2010. The NRC regulations are found in Title 10, Part 71 of the CFR (most recent revision January, 2011).

There are a number of international bodies and organizations which deal with the transportation of radioactive material. The majority of these international bodies are sanctioned by or affiliated with the United Nations (UN). These agencies write regulations and recommend their adoption by member states as a basis for national regulations. In the United States shipments of radioactive material can be made both

domestically and internationally that conform to one of the codes discussed below rather than to the US DOT HMR regulations.

The International Atomic Energy Agency, IAEA, is a component of the United Nations that provides advice on matters relating to nuclear and radioactive material safety. IAEA has developed recommended procedures for the safe transport of radioactive materials in publication TS– R–1.

The International Maritime Organization (IMO) implements the UN recommendations in the International Maritime Dangerous Goods (IMDG) Code. The IMDG Code contains regulations applicable to the transport of dangerous goods by sea. If all or part of a shipment of hazardous materials is made by vessel to, from, or within the United States, the HMR allow the shipment to be made in accordance with the IMDG Code, provided certain additional provisions are satisfied.

The International Civil Aviation Organization's **Technical Instructions on the Safe Transport of Dangerous Goods by Air** (ICAO TI) establishes requirements necessary to ensure hazardous materials are safely transported in aircraft while providing a level of safety that protects the aircraft and its occupants from undue risk. The U.S. Hazardous Materials Regulations authorize transport in accordance with the ICAO TI.

Air carriers have adopted their own regulations through the International Air Transportation Association (IATA). These IATA Dangerous Goods Regulations (DGR) are based on the ICAO TI, but they are generally more restrictive in certain operational respects. Most domestic carriers have chosen to only accept shipments prepared under the ICAO TI as implemented by the IATA DGR. Thus, virtually all shipments of hazardous materials transported internationally by air, as well as most domestic US shipments, are transported in accordance with the IATA DGR and ICAO TI.

In the United States, postal shipments are under the jurisdiction of the US Postal Service (USPS). As of 2011 the USPS has not adopted the definition and limits found in TS-R-1. Shipments made using the United States Post Office need to follow the regulations found in USPS Publication 52. The U.S. Postal Service will accept some radioactive packages. The quantity of radioactive material is limited to 1/10 the values listed in the "Excepted Package" DOT regulations below shown in Figure 27.

Definition

Before discussing packaging, labeling and shipping papers, it might be a good idea to define a "radioactive package." The HMR defines nine classes of hazardous materials. Radioactive material is assigned to Class 7. Prior to 2004, the HMR used a specific activity threshold of 70 Bq/g (0.002 μCi/g) for defining a material as radioactive for the purposes of transportation, and the material was not subject to the requirements of HMR if its specific activity was equal to or below that value. In 2004, the HMR was revised by establishing two radionuclide-specific values. (See Fig 21.)

The first limit is a radionuclide-specific activity concentration limit for exempt material. Materials that have a concentration of radioactive material equal to or below this value are not regulated as a dangerous good and are termed "exempt." The second limit is an exempt consignment activity limit. Shipments where the total radioactivity in a consignment is below the consignment activity value have a risk so

Radionuclide	Element and Atomic #	Exempt Material Limits	
		Activity Concentration Bq/g (µCi/g)	Consignment Activity Bq (µCi)
T (H-3)	Tritium (1)	10,000 (0.27)	10,000,000 (270)
C-14	Carbon (6)	10,000 (0.27)	10,000,000 (270)
Co-60	Cobalt (27)	10 (0.00027)	100,000 (2.7)
Mo-99	Molybdenum (42)	100 (0.0027)	1,000,000 (27)
Cs-137	Cesium (55)	10 (0.00027)	10,000 (0.27)
Ra-226	Radium (88)	10 (0.00027)	10,000 (0.27)

Fig. 21 - Exempt Material Activity Concentrations and Consignment Limits for Selected Radionuclides

small that the material can be transported without being subject to the transportation regulations. "Consignment" means a package or group of packages or load of radioactive material offered by a person for transport in the same shipment.

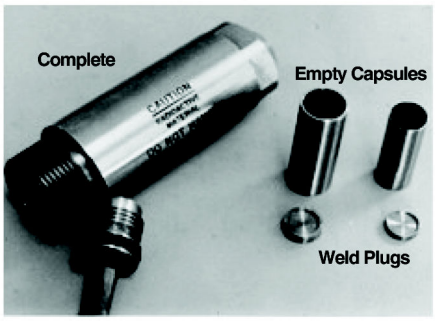
To be considered a radioactive material under the HMR, the material must exceed both the nuclide specific concentration limit and the consignment exemption limit. Radioactive material that is less than either one of these requirements is not regulated by the DOT as radioactive and cannot be shipped or declared as a Class 7 material.

Classification

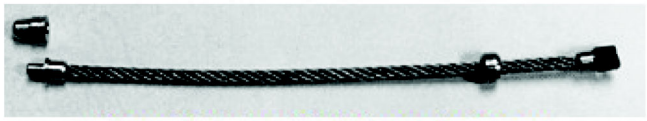
Once it is determined that the radioactive material meets the Class 7 definition, the next step is to properly classify the material. In the United States there are five general classification terms for radioactive material that depend on three factors – radionuclide, quantity and form. A radioactive material may meet the definition of one or more of the following classifications.

Low specific activity or LSA material is radioactive material that has a low activity per unit mass (specific activity). LSA material is divided into three sub-groups of increasing concentrations: LSA-I, LSA-II, and LSA-III. Most LSA materials have a characteristic of presenting limited radiation hazard, because of their relatively low concentration of radioactivity. When the specific activity of an LSA material is computed, the radioactivity is divided by the mass of material in which the radioactivity is distributed; the mass of the packaging that may surround the LSA is excluded from the calculation.

Surface contaminated objects (SCO) are solid objects which are not radioactive by themselves but which have radioactive material distributed on their surfaces (rather than distributed within the material as for LSA materials). SCO are classified into two sub-groups SCO I and SCO II (The latter allows for higher contamination levels than SCO-I). LSA implies activity within a material, while SCO implies activity on a material. LSA and SCO are extremely important classifications for shipments of low-level radioactive waste.



Neutron Source



Industrial Radiography Pigtail Source

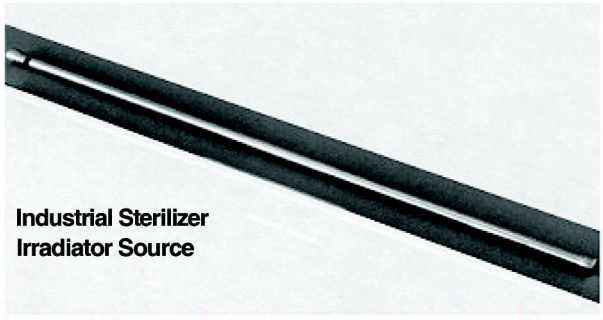


Fig. 22 - Special Form radioactive material

A fissile material is capable of sustaining a chain reaction of nuclear fission. By Hazardous Material regulation only the following four radionuclides are considered fissile; uranium-233, uranium-235, plutonium-239, plutonium-241, or a mixture containing any one of these isotopes. In addition to radiation safety considerations, shipments of fissile material must meet a number of other stringent requirements to ensure against accidental nuclear criticality.

Special Form material is radioactive material in an indispersible solid form or in a sealed capsule. Usually, due to the high physical integrity of a Special Form material, radioactive material contamination is not expected even under severe accident conditions. Manufacturers of Special Form material must apply to the DOT for a Special Form Certificate prior to shipment. Special Form certification can become "acquired" by encapsulating a Normal Form material in a very durable container which can be opened only by destroying the capsule. Figure 22 illustrates the Special Form definition. Special Form capsules must pass a 9 meter drop test, a percussion

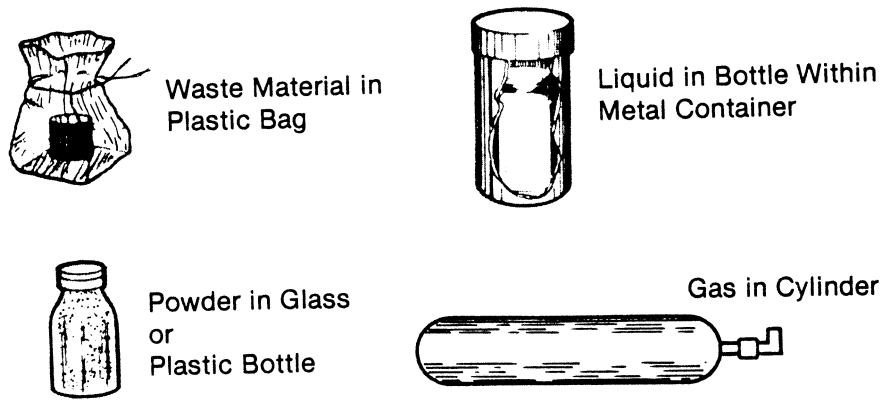


Fig. 23 - Examples of "Normal Form" radioactive materials

test, a bending test, an 800 °C heat test, a leak test and a four-hour water immersion test (see 49 CFR 173.476.).

Normal Form materials (a.k.a. "other form") are any radioactive materials which have not been tested and certified as Special Form. Normal Form includes all solid, liquid or gaseous radioactive material that does not meet the definition of a Special Form. See Figure 23. Any radioactive material that does not have a Special Form certificate and cannot meet the characteristics of another classification is, by default, Normal Form.

Both Special Form and Normal Form radioactive materials are subdivided into two types, Type A and Type B, based on the amount of activity in a package. The distinction is illustrated in Figure 24. A_1 and A_2 are quantities of radioactivity which are

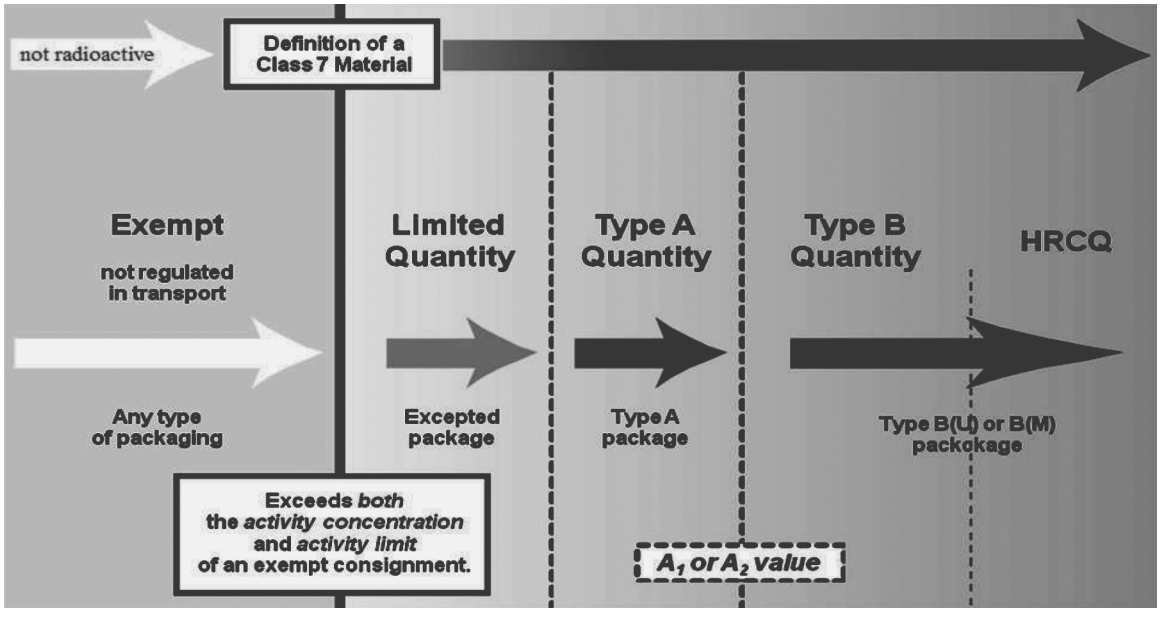


Fig. 24 - Radioactive Material Quantity and Package Categories

used in the regulations to determine factors such as the type of packaging necessary for a particular radioactive material shipment. Each radionuclide is assigned an A_1 and an A_2 value expressed in terabequerels. A_1 applies to Special Form and A_2 applies to Normal Form material. 49 CFR 173.435 contains a table of A_1 and A_2 values for over 250 of the most common radionuclides. Figure 25 shows the values for some typical radionuclides.

A_1 is the maximum activity of Special Form material that is permitted in a Type A package, and A_2 is the maximum activity of Normal Form radioactive material that is permitted in a Type A package. Activities below the respective Special Form, A_1 values or Normal Form, A_2 values are referred to as Type A quantities. These represent the maximum amounts that can be shipped in a "Type A" container. Quantities of material with activity greater than A_1 or A_2 value are called Type B quantities.

Radionuclide	Element and Atomic #	A_1 Special Form TBq (Ci*)	A_2 Normal Form TBq (Ci*)
T (H-3)	Tritium (1)	40 (1100*)	40 (1100)
C-14	Carbon (6)	40 (1100*)	3 (81*)
Co-60	Cobalt (27)	0.4 (11*)	0.4 (11*)
Mo-99	Molybdenum (42)	1 (27*)	0.6 (16*)**
Cs-137	Cesium (55)	2 (54*)	0.6 (16*)
Ra-226	Radium (88)	0.2 (5.4*)	0.33 (0.081*)
U-238	Uranium (92)	Unlimited	Unlimited

*** The values in curies (Ci) are approximate and NOT the regulatory standard.**
**** Mo-99: 0.74 TBq (20 Ci) for domestic shipments**

Fig. 25 - Type A package quantity limits for selected radionuclides

But how are these A_1/A_2 limits really calculated? In the 1970s, transportation experts developed the Q-System to predict radiation doses in the event of a transportation accident. The Q-System, as it has presently evolved for both, sets up scenarios for external gamma ray exposure, beta dose to skin (external irradiation, submersion and from deposited contamination), lung dose from breathing in radiocontaminants, and internal dose from ingestion of contaminants. Radionuclide decay information is then used to calculate the dose (per TBq of packaged activity) to a person standing near the damaged package for the 250 radionuclides tabulated in 49 CFR 173.436. Dividing the scenario dose limit of 5 rem by the dose/TBq gives the A_1/A_2 limits. Some radionuclides, such as natural uranium, are so stable that it would take an unrealistic shipping amount to pose a Type B risk. These radionuclides have an unlimited A_1 and A_2 value, thus, there is no Type B quantity.

Packaging

Safety in transporting radioactive material primarily depends upon the use of the proper packaging for the type, quantity, and form of the radioactive material to be transported. Figure 24 provided an illustration of the "spectrum" of radioactive quantities and package types. In addition, packaging design is performance oriented, with the packaging integrity being dictated by the hazards of the radioactive content. Small

quantities of radionuclides that pose low radiological risk can be shipped in a Type A package. Large quantities require a Type B package.

If the material to be shipped falls below the A_1/A_2 limits, then a Type A package is legally usable. This package must pass a one-hour water spray test, a four-foot drop test, a 24-hour compression test and a 1.25-inch diameter penetration test. If the package is to ship radioactive liquids or gases it must also pass a 30-foot drop test and contain enough absorbent material to absorb at least twice the volume of the liquid contents. Figure 26 exhibits some Type A packages.

When only a small fraction of the A_1 or A_2 activity is being shipped the package can be shipped in an "excepted package" where some of the requirements of the HMR are waived. The following types of materials may be eligible for such exceptions:

- Limited quantity of radioactive material
- Radioactive material incorporated in instruments or articles
- Articles manufactured from natural or depleted uranium and thorium.
- Empty packagings

The activity limits for "excepted packages" radioactive materials, instruments and articles are given in Fig. 27. A "limited quantity" of radioactive material is an amount of radioactive material that does not exceed the quantity limitation specified in §173.425 (see Fig. 27, last column) and conforms to the requirements specified in §173.421

"Radioactive instruments or articles" are manufactured items such as clocks, electronic tubes, gauges, smoke detectors, electronic apparatus, or similar devices

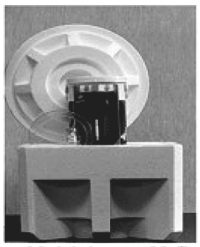


Figure A - Molybdenum 99 Generator
(Cutaway shows outer carton, foam spacer, shielding, ion column, and tubing for saline solution)

Figure B - Moisture Density Gauge & Carrying Case

Figure C - 55 Gallon Steel Drum

Figure D - Components of a Type A Package for Isotopes

Fig. 26 - Typical type A package configurations

	Instruments & Articles		Materials
	Instruments & Articles Limits	Package Limits	Package Limits
Solids: Special Form	**0.01 A_1**	**A_1**	**0.001 A_1**
Other Forms	**0.01 A_2**	**A_2**	**0.001 A_2**
Liquids: Tritiated Water			
<0.1 Ci/liter	-	-	**37 TBq**
0.1 to 1.0 Ci/liter	-	-	**3.7 TBq**
> 1.0 Ci/liter	-	-	**0.037 TBq**
Other liquids	**0.001 A_2**	**0.1 A_2**	**0.0001 A_2**
Gases: Tritium	**0.8 TBq**	**8 TBq**	**0.8 TBq**
Special form	**0.001 A_1**	**0.01 A_1**	**0.001 A_1**
Other forms	**0.001 A_2**	**0.01 A_2**	**0.001 A_2**

Fig. 27 - Values for "excepted packages" of radioactive shipments 49 CFR 173.425

having radioactive material in gaseous or non-dispersible solid form as a component part. Allowance is made for the additional protection provided by the structure of the instrument or article and they are considered excepted quantities if they do not exceed the limits in §173.425 and conform to the requirements specified in §173.424. As shown in Fig. 27, there are two sets of limits, one for the item and another for the package. The first column applies to the activity per piece while the second (Package Limits) applies to the complete package which may hold several individual instruments or articles. If the shipment qualifies as an "Excepted Package" then only a strong, tight package that meets the general packaging design requirements is needed.

If the A_1 or A_2 (as appropriate) limit is exceeded, a Type B package must be used. In addition to meeting all of the general packaging and performance standards for Type A packages, Type B packages must also have the ability to survive serious accident damage tests. There can only be a very limited loss of shielding capability and no loss of containment during the testing. The performance criteria for a Type B package include:

- 30-foot free drop striking the package's weakest point
- 40-inch free drop puncture test
- Exposure of the entire package to 1475° Fahrenheit (800° C) for 30 minutes
- Immersion of the package under 50 feet of water for at least 8 hours.

Examples of these containers are shown in Figure 28. Manufactures and users of Type B Packages must be registered with the NRC. Note that both Type A and Type B packages must incorporate a security seal which is not readily breakable and is evidence that the package has not been opened during transit. The Type B package design requirements guarantee a very low probability of significant activity release in the event of a very severe accident.

There are cases where very large activity shipments are made, such as spent fuel from a nuclear reactor. If the package contains more than 3,000 times the A_1 or A_2 limits or 1,000 TBq of activity (whichever is less) then it is designated a "Highway Route Controlled Quantity" and is subject to additional restrictions. Only certain roads are approved and the route taken must be reported to the shipper and the governors of each state that the material will be transported through.

Figure A - RH-TRU 72B Cask

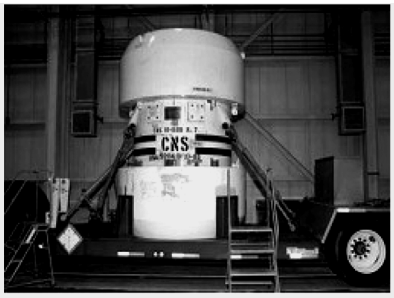


Figure B - CNS 10-160B

Figure C - 3 TRUPACT-II Packages

Figure D - Industrial Radiography Exposure Device (cutaway shows "S" tube for source in the shielding material)

Fig. 28 - Examples of Type B packages

Markings and Labeling

The next step prior to getting the package out the door is proper marking and labeling. Package markings and labels are used by carriers to ensure compliance with loading and stowage requirements. These requirements are designed to prevent potentially dangerous situations that may occur with incompatible hazardous materials. Package markings and labels convey information on packages, such as the proper shipping name, identification number, and hazard class (Class 7) of a hazardous material. This information identifies that a package contains a hazardous material. The information provided by package markings and hazard warning labels can be used by emergency responders when shipping papers are destroyed or otherwise not immediately available.

Packages containing radioactive materials need to have the required markings on the outside of the container. Some examples of these include the proper shipping name, UN ID, name and address of the shipper and recipient, the type of packaging e.g., TYPE A or TYPE B(U). Type B packages require that a trefoil symbol be embossed or stamped onto the package so that it is resistant to the effects of fire and water (a sticky label just won't do it.)

Fig. 29 - Hazard labels for radioactive shipments

Each radioactive package must be normally labeled on two opposite sides with a radioactive hazard label, unless shipping LSA materials via "exclusive use" arrangements or excepted packages. The radiation hazard warning labels (see Figure 29) fall into three classes – an all white background (White I), a yellow upper half label with two red stripes (Yellow II) and a yellow upper half with three red stripes (Yellow III). To determine which label is needed, the surface radiation level and the "transport index," or TI, is determined for each package. The transport index is merely 100 times the maximum dose equivalent rate, expressed in mSv/hr, measured at one meter from the external surface of the container. (Since 1 mSv =100 mrem the TI also = maximum mrem/hr @ 1 meter). The TI is rounded up to the nearest tenth (except a TI less than 0.05 may be taken as zero) and is shown on shipping papers and radioactive material labels without units. The table in Figure 30 determines the correct label.

Label Category	Transport Index (T.I.)	Radiation Level at Package Surface (RL)
White - I	0	RL ≤ 0.005 mSv/hr
Yellow - II	T.I. ≤ 1.0	0.005 mSv/hr < RL ≤ 0.5 mSv/hr
Yellow - III	1.0 < T.I.	0.5 mSv/hr < RL ≤ 10 mSv/hr

Fig. 30 - Transport Index limits for shipments

Packages that contain fissile material are labeled with a fissile hazard label (Figure 31) to warn against the possibility of a criticality event, (i.e., uncontrolled nuclear fission caused by exceeding the critical mass of the fissile material). In addition to a transport index, packages containing fissile material (those not excepted under § 173.453) must be assigned a criticality safety index (CSI). Like the TI, the CSI is a dimensionless number, rounded up to the next tenth, which is used to provide control over the accumulation of packages, overpacks or freight containers.

Documentation

There are now two steps remaining to complete the process. First, the shipping papers must be filled out properly, using, as mentioned above, the correct SI units. To ensure compatibility with international transportation standards, Title 49 CFR

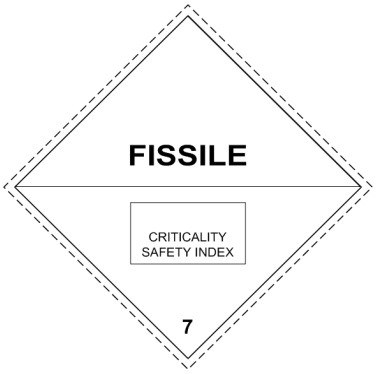


Fig. 31 - Fissile hazard label

171.10 mandates the use of SI units on labels and shipping documents. This went into effect in 1997. The old mrem & Ci units can be placed in brackets after the necessary SI unit, but this is not recommended as it can potentially create confusion. (This just reinforces the notion that technologists really do need to get "up to speed" on these SI units!)

Information on the shipping papers includes the proper shipping name, hazard class, ID number, radionuclide, form, activity, label category, transport index, required package markings and Shipper's Certificate. In the event of an accident, this information is very useful to the emergency personnel on the scene.

These proper shipping names (PSN) have been harmonized with those used internationally; there are no longer any generic proper shipping names for radioactive material with the phrase "not otherwise specified (n.o.s.)." Most of the proper shipping names are based on the type of package used for the shipment. Figure 32 lists some of the more common PSNs for radioactive material shipments

I.D. Number	Proper Shipping Name
UN 2910	Radioactive Material, excepted package - limited quantity of material
UN 2911	Radioactive Material, excepted package - articles
UN 2915	Radioactive Material, Type A package
UN 3332	Radioactive Material, Type A package, Special Form
UN 2916	Radioactive Material, Type B(U) package
UN 2912	Radioactive Material, Low Specific Activity (LSA-I)

Fig. 32 - Some commonly used proper shipping names (PSN)

Section 172.600 requires shippers to provide emergency response information on hazardous materials shipments. At a minimum, the emergency response information must provide:

- The basic description and technical name of the hazardous material
- Immediate hazards to health
- Immediate precautions to be taken in the event of an accident or incident
- Immediate methods for handling fires
- Immediate methods for handling spills or leaks in the absence of fire
- Preliminary first aid measures.

This information must be on a shipping paper or an associated document and kept in the vehicle as well as maintained at all locations where the shipment is handled.

Shippers are required to provide an emergency response telephone number which must be monitored on a 24-hour basis while the shipment is in transport. The number must be of a person or entity who is knowledgeable regarding mitigation information or has immediate access to such a person.

Carrier Requirements

The final step in getting the package out the door is placarding and loading of the conveyance. Before releasing the package to the carrier, it is also necessary to measure the removable surface contamination with a wipe test (49 CFR 173.443). The smear is taken "using moderate pressure" over an area of 300 square centimeters, NOT THE 100 square cm MORE COMMONLY WIPED BY TECHNOLOGISTS. Maximum permissible limits ON THE PACKAGE are 220 dpm/cm^2 for beta/gamma emitters and low toxicity alpha emitters and 22 dpm/cm^2 for all other alpha emitters. These levels are the surface limits for removable contamination. Usually, smears are used to assess the removable contamination levels. It is assumed that the smear technique has 10% efficiency. Therefore, shippers should multiply the smear data by 10 before comparing it to the limits.

The carrier is the company that physically picks up and delivers the package. The carrier relies on the Shipper's documentation as evidence that the package meets current regulations. If the shipment contains any Yellow III labeled packages, the rail or highway vehicle must be placarded with a 12" diamond-shaped radioactive placard on each end and on both sides of the conveyance (see Figure 33). Highway Route Controlled Quantity (HRCQ) shipments must have the required Radioactive placard placed on a square white background.

In order to protect handling personnel, the carrier must assure that the total transport indices for all packages in a single shipment or storage location does not exceed 50. In addition, there are limits on the dose rates for individual packages that can legally be shipped. Generally, no package can be accepted if the external surface dose rate exceeds 2 mSv/hr or the transport index exceeds 10. One exception is allowed, the provisions of "exclusive use."

Exclusive use means sole use by a single shipper of a conveyance for which all initial, intermediate, and final loading and unloading are carried out in accordance with the direction of the shipper or recipient. The shipper and the carrier must ensure

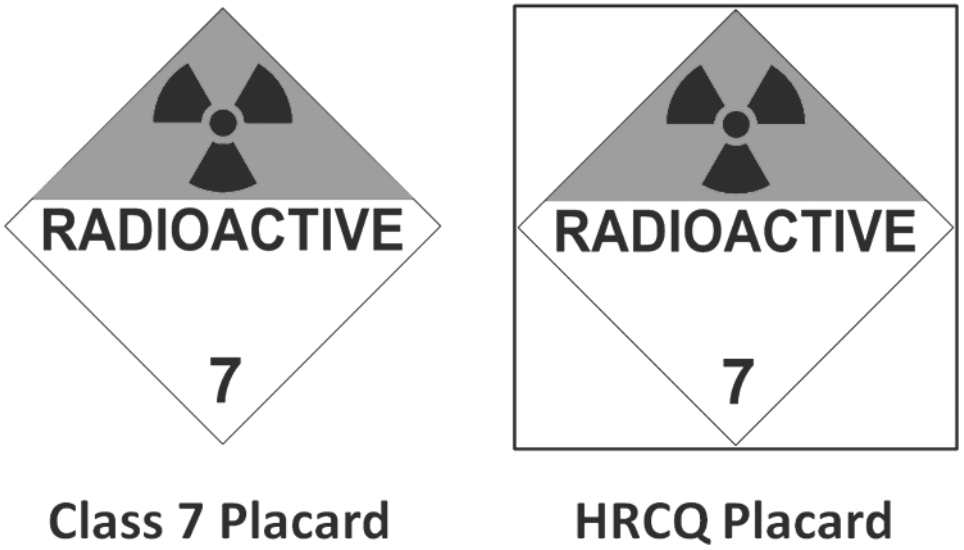


Fig. 33 - Class 7 and Highway Route Controlled Quantity placards

that any loading or unloading is performed by personnel having radiological training and resources appropriate for safe handling of the consignment. Under exclusive use transport, the following limits apply:

- 10 mSv/hr limit on package surface (if closed vehicle & pkg. secured)
- 2 mSv/hr limit on vehicle surface
- 0.1 mSv/hr limit at two meters from vehicle surface
- 0.02 mSv/hr limit at any normally occupied position in vehicle.

Security and Other Requirements

As part of the NRC's efforts to improve radioactive material security after the events of September 11, 2001, the NRC requires additional security measures when an individual and/or company is engaged in certain NRC-licensed activities. These additional security measures include advance notification to the NRC and state governors or their designated representatives about certain radioactive material shipments. The individuals and/or companies engaged in NRC-licensed activities were issued these security measures through a modification of their NRC license and they are cognizant of its specific requirements. The NRC regulations in 10 CFR 71.97 and 10 CFR 73.72 require that licensees shipping HRCQ of nuclear waste in Type B packages, spent nuclear fuel, and special nuclear materials provide advance notification to state governors or their designated representative.

No person may export or import any radioactive material from or to the United States unless authorized by a general or specific license issued by the NRC. Some quantities of radioactive material can qualify for a NRC general license for export or

import under 10 CFR Part 110. The general licenses in Part 110 have limitations on the type and quantity of radioactive material and permit the export of radioactive material to certain countries. Other countries may be Embargoed or Restricted which means that no radioactive material can be exported under a NRC general license to these countries. If an export or import is not covered by the NRC general licenses, the shipper must file an application with the Commission for a specific license. In addition to the license requirements, the NRC may also impose reporting requirements for certain quantities of material being exported or imported.

The NRC requires the recipient to establish, maintain, and retain written procedures for safely opening packages. The recipient must make sure that these procedures are followed and that due consideration is given to special instructions for the type of package being opened.

In 10 CFR 20.1906(d), the NRC requires that recipients perform monitoring on packages they receive that contain licensed quantities of radioactive materials. The monitoring must be done as soon as practical, but no later than 3 hours after the package is received. If there is evidence of degradation to the package, such as a package that is crushed, wet, or damaged, the recipient must measure for potential surface contamination, measure surface radiation level, and measure the radiation levels 1 meter from the package. If the package is in good condition, the type of monitoring is dependent on the package type, form, physical state and hazard label. When surface contamination or external radiation levels exceed the limits established in the HMR, the NRC requires that the licensee immediately notify the final delivery carrier and the NRC by telephone.

Training

All hazmat employees involved in the transport of radioactive material must be trained and tested in four areas. The first area is general awareness training designed to provide familiarity with requirements of the DGR and to enable the employee to recognize and identify hazardous materials. The second area is security awareness training instructing users to recognize possible security threats and enhance transportation security. Title 49 CFR Part 172, Sections 800-804, establishes the requirements for the development and implementation of detailed site-specific security plans for shippers and carriers of all Yellow-III and HRCQ packages, a security plan must be developed by the employer. More in-depth training on this specific security plan would then be required for all hazmat employees. The third area, function-specific training, must address DGR requirements which are applicable to the functions the employee performs. The final area, safety training, is mandated which includes measures to protect the employee from the hazards posed by materials, procedures for avoiding accidents, and emergency response protocol.

The proper procedures for a given radioactive shipment are complex. It is important that technologists who have a responsibility for preparing such shipments should rely on the actual DOT regulations rather than the simplified summary included here. In addition all shippers of radioactive material need to be trained, tested and certified in the regulations prior to shipping radioactive material. Finally,

changes are continually being made to bring the U.S. into conformity with the ever changing IAEA provisions for international transportation. Good luck!

Internal Protection

Introduction

While time, distance and shielding are valid protection principles for external radiation fields, different methods are used to protect personnel against internal hazards associated with loose radioactivity. One of the most common intake pathways is inhalation. This section will focus primarily on respiratory protection principles. It begins with a discussion of respirators. Then, engineering controls are briefly discussed, i.e., the use of chemical fume hoods and sealed glove boxes to reduce inhalation risks. Finally, a short discussion is held concerning the use of various types of protective clothing.

Note that respirators are NOT the first choice in protecting workers from airborne hazards. In fact, Part 20 of Title 10 of the Code of Federal Regulations requires that a "licensee shall use, to the extent practicable, process or other engineering controls (e.g., containment or ventilation) to control the concentrations of radioactive material in air." Therefore, respirators are used only when other methods of airborne contamination control are not feasible.

Respirators

Probably the biggest change in recent years in internal protection practices involves respirators - or more correctly, the non-use of respirators. In the mid 1990s, the 50 year old Code of Federal Regulations was changed. As discussed in Chapter 9, it now requires the control of a worker's TEDE rather than just the external dose component that was controlled under the old law. In practice, this allows a much more flexible approach to dose management. Internal doses are no longer "to be avoided at any cost." It has been found in the nuclear utility industry, in particular, that total doses are often less if a small internal component is allowed. Many jobs can be completed more rapidly and comfortably when respirators are not mandated. In one two-unit nuclear utility polled, respirator use averaged 6000 units per year in the mid 80s but dropped to 10 per year in the mid 90s.

The current 10 CFR 20 still allows respirators to be used in airborne radioactivity areas if a number of conditions are met. The items listed here are taken from 10 CFR Part 20.1701 through 1704. The respirator selected must provide a "protection factor" sufficient to lower the inhaled concentration of radioactivity below the allowed occupational air concentration. The protection factor is defined as follows. See Sample Problem 8.

$$\textbf{Protection Factor} = \frac{\textbf{Ambient Airborne Concentration}}{\textbf{Concentration Inhaled}} \qquad \textbf{[Eqn. 5]}$$

The individual wearing the respirator must have received training in proper use and the respirator must have been tested for operability immediately before being

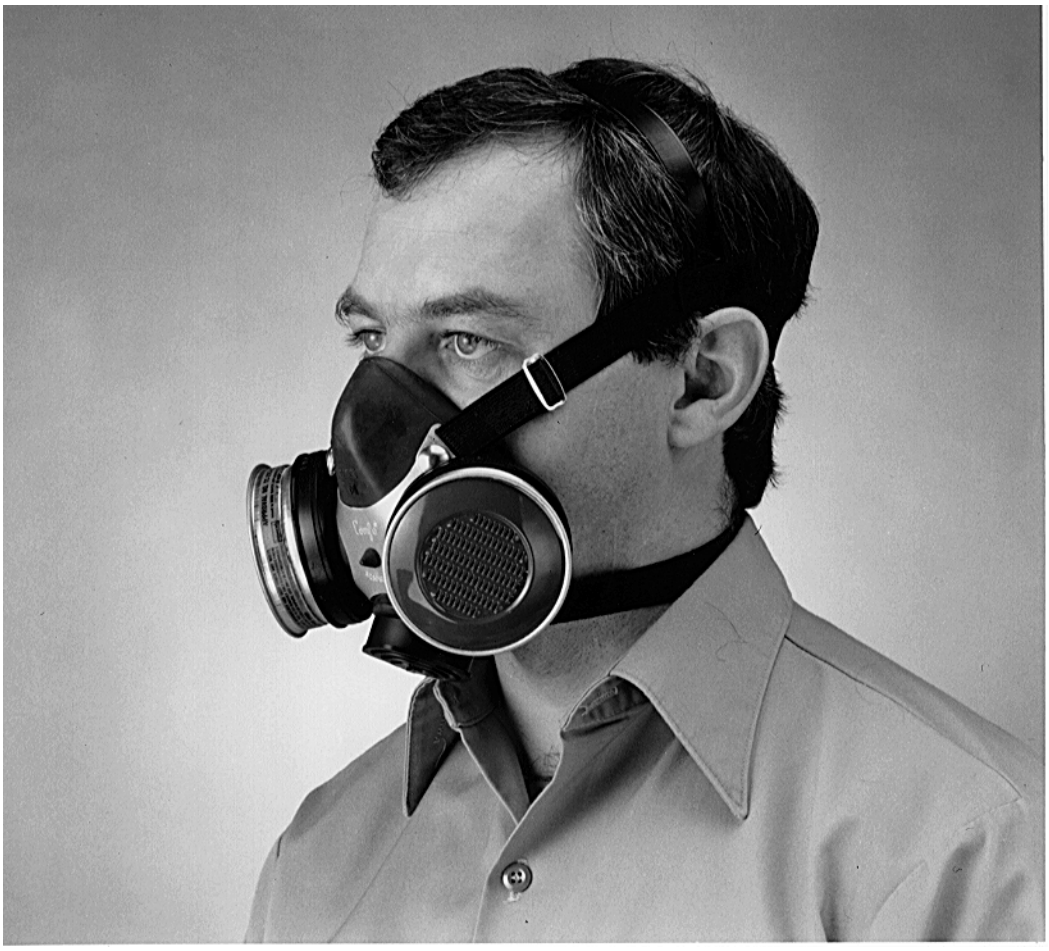
Courtesy, Mine Safety Appliances Co.

Fig. 34 - A full-face respirator

used. The licensee must have written procedures for selecting, fitting, maintaining, testing, training and record-keeping. Physician approval is needed before initial fitting of a respirator and periodically thereafter certifying that the person is physically able to wear respiratory protective equipment. A rule change in 1995 allows the physician to specify the frequency of reexamination. Only respirators approved by the National

Sample Problem 8

GIVEN:
A technician plans to work in an area where the air sampler indicates 20 DAC.
FIND:
What protection factor would be needed for the technician to not exceed 10% of DAC?
SOLUTION:
The desired concentration to be inhaled = 0.1 DAC. Thus, from Eqn. 5, Protection Factor = 20 DAC ÷ 0.1 DAC = 200.



Courtesy, Mine Safety Appliances Co.

Fig. 35 - A half-face respirator

Institute for Occupational Safety and Health/Mine Safety and Health Administration, NIOSH/MSHA, can be utilized.

There are four categories of respiratory equipment generally available:

- Air-Purifying Respirators – full-face or half-mask
- Air-line Respirators – full-face or half-mask
- Self-Contained Breathing Apparatus
- Air-Supplied Hood

The air-purifying type respirator includes a rubber face mask and an associated filter cartridge. Figures 34 and 35 show examples of the full and half-face models. The air-purifying respirator always operates in "negative pressure mode." This means that the pressure inside the mask is below ambient pressure during inhalation. This is, of course, a more hazardous mode than a "pressure demand mode" or "positive pressure mode" in which the mask pressure is always higher than ambient pressure. Positive pressure greatly reduces the possibility of radioactivity leaking past the mask seals and entering the inhaled air. A filter cartridge cannot be used unless it is of the high efficiency particulate (HEPA) type that removes more than 99.97% of 0.3 micron diameter dioctyl phthalate (DOP) particles. Clearly, no protection is provided

against radioactive gases and vapors. Even if special adsorbents are contained in a cartridge, the law does not recognize their value since it is impossible to know when the cartridge is used up and gases and vapors begin to break through.

Respirator filters are also rated on their ability to perform in atmospheres containing oil mists. An "N" rating means not resistant while a "P" rating means oil proof. Thus, the commonly available "P100" respirator cartridge is usable in oil mists and will filter out 100% of 0.3 micron particulates while an "N95" disposable mask is not permitted in oil mists but it will remove 95% of the 0.3 micron particulates.

An air-line respirator (Figure 36) consists of a mask and an attached hose which is connected remotely to a source of clean, filtered air. Many facilities have an

Courtesy, Mine Safety Appliances Co.

Fig. 36 - An example of an air-line respirator

air system permanently installed with convenient connection points located throughout the plant. A pressure regulator is usually attached to the person's belt. These units usually supply a continuous flow of air to the mask. Because they operate in a positive pressure mode, the protection factors are much higher than the air-purifying cartridge respirator. Air-line respirators are legal for use in atmospheres containing radioactive iodine while the cartridge respirator is not.

The third category is self-contained breathing apparatus. Figure 37 shows an example of a SCBA.This equipment provides the highest level of protection. It includes a mask, a pressurized cylinder of air and a demand type pressure regulator that keeps the air pressure inside the mask slightly higher than ambient. SCBA equipment, along with air-line respirators, can be used in oxygen-deficient atmospheres, i.e., air containing less than 16% oxygen by volume.

Courtesy, Mine Safety Appliances Co.

Fig. 37 - A self contained breathing apparatus

Fig. 38 - A supplied air hood respirator

There are a number of problems to be overcome if a successful respirator program is to be achieved. The major difficulties in obtaining a proper fit are caused by facial hair and by wearing corrective eyeglasses. Sideburns and beards interfere with the proper sealing of the mask to the facial skin surface. This is most critical in the negative pressure cartridge respirator. Ordinary eyeglasses also interfere because of the ear pieces. Special inserts are available to hold prescription lenses inside full-face masks. These, of course, must be fitted ahead of time. Note that contact lenses are not an acceptable substitute. They become unattached if used with a positive pressure mode respirator.

One solution to several of the problems noted above, particularly the eyeglass problem, is to use the last category of respirator, a supplied-air hood. If the airflow rate is 6 cubic feet per minute minimum, a protection factor of 1000 can be used. No special fitting requirements are needed for this loose fitting respirator. Facial hair does not have to be shaved to use one successfully. Comfort level is relatively high as the exhaust from the air hood is usually routed through the neck and vented into the coveralls to provide cooling. Voice communication is not as muffled as with a rubber respirator. Figure 38 shows one in use. PAPRs, powered air purifying respirators, are becoming quite popular. When combined with a hood, a very easy to use system with unlimited mobility results. No fit program is needed, yet the protection factor is 1000.

If a supplied-air suit is used, federal regulations require “a standby rescue person equipped with a respirator or other apparatus appropriate for the potential hazards and communications equipment.” When working in suits, persons tend to dehydrate rapidly if heavy exertion is involved. The radiation protection technologist, as part of a work party, needs to be alert for this condition. Along similar lines, some workers experience high stress levels or become claustrophobic in a respirator or supplied-air suit. Again, the technologist should be aware of this possibility and be prepared to escort affected workers from the job site.

Mask Type	Mode	Particulates	Particulates + Gases + Vapors
I. Air Purifying			
Half-Mask	NP	10	
Full-Face	NP	100	
II. Air-Line			
Half-Mask	CF, PD		50
Half-Mask	D		10
Full-Face	CF, PD		1000
Full-Face	D		100
Hood (6cfm)	CF		1000
III. S C B A			
Full-Face	PD		10,000
Full-Face	D		100

Fig. 39 - Allowed protection factor values

In order to decide if a given respirator type is adequate for the job, the formula given earlier for the protection factor is used. The minimum protection factor required is obtained by dividing the measured air concentration at the worksite by the allowed concentration. The allowed concentration value in the U.S. is the Derived Air Concentration or DAC. Again, numerical values are found in Appendix B to Part 20 of Title 10 CFR. The maximum protection factor that can legally be used is given in these same regulations. The table in Figure 39 lists the values that are acceptable under 10 CFR 20 as of 2011. The abbreviations used for the Mode are as follows: NP = Negative Pressure, CF = Continuous Flow, D = Demand (mask at negative pressure during inhalation) and PD = Pressure Demand (mask always at positive pressure).

Finally, the law requires a written policy statement on the use of respirators at the licensee's facilities. This must discuss the use of engineering controls instead of respirators, the routine, nonroutine and emergency use of respirators and must state the periods of use and relief from respirator use. The user of a respirator has the right to leave the work area at any time when the worker feels that the equipment has failed, communications have failed, the wearer is experiencing physical or psychological distress, operating procedures or conditions have broken down or for any other condition that might require such relief. The licensee is responsible for assuring that respirator users are aware of these rights.

Fume Hoods

One of the more common engineering controls used for airborne contaminants is the chemical fume hood. A hood is generally acceptable when working with quantities in the range of 1 to 10 times the ALI for a radioisotope. The proper design and installation of hoods is beyond the scope of this book. However, some general principles will be covered.

The airflow into a hood must be taken into account in the overall building heating and air conditioning design. Losses of conditioned air up a hood stack can be a major problem. It is important to take into consideration the effect of failure of a hood motor on room airflow. The design must prevent the sucking of air from the hood out into room air under this failure condition. Also, under normal operations, the pressure in the hood stack must be negative relative to ambient room pressure until the stack clears the building. This means that the hood motor must be installed at the point of release (usually the building roof) so that if the stack develops leaks in the future, it will still suck air into the stack rather than eject particulates out into the room or building attic.

A fume hood used with radioactivity requires a high efficiency particulate (HEPA) filter on the exhaust. The filter should have 99.97% collection efficiency for 0.3 micron DOP particles. It must also have gasket seals, be fire resistant and be readily replaceable.

A major design parameter is the average face velocity of air into the hood. If the airflow is too great, turbulence (caused by bottles, lead bricks etc. in the opening) will greatly increase the risk of backflow of contamination out the hood face. If the airflow is too low, the "capture radius," or distance from the opening over which the hood is effective, will be too small to protect workers. The generally accepted average flow rate needed is from 125 to 275 linear feet per minute. The average is calculated by dividing the hood opening into several rectangular areas. Then, the air velocity at the center of each area is measured and a simple average calculated.

Glove Boxes

When working with unsealed radioactive sources in excess of about 10 ALIs, the hood can no longer provide adequate protection. The glove box is useful in this case. It consists of an enclosed work space with glove ports through which work can be performed in complete isolation, see Figure 40. Typically, only a very small airflow is maintained, just sufficient to keep the box at a negative pressure of about 0.5" water. As a rule of thumb, assume that contamination corresponding to 10^{-8} of the contained activity will leak out of the box during operations. Although almost any glove box will stop alpha emitters, beta sources are considerably more penetrating. If view ports or windows are too thin, unprotected eyes could receive a radiation dose as well as the skin of the hands which are shielded only by the glove thickness. In some cases, double gloves are necessary to prevent this problem.

Protective Clothing

Technicians are generally quite familiar with protective clothing. They may be less familiar with the things that can go wrong. Anti-contamination clothing or anti-C clothing is not a guarantee against personal contamination. Even if the correct items are chosen, they must be put on and used properly.

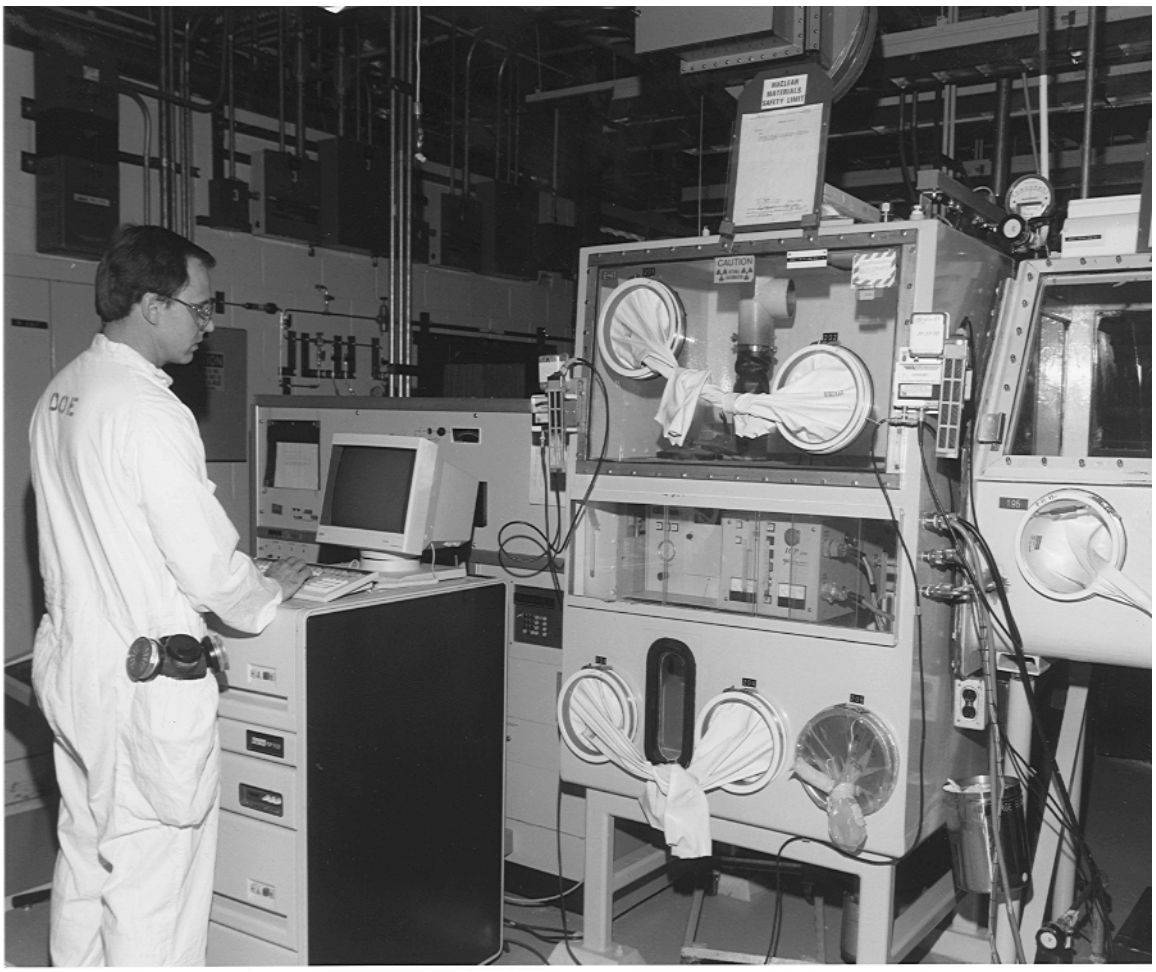


Fig. 40 - Examples of glove boxes in use at EG&G Rocky Flats

Work under strenuous conditions is the most common cause of failure of the clothing. Perspiration soaked areas of coveralls are likely spots for contamination to penetrate. Choice of the coverall fabric greatly influences the performance. Cotton suits are popular because they allow reuse after laundering. However, particles up to 300 microns can penetrate cotton. Water vapor and liquids can pass right through. On the other extreme, Tyvek® will block water vapor and liquids and is impenetrable by particulates. Unfortunately, tyvek® blocks airflow. Thus, the technologist can be flooded and overheated by perspiration build-up inside the suit. An ideal solution would be a waterproof but breathable **f**abric. The Kimberly-Clark Kleenguard® coverall comes close. It blocks particles down to about 0.5 micron, withstands water penetration up to a pressure of 61 cm of water, yet allows an airflow rate of 48 cfm per square foot through the fabric for breathability.

Experience has also shown that fewer layers of anti-Cs will reduce the likelihood of skin contamination. This seemingly illogical situation is caused by the extra heat stress placed on the worker with extra layers. With less stress, there may be less perspiration and soaking of the clothing and thus less penetration by contamination.

Problem Set

1. What is the ALARA philosophy? Why is there a need for ALARA when we have a complete set of radiation protection standards?

2. Describe some practical tasks which are required of both "management" and the "radiation protection staff" under the ALARA Regulatory Guide 8.10.

3. Calculations show that if a lead brick enclosure is constructed around the radioisotope generators in a nuclear medicine lab, the two technicians would each receive about 200 mrem less annual dose equivalent. The 50 bricks required would cost $40 each. Do you consider this a cost-effective change? Why or why not?

4. How is the nuclear power plant radiation work permit an example of the use of TIME to control exposures?

5. What is the recommended distance to be maintained by visitors from patients who are receiving treatment with therapeutic amounts of radionuclides? Who made this recommendation?

6. Estimate the dose equivalent that would be delivered for the two techniques described in Sample Problem 2 for loading a new 8 Ci Co-60 source.

7. A technician measures the exposure rate from a small lead source storage pig to be 15 mR/hr at 4 feet. The storage pig will be located in a room where radiation workers are stationed at a distance 2 meters away from it. About how many hours per week can the radiation workers spend, on the average, at the work station without exceeding 50 mrem/week?

8. Calculate the thickness of lead needed to reduce the dose equivalent rate produced by a ^{226}Ra source from 50 mSv/hr to 2.5 mSv/hr. See Figure 6.

9. Why does the simple exponential photon attenuation law (Equation 2) underestimate the exposure rate behind a thick shield wall? How is the actual exposure rate estimated under these conditions?

10. Why is it improper to shield a strong beta source with just a piece of steel with a thickness equal to the range of the most energetic beta particle?

11. Calculate the exposure rate from bremsstrahlung radiation at 50 cm for the source in Sample Problem 6 if the first shield layer is made from iron.

12. Estimate the range in mm in aluminum (density = 2.7 gm/cubic cm) of the most energetic beta from a ^{90}Y source.

13. What might be the cause of an ion chamber instrument reading a gamma ray level from a 1 MeV pure beta source shielded by 2" of lead?

14. Low energy positive ion accelerators are often used to produce neutrons of about 15 MeV in energy through a nuclear reaction involving deuterons bombarding a tritium target. About how much concrete is needed to shield such a machine if it is estimated that the neutron flux at the closest point of occupancy is about 1.7×10^5 n/sq cm-sec without any shielding? Use a design level of 2.5 mrem/hr. (Also see Chapter 5).

15. What radiation protection problems are posed by an industrial radiographer working on welds in a large cross-country pipeline? How are they solved?

16. How often does the law require a physical inventory be taken of industrial radiography sources?

17. Where in the federal regulations are the sections dealing specifically with radiation protection for industrial radiographers?

18. A technician measures a dose rate of 185 mrem/hr maximum in contact with the side of a company truck being used as an "exclusive use" vehicle to transport radioactive material. Does this shipment meet current regulatory dose rate requirements?

19. A radiopharmaceutical supplier packs 3 vials containing 2 ml each of ^{125}I labeled drug in a single package for shipment. How much absorbent material is required?

20. A Type A wooden shipping crate used to transport a Co-60 sealed source is measured to have a maximum dose rate in contact with the box of 53 mR/hr. The reading is 0.8 mR/hr at one meter. What type of radioactive label must be affixed? What is the maximum amount of surface contamination allowed for this shipment?

21. Name three advantages of an air-supplied hood respirator over other respirator types.

22. What is the legal maximum protection factor that can be used with an air-purifying respirator in an atmosphere containing krypton-85 noble gas? In an atmosphere containing iodine-131 vapor?

23. Which federal agency is charged with approving respiratory protective equipment?

24. Which respiratory equipment type has the highest protection factor? What value of protective factor does this equipment have?

25. Does the precaution of adding more layers of anti-C clothing always reduce contamination risk?

S-1. Under what conditions does the inverse square law "fail," i.e., give the wrong estimate of the dose rate at some farther distance compared to a closer distance? In those cases where it does not work, if the dose rate is measured at a distance of 4 meters, is the dose rate at 2 meters likely to be more than or less than 4 times the rate measured at 4 meters?

S-2. A technician measures the thickness of steel needed to reduce the photon exposure rate to 50% of the initial rate. He then adds more steel plates to reduce the rate to 25% of the initial rate. He finds that it takes more steel to reduce from 50% to 25% than from 100% to 50%. What can be concluded about the photon field from this observation?

S-3. What three materials might be used to shield 100 MeV neutrons? In what order are they placed?

S-4. What type of radiation is the prompt field associated with a 90 MeV electron accelerator? What shielding material would commonly be used for such an installation?

S-5. Excessive skyshine is found to be the problem at a 40 MeV positive ion accelerator laboratory. What radiation type is probably the largest contributor to the skyshine field? How could the skyshine be reduced?

S-6. Is heavy concrete more effective in attenuating photons or neutrons when compared to ordinary concrete?

S-7. Describe some methods that would be useful for localized shielding of hot spots in a research reactor coolant pipe or valve.

S-8. Why is lead a particularly effective shield for syringes used to inject Tc-99m into medical patients?

S-9. As used for x-ray machine shielding design, what does the term "W U T" represent?

S-10. In working with an unsealed 0.33 MBq lead-210 source, what engineering controls would you recommend to prevent inhalation? Assume an inhalation ALI of 0.2 microcuries.

Other Resources

1. NCRP Report 107, "Implementation of the Principle of As Low As Reasonably Achievable (ALARA) for Medical and Dental Personnel," National Council on Radiation Protection & Measurements, Bethesda, MD, 1990.

2. "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonable Achievable," US Nuclear Regulatory Commission Regulatory Guide 8.10, Revision 1-R, Washington, DC, 1977

3. "Acceptable Programs for Respiratory Protection," US Nuclear Regulatory Commission Regulatory Guide 8.15, Revision 1, Washington, DC, 1999.

4. NCRP Report 155, "Management of Radionuclide Therapy Patients," National Council on Radiation Protection & Measurements, Bethesda, MD, 2006.

5. The following NCRP reports form a complete "package" for medical and dental shielding design:

- Report 145, "Radiation Protection in Dentistry," 2003.
- Report 147, "Structural Shielding Design for Medical X-Ray Imaging Facilities," 2004.
- Report 151, "Structural Shielding Design and Evaluation for Megavoltage X- and Gamma-ray Radiotherapy Facilities," 2005.

6. Formal training on the regulations for transporting radioactive materials can be found at http://www.class7training.com.

12

Surveys, Calibrations and Data Analysis

Outline of this Chapter

Chapter Summary

The important operational task of conducting a proper radiological survey with portable, calibrated instrumentation is the focus of this chapter. Techniques used to monitor alpha, beta, gamma and neutron fields are discussed and these principles illustrated by monitoring examples from a variety of sites. MARSSIM surveys specific to decommissioning a radiation facility are then covered. The chapter concludes with a discussion of counting statistics for the interpretation of survey readings.

Alpha surveys usually use a proportional counter or alpha scintillator. The low penetrability of alphas leads to some special requirements. Gamma fields are surveyed with Geiger counters (low field intensities) or ion chambers (high field intensities). The large volume of an ion chamber can lead to erroneous interpretations. Beta contamination is usually monitored by geiger counter while beta fields are measured with an ion chamber. Correction factors must be applied to account for gamma interference and for the non-uniform irradiation of the detector. A variety of instruments are used to monitor neutrons. Rem meters read dose equivalent directly. Neutron fluxes can be converted to dose rates. Removable surface contamination is assayed by a wipe test. The same technique is useful for leak testing sealed sources.

Decommissioning of a nuclear facility provides many survey opportunities. The U.S. NRC, DOE, DOD and EPA have collaborated on decommissioning criteria. Guidelines are specified in the MARSSIM Manual in terms of direct radiation levels, surface activity concentrations, and volume concentrations in soils. They also specify a series of survey procedures leading to license termination – scoping, characterization and final status surveys.

To decide on the radioactivity of a sample that counts close to background, counting statistical laws must be invoked. The standard deviation is usually used to represent the error in a counting measurement. Taking into account background corrections, a confidence level can be established for each nuclear measurement. The sensitivity of a counting procedure is measured by the minimum detectable concentration or MDC.

Principles of Monitoring & Calibration

Introduction

This chapter discusses the techniques used for radiation monitoring in a variety of field situations. Each radiation field type will be discussed separately. For each field, the applicable portable instruments will be listed along with suggestions for proper use. Before monitoring can be conducted, the technologist must have an appropriate, calibrated survey meter. Thus, the beginning point will be a short discussion on generally applicable calibration techniques. Unique calibration problems for the specific instrument types will be discussed under the various radiation field categories that follow the general calibration topic.

As of 1998, NRC licensees have been required to adhere to a new Subpart E, **Radiological Criteria for License Termination**, of Title 10, Part 20 of the Code of Federal Regulations. This section was added to the law to deal with decommissioning and decontamination, "D and D," activities. Standard procedures covering survey instrument choice, sampling design, survey grids, survey procedures and quality assurance procedures have been codified in The Multi-Agency Radiation Survey and Site Investigation Manual, MARSSIM. The MARSSIM has been accepted by the U.S. Nuclear Regulatory Commission, The Environmental Protection Agency, The Department of Energy and The Department of Defense. Introductory material on MARSSIM surveys is included in several places in this Chapter. Detailed information on conducting MARSSIM D and D projects is presented in Supplemental Chapter S-3.

General Calibration Principles

At its simplest, calibration means the adjustment of an instrument to accurately read the radiation level from a reference source. Three levels of "calibration" are generally recognized. These include a full characterization (usually only done by the instrument manufacturer), a calibration for specific acceptance (when an instrument is used under abnormal conditions) and a routine calibration for instruments used under normal working conditions.

The evaluations that are done as part of a full characterization are listed in Figure 1. In the case of a specific acceptance calibration, the response of the instrument under the abnormal condition is assessed. Such conditions might include operations at lower or higher temperatures than the manufacturer-specified operating conditions, or, for example, in a strong magnetic field. The abnormal condition is re-created in the calibration lab and the instrument adjusted to the required accuracy.

The routine calibration is probably the most familiar to a practicing radiation protection technologist. It is called for when a survey meter is going to be used for a radiation type specified by the instrument manufacturer within the designed energy range. The radiation sources used for calibration should have been assayed with reference to a national standards laboratory source. In the USA, this laboratory is NIST, the National Institute of Standards and Technology in Gaithersburg, MD. Calibration

Energy dependence
Linearity
Interference by other radiation types
Dependence on temperature, pressure, humidity
Effects of radiofrequency, magnetic and/or electrostatic fields
Geotropism
Effect of mechanical shock
Dose rate dependence or dead time
Angular response
Temperature shock

Fig. 1 - Parameters measured in a full characterization calibration

lab sources which have been compared, perhaps through intermediary sources, with the NIST sources are said to be "NIST Traceable."

Using NIST traceable sources, instrument calibration laboratories in the commercial sector can become accredited by the Health Physics Society. As of 2011, there was only a single HPS Accredited Laboratory, K&S Associates in Nashville, TN. Following a committee review of an application and qualifications, testing with NIST supplied ion chambers verifies the lab's ability to reproduce the NIST field conditions to within ± 5%. Finally, a field visit occurs by an HPS assessor who observes calibrations in progress and reviews quality control and operating procedures. The accreditation is valid for three years and then the lab is reassessed.

The instrument must also be used within the specified environmental parameters (temperature, humidity, air pressure, electrostatic, radio frequency and magnetic fields) and within specified mechanical stress limits. (If the manufacturer says it is designed to withstand a 30 foot free-fall onto concrete – check it out! Otherwise, try not to bang it around too much.) Finally, some instruments show marked geotropism – the unfortunate tendency of a meter movement to read different values dependent on how you orient the meter (i.e., the meter movement is not properly compensated for the effects of gravity). Under recommendations of the American National Standards Institute, an instrument that changes by not more than a few percent when pointed in different directions has acceptable geotropism.

In between calibrations, the technologist is responsible for conducting performance checks on survey meters. The purpose is to show that, at a minimum, the survey meter is capable of responding to the check source radiation. The initial performance check is done at the calibration laboratory, immediately after a routine or specific application calibration. A particular check source is placed in a reproducible location and the reading noted. If later "field checks" fall within ± 20% of the initial check source reading, the meter is considered to have demonstrated that it is working. Before using the instrument for a survey, the technologist should verify that the meter has a current calibration sticker, that the batteries have sufficient power and that the meter passes the field check.

NRC Regulatory Guides specify that radiation survey instruments should be calibrated at twelve month intervals. The complete procedure is shown in Figure 2.

Daily or other frequent checks of survey instruments should be supplemented every twelve months with a calibration of each instrument at two points separated by at least 50% of each linear scale that is used routinely or with a calibration at one point near the midpoint of each decade on logarithmic scales that are used routinely. Survey instruments should also be calibrated following repair. A survey instrument may be considered properly calibrated when the instrument readings are within ± 10% of the calculated or known values for each point checked. Readings within ± 20% are considered acceptable if a calibration chart or graph is attached to the instrument.

Fig. 2 - NRC recommended instrument calibration procedures

The Regulatory Guides also point out that calibration with an electronic pulse generator is not acceptable except on the highest ranges of an instrument if those ranges are not used routinely.

The calibration lab itself must meet certain conditions to enable acceptable results to be produced. Ideally, the ambient background radiation level is so low that background never contributes more than 1% to the calibration field intensity. In some cases, this can be met by use of shielding. In other cases, background subtraction must be done to reduce the overall uncertainty of the calibration. Proper account must be taken of scattered radiation. This is particularly a problem in neutron calibrations. Air scattering and backscattering are also problems in beta calibrations. Finally, reasonable environmental conditions must be maintained. These include a relatively constant temperature between 18° and 27° C, a relative humidity between 10% and 70% and the absence of interfering magnetic, radio frequency or electrostatic fields.

If the calibrated survey instrument is to be used for D and D residual contamination monitoring, the MARSSIM identifies several problem areas and factors affecting calibration. These are listed in Figure 3.

Alpha Radiation Monitoring

Generally, alpha fields are encountered only as a result of surface contamination or in the form of airborne particulates. Only the surface contamination case will be covered here since alpha air sampling was discussed in Chapter 10. As contamination, alpha particles present the potential for internal deposition. They do not usually present an external hazard. As discussed in Chapter 2, energies of natural and artificially produced alpha emitters fall between 1.8 MeV and 11.6 MeV (85% are between 4 and 7 MeV). A 4 MeV alpha travels 2.5 cm in air while a 7 MeV alpha has a range of 6 cm in air. Due to the high stopping power, alpha radiation will not penetrate clothing or the dead layer of skin. Thus, the purpose of monitoring is to locate contamination before human uptake takes place. The portable survey instrument must have an extremely thin window in order for the alpha particles to penetrate. In addition, the counting rates that need to be measured are well below the normal background rate in a Geiger counter. This means that two types of instruments are in practical use – the portable proportional counters and the portable scintillation counters.

Meter is used for type of radiation the meter was designed for
Radiation energy is within meter design range
Environmental conditions are within meter design range
Magnetic & electrostatic fields are within meter design range
Meter is oriented such that geotropism is not a problem
Meter is used within mechanical and thermal stress design range
Calibration lab energies are similar to survey field energies
Source-to-detector distance is similar between calibration lab and field use
Surface condition and composition are taken into account

Fig. 3 - Calibration issues for MARSSIM surveys

There are two different models of portable alpha proportional counters. One uses air at ambient pressure as the counting gas while the other makes use of propane gas or P-10 gas (10% methane, 90% argon). The instruments include a case containing the batteries, power supply, amplifier and ratemeter circuit. The external probe assembly connects to the case with a coaxial cable. The actual detector (probe) usually has a rectangular entrance window with an area of about 100 square cm. The windows are often covered by aluminized mylar or carbon-coated plastic film to make them electrically conducting (they are part of the detector cathode). Alpha windows must be extremely thin because of the high stopping power of the alpha particle. They typically have a density thickness of 1 mg/cm^2 which translates into a linear thickness of about a quarter of one thousandth of an inch (0.00025"). A wire mesh screen or metal slats are frequently used to reduce the chances of puncturing the screen. If it does become torn, emergency field repairs are possible. The torn section can be repaired by application of tape. Note that this consequently reduces the size of the sensitive area of the probe since the alpha radiation cannot penetrate the tape.

As a result of the high energy and short range of the alpha particles, they produce a rather large electrical pulse in the proportional detector. This means that good discrimination is possible against beta and gamma ray interference. Often, commercial instrument specification sheets show rejection of gamma ray pulses in fields with exposure rates of over 100 R/hr. If it is detected, the gamma interference produces a "hiss" in the earphones.

The other possible interference problem is caused by fast neutrons elastically scattering from counting gas molecules. The recoiling molecules will expend their energy in the gas and produce a pulse very similar to true alpha counts. In an actual monitoring situation this problem can be detected quite easily. The alpha probe is moved about 10 cm away from the surface being monitored. If the counts cease, then true alpha contamination is being detected. If the instrument still records the same count rate, then neutron interference is being recorded or else the probe face has become contaminated. Moving to a low background location will determine if this is the case. If the window is contaminated, it usually must be replaced. Practice has shown that it is usually not possible to satisfactorily decontaminate the thin alpha windows. Most modern alpha survey meters have an audio output available. This is in the form of a built-in speaker or earphone. If you are monitoring for contamination at

Nuclides	Acceptable Levels in dpm/100 cm^2	
	Total Fixed + Removable	Removable
U-nat, U-235, U-238, and decay products	5,000	1,000
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	500	20
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1,000	200
Beta-gamma emitters (no alpha) except Sr-90 and others listed above	5,000	1,000
H-3, C-14 except as DNA precursors	10,000	10,000

Fig. 4 - Acceptable DOE surface contamination release limits

counting rates near the allowed levels, the audio output is almost mandatory as the rate is too low for reasonable meter readings on the lowest range.

Since 1998, Subpart E, "Radiological Criteria for License Termination" of Title 10 Part 20 of the Code of Federal Regulations offers guidance as to acceptable surface contamination levels (release limits) for NRC licensees. These licensees can use the NRC developed "DandD Code" on a personal computer to determine acceptable limits. In the DOE community, acceptable surface limits are in the DOE Radiological Control Manual and 10 CFR 835, Appendix D. The Rad Con Manual table is reproduced in Figure 4. As can be seen, alpha surveys need to be able to detect 500 dis/min over 100 square cm for fixed and 20 dis/min over 100 square cm for removable.

Using manufacturer data, typically an alpha probe may have a sensitive area of 50 to 100 square cm and a total efficiency (cpm/dpm) of 25%. Thus, the expected count rate for 100 dpm spread uniformly over 100 cm^2 is about 12 counts per minute (a count every 5 seconds, on the average). Now, in the practical monitoring case, a large area is being surveyed. Even if the probe is moved rather slowly, if it encounters a small spot of contamination as it passes over, the meter will not have time to respond. This is caused by the "time constant" built into the ratemeter circuit. (The time constant is the length of time it takes for the meter needle to reach 90% of the final reading when the instrument is suddenly placed in a radiation field). On the low ranges, an alpha proportional survey meter has a time constant of several seconds. It has purposely been designed to respond that way so that the meter smooths out the fluctuations caused by the random decay of radionuclides. The meter averages the

incoming counts over the time constant period. Thus, the audio output is the only feasible way to find low level alpha contamination. The time constant is zero for the audio output, i.e., the earphone or speaker responds instantly to each individual count detected and processed by the circuitry. To survey properly, then, the probe is moved slowly over the surface with the audio output operating. If contamination is found, the meter can be used to estimate the activity by holding the probe stationary. Note that on high ranges, the time constants are made progressively shorter. It is easier to average the count rate when many pulses arrive in a short time.

It was mentioned earlier that air, propane and P-10 are used as counting gases. The advantages of each will now be covered. The propane proportional counter used a small cylinder of purified liquid propane as a source of counting gas. Both Ludlum and Eberline have discontinued their models. The tank was enclosed in the instrument case. The gas flowed through the probe and then was vented to ambient air. Proper gas flow was verified by igniting the gas at the vent and adjusting the gas flow by observing the height of the flame. A currently available P-10 floor monitor for alpha contamination is shown in Figure 5.

The air proportional counter uses a similar size probe which has a vent hole to allow for air exchange with barometric pressure changes. The chief physical difference from the propane model is the lack of the gas cylinder and associated plumbing.

Comparing the three types of alpha proportional counters, the air proportional counter is lighter in weight and much less fuss when setting up. However, the air counter is almost useless under conditions of medium to high humidity where the moisture gets into the probe and causes the high voltage to arc. The propane version is more stable under field conditions but the propane regulator and needle valve tended to plug up after long use. This was a major factor in their discontinuance by

Fig. 5 - A gas flow alpha-beta proportional counter

Courtesy, Ludlum Measurements, Inc.

manufacturers. (And there really are times when you literally "run out of gas.") P-10 is cleaner and a higher flow rate eliminates the plugged needle valve problems. In terms of efficiency, the air-filled instrument generally comes out below the gas-filled counters. This is due to the problem of air undergoing electrical breakdown before the counter reaches a good plateau operating point. A well designed propane-filled counter will have an efficiency of about 50% for 2π alpha emissions (i.e., it will count about 25% of the total disintegrations when placed close to a distributed alpha source). Since the air-filled counter is operated below the ideal voltage, the gas multiplication factor is smaller and the resulting alpha pulses are small. Therefore, the air-filled instrument needs higher gain amplifiers which can lead to the problem of "microphonics" (counting of noise pulses caused by mechanical shock and motion of the connecting cable). One final disadvantage of the propane-filled counter instrument is the result of propane being a flammable gas. This raises a possible problem from a fire safety point of view. In particular, the propane releasing counters should not be used in areas containing large quantities of plutonium. This metallic element is pyrophoric – it spontaneously ignites in air.

The other type of instrument that is commonly used for alpha surveys is the portable scintillation counter. Commercial instruments are available with rectangular windows of the same dimensions as the alpha proportional counters as well as instruments with circular entrance windows. (See Figure 6.) Again, the window thickness is extremely thin to allow the alphas to enter. A common detector probe assembly is shown schematically in Figure 7.

A commonly used alpha scintillator is silver-activated zinc sulphide, ZnS(Ag). This phosphor has been found to give large light pulses when struck by charged particles. The phosphor is usually coated directly on the surface of a piece of clear acrylic plastic. This material tends to confine light rays due to internal reflection at the surfaces and so it is termed a "light pipe." The photomultiplier tube then attaches to the light pipe and generates the signal sent to the instrument case. The phosphor coating is covered by a light-tight plastic entrance window. Also, the remainder of the probe

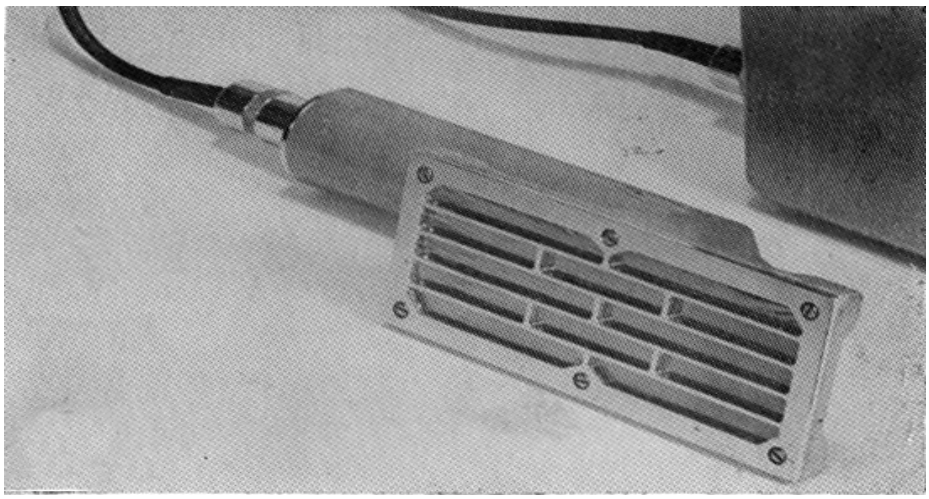


Fig. 6 - An alpha scintillator probe

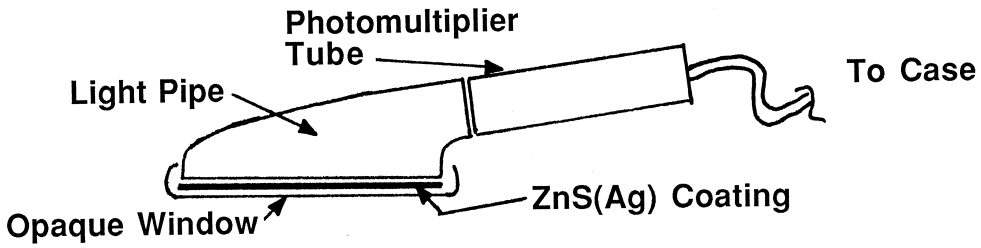


Fig. 7 - Details of an alpha scintillator probe

assembly must be contained within a light-tight case to prevent spurious signals from ambient room light.

As a rule, the alpha scintillation instruments are less sensitive to microphonics than the proportional counters. This is due primarily to the larger size signal produced by the scintillator compared to the proportional counter (volts vs. millivolts). On the other hand, the scintillation counter is less rugged. It is usually possible to continue monitoring with a proportional probe after accidentally dropping it, but the fragile photomultiplier assembly of a scintillation counter cannot survive large shocks. In terms of absolute alpha detection efficiency, the scintillator ranks between the air proportional counter and the propane proportional counter. It would typically count 30% to 35% of the incident 2π alpha particles.

The alpha scintillation counter is not sensitive to changes in humidity or barometric pressure as are the proportional counters. However, it does have two other disadvantages. It is very difficult to repair a torn entrance window under field conditions. The extreme light sensitivity of the photomultiplier tube requires that <u>all</u> extraneous light be excluded. Secondly, the alpha scintillator is much more sensitive to gamma ray interference than the alpha proportional models. A typical scintillation instrument will not be able to reject more than about 10 to 20 mR/hr (0.1 to 0.2 mSv/hr) of photon exposure rate. This could pose problems in some monitoring situations.

Before beginning a survey, check counter operation with the check source in a low background area. Deviations from the previous readings of the check source mean that the instrument selected is defective. Remember that it takes a minute or two for the propane or P-10 gas flow counter to flush out air molecules that have diffused into the counter since the last operation. When the instrument is operating properly, enter the survey location. The probe should be held <u>within one quarter of an inch of the surface</u> that is being surveyed. This is due to the very short range of alpha particles in air.

One popular technique to assure that the proper distance is maintained is illustrated by Figure 8. The glove protects the fingers of the technologist from alpha contamination. The fingertips are extended about one-quarter inch past the probe face to provide the fixed survey distance when the fingertips lightly contact the surface being surveyed. If the probe gets too close to the surface, it can become contaminated. If it is too far away, the alpha efficiency drops dramatically. Compared to direct

Fig. 8 - Technique to maintain a fixed distance from a surface

contact between an alpha source and the entrance window, a spacing of one-quarter inch will cause a loss of 25% for 5 MeV alpha particles while a half-inch spacing will cause a loss of 50% of those alphas.

In surveying the surface, move slowly. For alpha contamination scanning, the MARSSIM suggests a scanning speed of 3 to 5 cm per second. If a couple of close-spaced counts are heard in the audio output, stop and wait to see if it was a random background event or if it indicates low-level contamination. Try to minimize cable movement by coiling up the excess cable and then moving the instrument and probe together as much as feasible. If alpha contamination is found, lift the probe about 10 cm away from the surface to check for interference. Make frequent operational checks with the check source to confirm continued instrument performance.

There are two final suggestions for nonroutine situations. If the surfaces to be surveyed are highly irregular shapes or if they are wet or covered with a film of oil, make use of the wipe test instead of a survey instrument. The high stopping power of the alpha prevents it from penetrating an oil or water film. In the case of out-of-doors surveys, any areas containing weeds should be thoroughly stamped down first to reduce the possibility of puncturing the window.

Contamination survey instrument calibration is a bit trickier than for many other radiation protection instruments. This is due to the option of performing a "2π" or a "4π" calibration. Consider a hypothetical point alpha calibration source suspended in space. If the alpha probe is placed in contact with the source, on the average one-half of the disintegrations will send an alpha particle into the probe while the other half of the alphas emitted will travel away from the probe. See Figure 9. If the

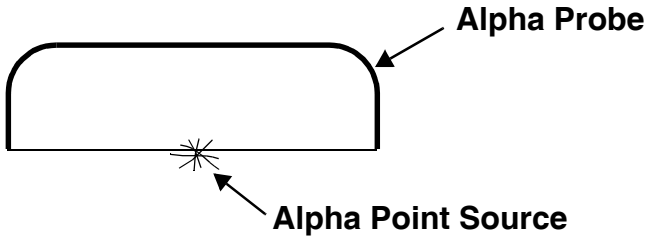


Fig. 9 - Calibration of an alpha survey meter probe

probe responded to all of the alphas hitting it, this instrument would count at a rate equal to one-half of the disintegration rate. But this is never the case. Some alphas are striking the window at such a glancing angle that they do not penetrate the window covering. However, the instrument calibration control can be adjusted to take this into account. If the survey meter is artificially set to read one-half the disintegration rate, then, the meter reading will represent 100% efficiency for the alphas hitting the detector probe. This is called a "2π" calibration. If the technologist is performing a MARSSIM survey, the release limits are specified in dpm/100 sq cm for surface contamination. In this case, the alpha meter is often adjusted to a "4π" calibration with 100% efficiency.

The terms "2π" and "4π" come from the field of solid geometry. The solid angle subtended by a point at the center of a sphere is 4π steradians. (This arises from the fact that the surface area of a sphere is 4π x the square of the radius. A unit sphere having a radius = 1 would have a surface area of 4π x 1^2 = 4π). A hemisphere subtends a solid angle of 2π steradians. Since we are counting all the alpha particles released into one hemisphere, this is just the 2π calibration. The 4π calibration would result if the meter were ARTIFICIALLY adjusted so as to read the disintegration rate of the point source. The instrument would then respond AS IF IT DETECTED EVERY ALPHA PARTICLE EMITTED BY THE SOURCE. Although this is physically impossible in a portable survey instrument, such an instrument would be a great CONVENIENCE when surveying for surface contamination. The calibration controls in most alpha survey meters purposely have a sufficient range of adjustment to allow either a 2π or a 4π calibration. The important point in this whole discussion is that the technologist MUST KNOW WHICH CALIBRATION WAS PERFORMED on a given instrument. The 2π vs. the 4π is a matter of personal preference. But if an instrument is used that has one type of calibration while the technologist thinks that it was calibrated the other way, an error of 100% will result in each reading. The two calibrations are summarized in Figure 10. When you have decided which to use, LABEL THE INSTRUMENT so other users know. See Sample Problem 1.

In choosing an alpha calibration source, backscatter is not a problem (the alpha particle mass is too large). However, the calibration source will not be "representative" if the surface being monitored has allowed the alpha activity to penetrate slightly or, as has been mentioned, is wet or oily. Since most alpha energies encountered in practice are in the 4 to 6 MeV range, and since most alpha survey instruments have a fairly energy independent response in this range, most any alpha with a

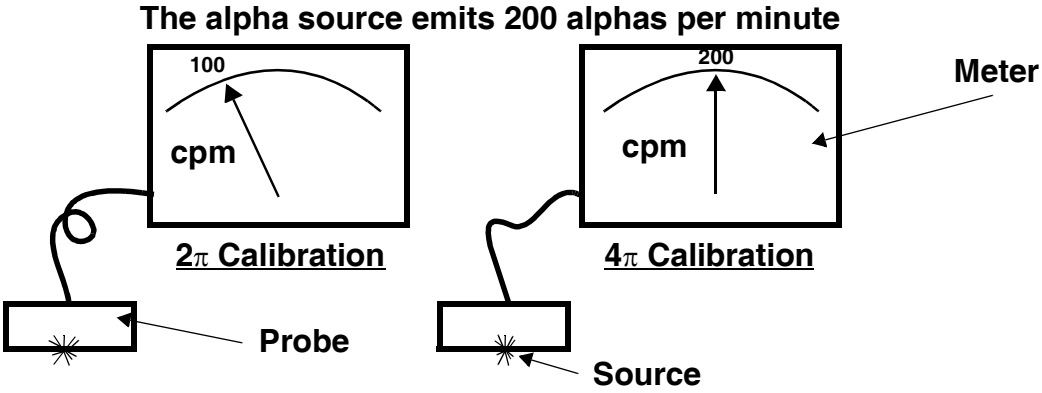


Fig. 10 - A 2π versus 4π alpha probe calibration

reasonable half-life will work. Popular radionuclides include Am-241, Pu-239 and Th-230. Under NCRP recommendations (NCRP Report 112), an alpha meter is considered capable of measuring surface contamination within ± 30% when properly calibrated. Remember that the air space between the probe window and the contaminated surface will absorb some of the alphas. Thus, the calibration should be performed at the same distance as that used for monitoring.

One final alpha calibration note. If a very small "point" source is to be used for calibrations, extreme care must be exercised to see that it is placed on the probe in such a way that one of the wires or metal strips of the protective screen for the window is NOT BLOCKING THE POINT SOURCE. Highly erratic readings are obtained in this situation. A large area "distributed" source is preferred for these calibrations.

Sample Problem 1

GIVEN:

An alpha scintillation probe with a 4π calibration reads 50 alphas per minute in contact with a small source.

FIND:

About how many alphas are actually being detected per minute?

SOLUTION:

A 4π calibration means that the source is emitting 50 per minute. Of these, only half, or 25, are traveling toward the probe. From the alpha scintillator section, the absolute efficiency of a the probe is about 33%, so 25/3 or about 8 alphas per minute are actually being detected by the instrument.

Gamma Radiation Fields

The technologist can be exposed to gamma ray fields from a variety of sources, including different types of machines and radioisotopes. Generally, radiation producing machines give a broad energy distribution of photons from almost zero energy up

Courtesy, Ludlum Measurements, Inc.

Fig. 11 - A Geiger Counter with a rotating "beta shield"

to the maximum rated machine energy. The photon field is not present when the power is turned off to the machine. Isotopic sources usually give off gamma rays at fixed energies. In contrast to machines, the photon field is still present when electrical power is turned off.

MARSSIM surveys of gamma fields usually are used to estimate contaminated soil concentrations of gamma emitters in open land areas. The detector is typically a NaI(Tl) scintillation probe. The probe response must be measured (cpm per μR/hr) for the radioisotopes being scanned and then a computer model is used to determine the exposure rate above a soil surface contaminated with a given picocurie per gram concentration of that radionuclide. These measurement procedures and scanning techniques are covered in detail in the MARSSIM.

Relative to the radiation protection technologist, gamma ray monitoring presents an external hazard. Two general types of portable survey instruments are commonly used – Geiger counters and ionization chambers. Each will be discussed now in detail.

The GM type of survey instrument usually uses a relatively thick-walled Geiger tube at the end of a coaxial cable that connects it to the instrument case. It is often supplied with a rotating or sliding "beta shield" as shown in Figure 11. The instrument case contains the batteries, a high voltage supply, a ratemeter circuit and a readout meter plus audio output jacks. The Geiger counter survey instrument normally does not require amplifier circuits since the huge gas multiplication factor in the tube produces large pulses that can be counted directly.

The ratemeter typically has three or four ranges which differ by factors of 10 in exposure rate intensity. The meter face is calibrated in terms of mR/hour, mSv/hr, or

mrem/hour. Usual full-scale ranges on a conventional Geiger counter (as opposed to special high range instruments) vary from 0.05 to 50 or 100 mR/hr.

Some commercial instruments have a time constant switch or control. As previously described, this is used for alpha proportional counters. The control is set with the time constant as short as possible for performing moving surveys. In this position, the instrument responds rapidly to changes in the field. For example, this would be particularly important in searching for radiation leaking through cracks in a shield wall. The time constant should be set as long as possible when calibrating the instrument or using it to obtain a precise reading at some fixed location. Again, note that the earphone or speaker audio output has a time constant of zero, i.e., the instrument responds instantly to radiation field changes.

In making surveys, once again the probe should normally be moved slowly to allow time for the instrument to respond. The performance check source should be used often to assure proper functioning of the instrument. If the photon field contains a large fraction of low energy gamma rays, then the manufacturer's specifications should be consulted to determine whether an energy correction factor will need to be applied to measured readings. As a result of the photoelectric cross section being high for low energy photons, the Geiger counter tube will overrespond in this region. (This behavior was discussed in Chapter 7.) The shield assembly provided by the manufacturer flattens out this overresponse, when the rotating shield is left closed. This is done by selectively filtering out lower energy photons more than medium energy photons.

The problem of Geiger counter saturation was also discussed previously in Chapter 7. This refers to the instrument reading near zero when placed in a field with an exposure rate well above the highest range on the survey meter. The long Geiger counter dead time can lead to the saturation condition. This poses a severe potential hazard to the radiation monitor. If such high fields can be produced in a given facility, care should be taken to assure that the Geiger survey instruments available are of the "non-saturating" type or else they should be replaced with ion chamber instruments.

Finally, there is a problem when trying to read exposure rates in a PULSED PHOTON FIELD with a GM survey meter. Two common examples of such fields are the prompt radiation field of many types of nuclear accelerators and the field associated with an x-ray machine. During the actual pulse, the exposure rate may be high. Between pulses the rate is background. Due to the long Geiger counter dead time, the Geiger survey meter will give one "count" for each pulse and thus READ THE MACHINE PULSE REPETITION RATE rather than the average exposure rate.

The air filled ionization chamber type of portable survey instrument is frequently used for gamma ray measurements. As a class, these instruments, have a very flat energy response. A typical response curve (e.g., Figure 11 in Chapter 7) shows that the meter reading is within 10% of the correct value from around10 keV to over 2 MeV. Standard ion chamber survey meters have full scale ranges running typically from 25 mR/hr to 50 or 500 R/hr. Thus, the ion chamber is less sensitive than the Geiger counter but it is able to accurately measure high field intensities.

As a result of the use of a DC coupled electrometer circuit, the majority of ion chamber survey meters still in use are subject to "zero drift," especially on the more sensitive ranges. This means that in the absence of a radiation field they will slowly indicate up to 20% or 30% of the full scale reading either above or below the zero position on the scale.

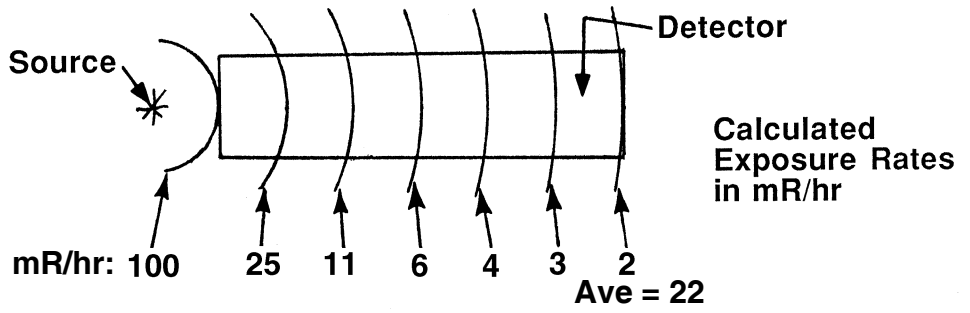


Fig. 12 - Non-uniformity caused by source being too close to detector

These instruments are thus provided with a zero setting control or controls (some have a coarse and a fine control). If the particular instrument chosen lacks a zero set button (it disconnects the chamber from the circuit so the zero can be adjusted even while in a radiation field) it is best to put the range switch on the highest range and then adjust the meter to read zero. When completely warmed up (typically 10 minutes) the zero setting becomes more stable and only needs occasional adjustment.

In surveying a pure gamma ray field at a large distance from the source, the result is expressed as a field reading in mSv/hr or mR/hr. If the instrument is being used to measure surface contamination from gamma emitters, the chamber is placed as close to the surface as possible without touching. The reading obtained with the chamber almost touching the surface is often reported at two inches (5 cm). This is a typical effective center distance for an ion chamber with a large sensitive volume. When placed close to a radiation source the inverse square fall-off of the field intensity with increasing distance from the source means that the chamber gas is being subjected to a very non-uniform radiation field. The gas close to the window (source) receives a much stronger dose rate than the gas at the far end of the ion chamber. See Figure 12 for an illustration of this effect. The actual meter reading represents an average for the gas contained inside the total ion chamber volume. Another way to think about what is happening is to realize that the meter is reading the value at the effective center of the chamber. Since this point is located inside the detector, a correction factor would have to be applied to determine the exposure rate at the window, i.e., the contaminated surface.

One final problem can produce readings which are greatly in error. If the ion chamber is exposed to a highly collimated beam of radiation so that only part of the chamber volume is irradiated, then the instrument will read low by an amount equal to the ratio of the volume exposed to the total chamber volume. For example, this situation would occur in measuring radiation levels behind a shield wall composed of stacked concrete blocks. The cracks would produce a narrow beam which would only be intercepted by a small fraction of the chamber volume as illustrated in Figure 13. Sample Problem 2 gives a quantitative example of this behavior.

On occasion, gamma radiation levels near background must be measured accurately. In this case, the instrument of choice is a microrem meter. As mentioned in Chapter 7, the detector is usually a scintillation crystal. By using a solid state device instead of a gas-filled counter, the sensitivity is increased. Typically, the most

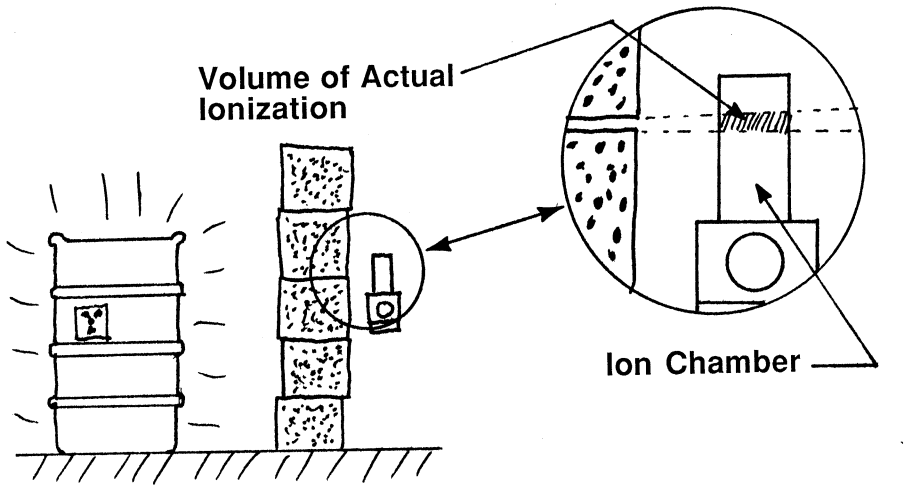


Fig. 13 - Low reading due to partial irradiation of chamber

sensitive range in a commercial microrem survey meter is 0 to 20 or 25 μrem, about 10 times more sensitive than commercial Geiger counters. Most instruments utilize internally mounted NaI(Tl) crystals and, thus, have poor energy response due to the high Z_{eff} caused by the 53 atomic number of iodine. The Bicron Corporation had a breakthrough design in their MicroRem™ model. It uses a tissue equivalent organic scintillator which gives a very flat energy response. Figure 14 shows the unit and Figure 15 gives its energy response curve. (Thermo Electron now markets this meter.)

In calibrating both GM counters and ion chambers, the usual calibration source provides a known exposure rate at some distance. However, recommendations of international organizations and regulations of the U.S. NRC require a knowledge of the dose equivalent rate at 1 cm depth in soft tissue. Conversion of roentgen rate readings to rem rate or sievert rate readings can be done using the data in Figure 12 of Chapter 5. Common isotopes used for calibrations include Co-60, Cs-137 and Ra-226. From the source calibration certificate the activity is calculated at the time of calibration using the exponential decay law. Then, the specific exposure rate constant

Sample Problem 2

GIVEN:

An ion chamber with 15% of its volume irradiated reads 0.2 mSv/hr.

FIND:

What is the actual radiation intensity in the beam?

SOLUTION:

The 0.2 mSv/hr is from irradiation of only 15% of the gas. If all the gas were irradiated uniformly, the intensity in the beam would be 0.2 mSv/hr x 100% ÷ 15% = 1.3 mSv/hour.

Fig. 14 - A tissue equivalent microrem meter

Courtesy of Bicron Corporation

can be used to give the exposure rate at any reasonable distance from the source. (Figure 17 of Chapter 5 lists rate constants for popular sources.) Deviations from the inverse square law caused by scattered photons can be minimized by keeping the calibration area free of other apparatus, using a large room and by keeping the source to detector distance small. If a survey meter is going to be used for low energy x-rays, it is usually calibrated with a filtered x-ray machine or low energy photon emitters. Sample Problem 3 deals with gamma calibrations.

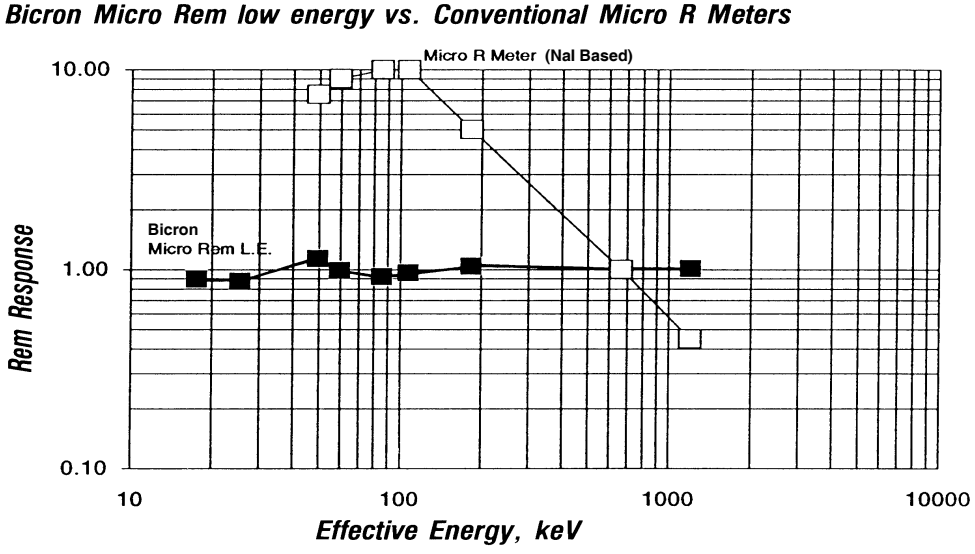


Fig. 15 - Energy response of Bicron micro rem™ instrument

Courtesy of Bicron Corporation

Sample Problem 3

GIVEN:

A miniature GM counter is to be calibrated, at 0.5 meter distance, to read dose equivalent rates up to 2 mSv per hour.

FIND:

What activity of Cs-137 is required?

SOLUTION:

To calibrate the highest range, the source must put out 2 mSv/hr at 0.5 meters. This distance is assumed to meet the "3 times the longest dimension" rule for a miniature GM tube. From Chapter 5, Fig. 17, the specific rate constant for Cs-137 is 8.0 x 10^{-8} μSv-m^2/hr-Bq. From Chap 5., Fig. 15, H/t = Γ A/r^2. Thus, A = (H/t) r^2 / Γ = 2 mSv/hr x 1000 μSv/mSv x 0.5^2 m^2 / 8.0 x 10^{-8} μSv-m^2/hr-Bq = 6.3 x 10^9 Bq = 6.3 GBq.

The principle of uniform irradiation of the detector volume must be kept in mind during calibration of an ion chamber instrument. The problem becomes acute when the gamma ray calibration source is too weak, i.e., the distance between the source and the calibration point is less than about three times the chamber's longest dimension (the "magic" rule for definition of a realistic point source in Chapter 5). This often occurs when trying to calibrate the highest ranges with a low activity calibration source. To minimize the non-uniformity in the field throughout the chamber volume, always calibrate the instrument with the long axis of the chamber pointing perpendicular to the source, even though it doesn't look like the "natural" way to point some meters. Figure 16 illustrates this. Strictly speaking, this would be the correct position to hold the instrument for making field readings, too. In most practical cases the field is uniform enough so that chamber orientation is not a problem.

There is a case where it is possible to place the point calibration source very close to the chamber for calibrating the high ranges. This means the chamber volume will be exposed to a non-uniform field. BUT, by calculating the amount of non-uniformity, it is possible to apply a "correction factor" to adjust the meter reading in compensation for the non-uniformity. This is achievable if the source is placed along the major axis of a

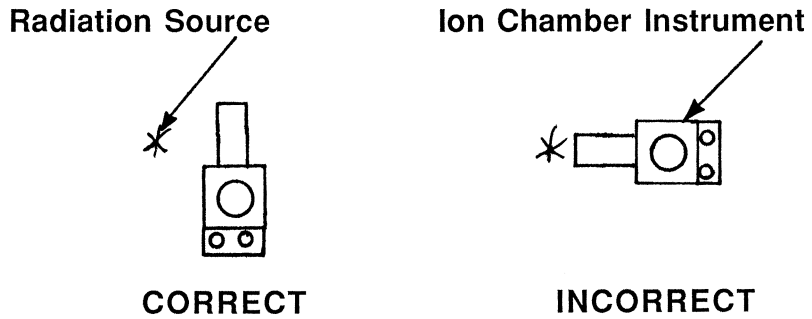


Fig. 16 - A long detector should be oriented perpendicular to a source

Z(cm)	E	Z(cm)	E	Z(cm)	E	Z(cm)	E	Z(cm)	E
7.00	0.3740	11.00	0.6904	15.00	0.8231	35.00	0.9660	135.00	0.9977
7.20	0.4068	11.20	0.6998	16.00	0.8433	40.00	0.9739	140.00	0.9979
7.40	0.4318	11.40	0.7088	17.00	0.8604	45.00	0.9794	145.00	0.9980
7.60	0.4536	11.60	0.7175	18.00	0.8749	50.00	0.9833	150.00	0.9981
7.80	0.4735	11.80	0.7259	19.00	0.8873	55.00	0.9862	155.00	0.9983
8.00	0.4920	12.00	0.7339	20.00	0.8979	60.00	0.9884	160.00	0.9984
8.20	0.5095	12.20	0.74 16	21.00	0.9071	65.00	0.9901	165.00	0.9985
8.40	0.5262	12.40	0.7490	22.00	0.9152	70.00	0.9914	170.00	0.9985
8.60	0.5421	12.60	0.7561	23.00	0.9222	75.00	0.9925	175.00	0.9986
8.80	0.5574	12.80	0.7629	24.00	0.9285	80.00	0.9934	180.00	0.9987
9.00	0.5720	13.00	0.7695	25.00	0.9340	85.00	0.9942	185.00	0.9988
9.20	0.5861	13.20	0.7758	26.00	0.9389	90.00	0.9948	190.00	0.9988
9.40	0.5996	13.40	0.7819	27.00	0.9432	95.00	0.9954	195.00	0.9989
9.60	0.6126	13.60	0.7877	28.00	0.9472	100.00	0.9958	200.00	0.9989
9.80	0.6251	13.80	0.7934	29.00	0.9507	105.00	0.9962	205.00	0.9990
10.00	0.6371	14.00	0.7988	30.00	0.9539	110.00	0.9965	210.00	0.9990
10.20	0.6486	14.20	0.8040	31.00	0.9568	115.00	0.9968	215. 00	0.9991
10.40	0.6597	14.40	0.8090	32.00	0.9594	120.00	0.9971	220.00	0.9991
10.60	0.6703	14.60	0.8139	33.00	0.9618	125.00	0.9973	225.00	0.9992
10.80	0.6805	14.80	0.8186	34.00	0.9640	130.00	0.9975	230.00	0.9992

Fig. 17 - Correction factors for non-uniform gamma ray exposures (14 cm long by 7.3 cm diameter chamber, ONLY)

cylindrical ion chamber. For this geometry, the calculus allows the average exposure rate to be computed by performing a volume integral over the chamber. The numerical results of such a computation are shown in Figure 17 for the standard Cutie Pie (14 cm long ion chamber with a 7.3 cm diameter). Consult the original paper for the formula that allows calculation of the correction factors for other sizes.

To use this information for calibration, first calculate the specific exposure rate (mR/hr at 100 cm) for the source activity decay corrected to the day calibration is to be done. Place the calibration source at a distance Z, measured in cm, away from the geometric center of the chamber, centered on the axis of the chamber. Calculate, from inverse square law, the mR/hr at the selected distance Z. Look up the correction factor, E, from Figure 17. Adjust the calibration pot until the meter reads 1/E times the calculated mR/hr. (In other words, the average field for the non-uniform irradiation conditions is 1/E x calculated mR/hr.) See Sample Problem 4 for an example.

Sample Problem 4

GIVEN:

A point gamma source is placed 2 cm in front of the window of a Cutie Pie.

FIND:

What correction for non-uniformity must be applied to assay this source?

SOLUTION:

From the text, a Cutie Pie has a 14 cm length. The chamber geometric center is 7 cm from the window, so the point 2 cm in front of the window has the value Z = 7 + 2 = 9 cm. From Fig. 17, the correction factor E = 0.572 @ Z = 9. The average field = 1/E x indicated field = 1/0.572 = 1.75 x indicated field. The assay is made by taking 1.75 x chamber reading to be the correct exposure rate at 9 cm from the point source. The Γ A /r^2 point source rule is then applied to find A.

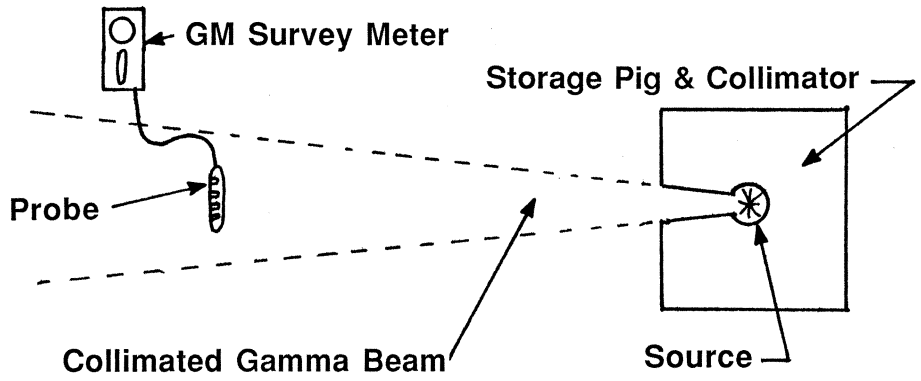


Fig. 18 - Geiger counter calibration setup

The Geiger counter survey meter is usually calibrated by use of relatively large activity gamma ray sources at distances which allow the source to be treated as a point source. The Geiger tube should be oriented so that the long axis of the tube is perpendicular to the incident photon beam. A lead collimator is frequently designed as a radiation safety feature to restrict the photon field to a narrow cone into which the tube is inserted for calibrations. The correct setup is shown in Figure 18.

When calibrating the lower ranges of the survey meter, remember that the manufacturer has placed a substantial time constant in the circuit to smooth readings. It takes the meter about 6 time constants to reach 99% of the final reading in the constant calibration field. Let it have the time needed!

The use of a collimated beam type calibrator calls for additional comment. Commercial calibrators are available with beam shutters or attenuators that enable a wide range of "known" field intensities to be produced without changing the source to detector distance. These devices are very handy for general routine calibrations but cannot be relied on for high accuracy fields, particularly under conditions of high attenuation (low field intensity). When a commercial unit was tested by Pacific Radiation, it was found that the "25%" shutter was low by 4%, the "10%" shutter was low by 6%, the "1%" combination was high by 4% and the "0.25%" combination of shutters was high by 44%!

Besides the accuracy problem at high attenuation, these calibrators have a second problem. Both the collimator walls and the attenuators will introduce lower energy scattered photons into the primary beam. If the survey meter being calibrated has an energy dependent response, a correction factor will have to be determined and applied to get accurate results. Again, this problem is most serious at the higher attenuation settings. In most cases, it is possible to measure the true field intensity, transmitted by the various combinations of attenuators, with a secondary standard type of laboratory dosimeter if high accuracy is necessary.

Beta Radiation Monitoring

Known beta emitting radionuclides cover a wide range in energy. Most have accompanying gamma rays, although a few "pure" beta emitters exist. As a general

rule, beta ray monitoring presents the potential for both an internal and external hazard to the technologist. This is due to the fact that two totally different types of "beta monitoring" exist. The internal hazard results from the real possibility of loose beta contamination becoming airborne or transferred to objects from which ingestion occurs. Beta contamination in MARSSIM surveys is measured in units of activity per unit area, e.g., dpm per 100 sq cm. It would typically be measured by a pancake GM or proportional counter survey meter.

Beta fields present an external hazard in terms of the possibility for skin and lens of the eye exposure. (Remember, the range in air is 4 meters per MeV.) In reporting the results of beta field measurements, absorbed dose or dose equivalent units must be used as the roentgen is defined only for photons. Thus, a beta field is reported in millirads per hour. It would typically be measured by an ion chamber.

Technologists sometimes make the mistake of trying to measure beta fields with a Geiger survey meter. The usual Geiger tube that is supplied with a general purpose survey meter has a wall density thickness of about 300 mg/cm^2. From the beta particle range-energy curve in Chapter 11 Figure 9 it can be seen that a beta would have to have an energy in excess of 800 keV just to penetrate the wall. For the average beta to be detected, as a result of the shape of the energy distribution for an isotopic beta source, the source would need to have an endpoint energy of 3 x 800 keV or 2.4 MeV. This criterion would eliminate 93% of the known beta emitters listed in the Radiological Health Handbook. The thin wall tubes have a density thickness of about 40 mg/cm^2. The necessary endpoint energy so that the average beta would be detected from an isotopic source becomes about 600 keV. Only about one-half of the known beta emitters would meet this criterion.

Assuming that the radiation protection technologist was fortunate enough to have a high energy beta and a thin-walled Geiger tube, the Geiger counter STILL cannot quantitatively measure the dose rate or the dose equivalent rate in a beta field. This is due to the fact that the Geiger counter is operating on the uppermost plateau of the characteristic curve for gas-filled detectors (refer to Chapter 7, Figure 5). In this region, the gas multiplication is so large that any ionization produced in the tube leads to complete discharge, thus eliminating any energy information. In other words, the Geiger counter is calibrated in terms of the exposure rate measured in roentgens/hour. But the roentgen is only defined for photons in air. Thus, while the beta shield can be opened to allow betas to enter, very little information is gained by opening the shield. In fact, with the shield open, the instrument no longer has a flat energy response for gamma rays. In addition, it will detect low energy photons now. Clearly, some instrument that responds to actual energy deposited is needed to measure beta ray fields.

We will now discuss beta contamination measurement, and then will return with details on beta field measurements. MARSSIM suggests two types of detectors are satisfactory - the pancake GM and a beta proportional counter. The efficiency of both is approximately 20%. The background rate in the pancake GM is typically 40 to 60 cpm (2" diameter). The usual beta proportional counter has a larger probe area, ranging from 100 sq cm up to 600 sq cm. The background rate varies from 300 to 1500 cpm over this size range. Beta scintillators with 125 square cm windows are also available and suitable.

An important concept emphasized in MARSSIM is the MDC_{Scan}, i.e., what is the smallest concentration of contamination, in dpm/100 sq cm, that a reasonably trained technologist might be expected to find in a beta contamination scanning survey? The MDC_{Scan} depends on the probe efficiency, size, background and scan speed. Using parameter values accepted by the NRC, the MARSSIM formula for surfaces is:

$$MDC_{scan} = \frac{1512\sqrt{R_B}}{\sqrt{i}\,\varepsilon A} \quad \textbf{(dpm/100 sq cm).} \qquad \textbf{[Eqn. 1]}$$

In the formula, the background count rate, R_B, is in cpm, the efficiency, ε, is counts per disintegration and A is the probe area in sq cm. The term "i" in the square root in the denominator is the interval of time, in seconds, that the probe is over a spot of contamination while scanning, i.e., the scan speed. For example, if the probe is 10 cm in size in the direction of movement, and it is moved at a speed of 5 cm/second, "i" is 2 seconds. The efficiency term in the equation should include the effects of backscatter of the beta contamination off of the surface (which increases the count rate) and self absorption in any overlying dust or surface coatings (which reduces the count rate). MARSSIM defines ε = Instrument Efficiency X Surface Efficiency to include these effects. If the calculated MDC_{Scan} is higher than the cleanup guideline value, then some elevated contamination spots will be missed by your scan survey. The solution is, of course, to increase "i", by slowing down the scan speed. Note that the above formula takes into account human performance factors such as training, experience and fatigue of the technician (included within the "1512" factor). See Sample Problem 5 for an example.

For the sake of the curious, the composite conversion factor of 1512 is calculated as follows. Each of the numbers on the right-hand side are taken directly from Section 6.7.2.1 of the MARSSIM.

$$1512 = \frac{1.38 \times \sqrt{1/60} \times 60 \times 100}{\sqrt{0.5}}.$$

Before covering beta <u>field</u> measurement, there is one additional topic that is very useful in practical contamination monitoring operations. Frequently, the technologist is measuring beta contamination of unknown origin. In this case, it would clearly be useful to have a measurement of the beta energy to aid in identifying the

Sample Problem 5

<u>GIVEN</u>:

A 20 sq cm pancake GM probe with a 60 cpm background rate is used to scan for Tc-99 on a concrete floor. The measured efficiency in this case is 9.9% and the scan speed is one probe width per 2 seconds.

<u>FIND</u>:

The scanning minimum detectable concentration, MDC_{Scan}.

<u>SOLUTION</u>:

The probe will spend 2 seconds over a contamination spot so i=2. Plugging the rest of the values into the formula,

$MDC_{Scan} = (1512\sqrt{60})/(\sqrt{2} \times 0.099 \times 20) = 4183$ dpm/100 sq cm.

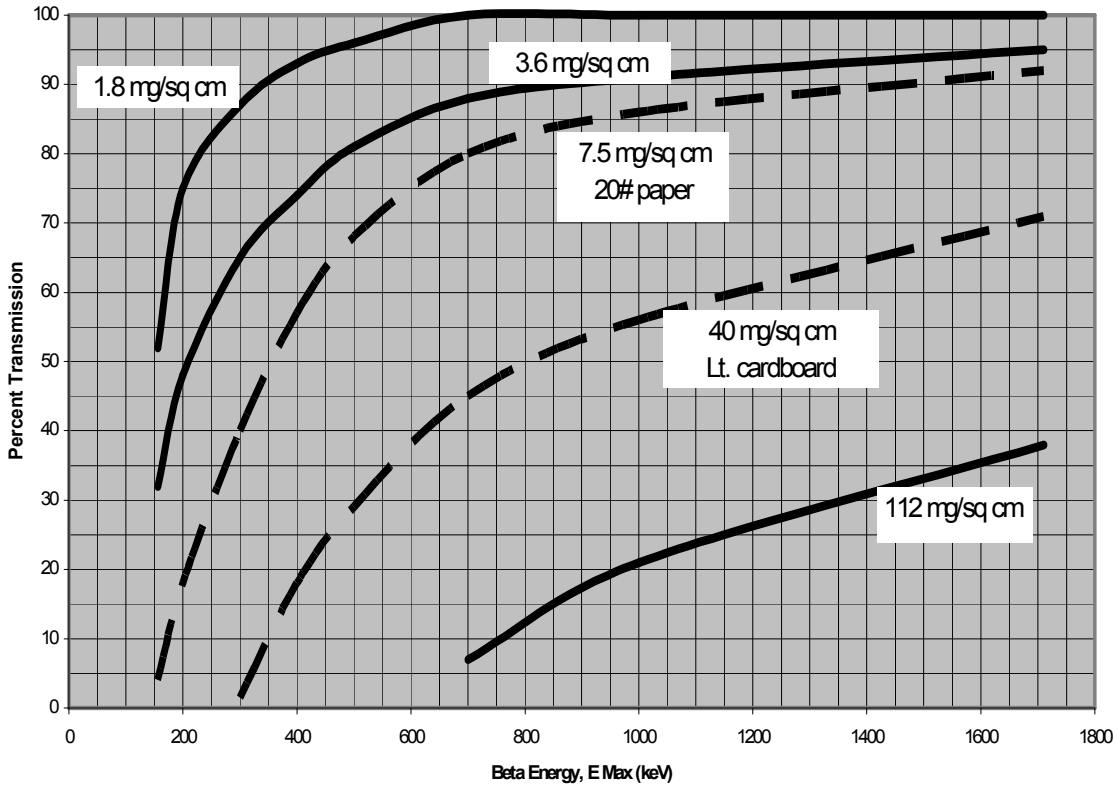


Fig. 19 - Beta transmission vs. E_{Max} for 2" diameter pancake GM tubes

nuclide. This might easily be sufficient for identification of the radionuclide in the contaminant if a licensee possesses only a few beta emitters. Pacific Radiation sometimes determines beta E_{max} energies using a 2" pancake GM probe with an assortment of absorbers. Figure 19 shows the percentage transmission for a "typical" pancake detector through different thicknesses of absorbers. (% transmission is the ratio of the count rate shown on the meter with the absorber inserted between the source and detector to the count rate reading with no absorber.) The Figure 19 data was taken in the Pacific Radiation lab. To determine the validity of the data, i.e., how "typical" are commercial pancake probes, we repeated the measurements for five different pancake GM probes from four different manufacturers. The various probes were all within about 5 percentage points of each other regarding beta transmission. Note that the "thicknesses" of the absorbers are in units of density thickness discussed near the end of Chapter 3.

Secret trick for savvy technicians: Looking at Figure 19, you can see that the whole range of beta E_{max} energies from 150 keV, e.g., carbon-14 up to 1700 keV, e.g., phosphorous-32, can be distinguished using <u>only</u> two absorbers, the 7.5 mg/sq cm and the 40 mg/sq cm. (These are shown as dashed lines in Figure 19.) It happens that the 7.5 mg/sq cm absorber is 20# bond copy paper and the 40 mg/sq cm absorber is a single sheet of

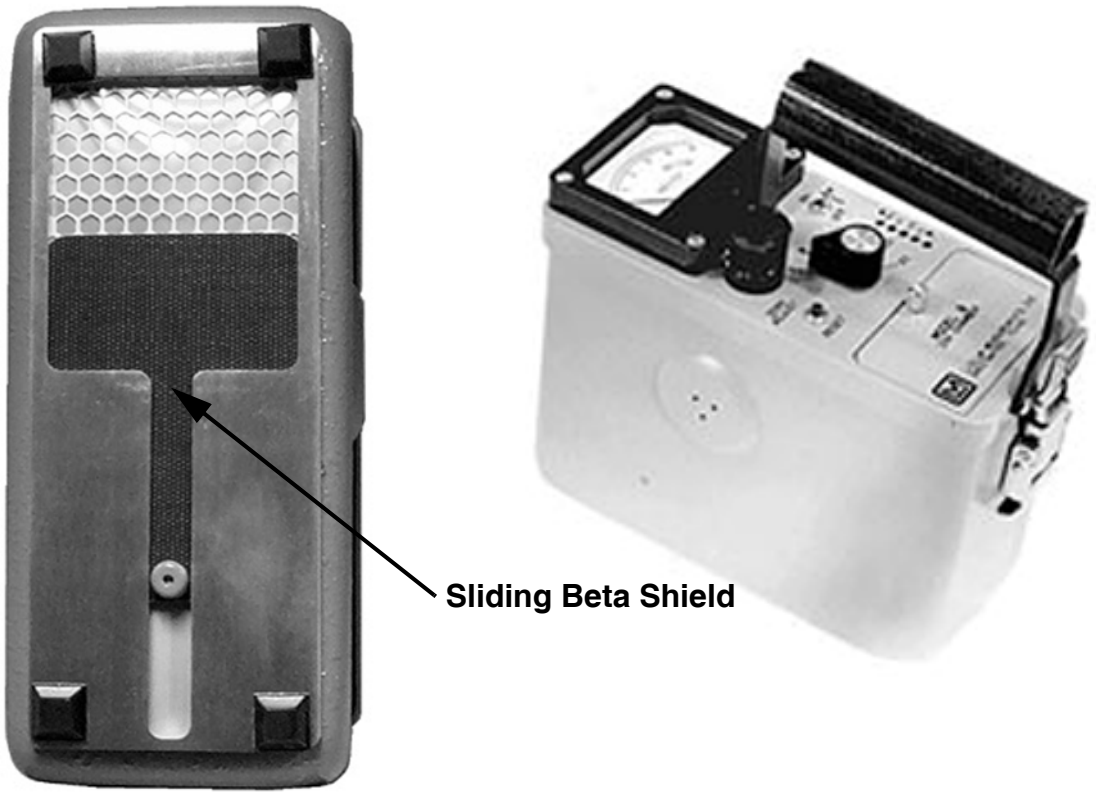
the lightweight cardboard found at the back of a pad of paper. (To verify the correct density thicknesses: for the paper, check your office supply store and read the packages to find one that states 75 grams per square meter, i.e., 7.5 mg/sq cm. For the cardboard, take some 8½" by 11" samples into your post office. Pick the one that weighs the closest to 0.85 ounces for the whole sheet.) Then, take a bare contact reading of the beta source and two additional readings with the paper in and the cardboard in and find the E_{max} energy from Figure 19.

A final note of caution - if the beta source/contamination contains betas with more than a single E_{max} present, this method won't work. Lower E_{max} betas will be absorbed more quickly than predicted by Figure 19.

Returning now to beta field measurement in contrast to contamination measurement, the proper instrument is the ion chamber. Beta field measurement is not part of the MARSSIM. An ion chamber can be used because the electrical signal produced is directly proportional to the energy deposited in the chamber gas. Most ion chamber instruments have a thin beta window that allows even low energy betas to penetrate. For gamma surveying, the window is covered with a beta shield cap. (See Figure 20). Unfortunately, the ion chamber instrument is calibrated for photon surveys in a uniform field. This means that the chamber gas is exposed to a relatively uniform gamma field of known exposure rate and the meter is then adjusted to read the correct value. However, in measuring a beta field originating from surface deposited beta contamination, the problem previously illustrated in Figure 12 again appears. If the chamber is placed so the beta window is almost touching the surface, it will be exposed to a very non-uniform field. The meter will read some average for the chamber, i.e., it will read the correct value at the effective center of the chamber. However, this is not the desired result. We really wish to know the dose rate (in mrad/hr or milligray/hr) in direct contact with the surface. For example, this would be the dose rate that skin would receive if a worker came in contact with the contamination. In order to obtain the desired result, it is necessary to multiply the ion chamber reading by a beta correction factor, CF, to obtain surface contact beta dose rates. The determination of the correction factor will be covered shortly.

Calibration sources for beta fields are usually assayed with an extrapolation chamber (covered in Chapter 7). For sources to be used to calibrate instruments to read shallow dose equivalent (skin dose), the extrapolation chamber is fitted with a 7 mg/cm^2 entrance window. In the case of eye dose equivalent measuring instruments, a 300 mg/cm^2 entrance window is selected to meet legal requirements. Both the U.S. NRC and the DOE define skin and eye dose in the same way. A list of some commonly used beta calibration sources is given in Figure 21.

Since most beta detecting instruments are energy dependent, a range of different energy sources should be available. In the past, it has been common practice to use the E_{max} energy of the radionuclide as the designated energy of the "source." This can be a problem, particularly in the lower energy range. The source window, the matrix that holds the radioactivity and scatter from within the source container or holder can all degrade the emitted energy spectrum. Thus, the NCRP, in their Report 112, recommends that a new parameter, the residual maximum beta energy, E_{res}, be used to characterize the beta source. (See the "Other Resources" list at the end of this Chapter for more details.)



Courtesy, Ludlum Measurements, Inc.

Fig. 20 - An ion chamber with sliding beta shield

The actual correction factors (CFs) for beta surface contamination depend on the chamber geometry (size and shape), the chamber wall thickness, and strongly on the size of the beta source. They are expressed in units of mrad/mR. In the case of the cylindrical Eberline RO-20 ion chamber with a diameter of 7.6 cm and a length of 4.6 cm, the correction factor at the window is about 3.3 mrad/mR for a surface-deposited

Radionuclide	Max Energy, MeV	Half-life
Carbon 14	0.156	5730 yr
Promethium 147	0.225	2.6 yr
Technetium 99	0.294	2.1×10^5 yr
Strontium 90	0.546	28.6 yr
Chlorine 36	0.710	3.0×10^5 yr
Thallium 204	0.763	3.8 yr
Bismuth 210	1.16	5 day (22.2 yr parent)
Yttrium 90	2.28	64 hr (28.6 yr parent)
Depleted Uranium	2.28	4.5×10^9 yr

Fig. 21 - Some commonly used beta calibration sources

uniform uranium beta source spread over an area greater than the beta window diameter of the instrument. As an example, assume that a uranium solution has spilled over a bench top. After evaporation, a reading is made with an RO-20, in the "open window" configuration, held as close as possible without touching the bench top. If the net instrument reading is 5 mR/hr, the results would be reported as a "contact reading" of 16.5 mrad/hour of beta (5 mR/hr x 3.3 mrad/mR = 16.5 mrad/hr). Since the quality factor is 1 for betas of all energies, this could equally correctly be reported as a contact or surface reading of 16.5 mrem/hr of beta.

In the case of spots of beta contamination which are smaller than the window diameter, the CF becomes much larger. Typical values for a 0.5 liter volume Cutie Pie reach 100 for a 1/2" spot size measured in contact. Clearly, when monitoring close to a surface, a single correction factor just won't do the job if contaminated spots are present.

Since all of these correction factors only apply for the particular type of ion chambers identified, it is best to measure the appropriate factors at a given radiation facility if spots of beta contamination are routinely monitored. This can be done fairly easily by either making up a set of standard beta sources of different diameters or by obtaining a large uniform beta source and then using a thin steel plate with different diameter holes as a "mask" to measure the correction factors for various spot sizes.

The fundamental problem causing ion chambers to exhibit poor beta energy and directional response is the non-uniformity of the irradiation of the filling gas. The extrapolation chamber discussed briefly in Chapter 7 gives the key to the design of an "ideal" beta ion chamber. If the chamber is constructed so the gas is in the shape of a wide-diameter, short-height cylinder (a flattened pancake) then betas of a wide range in energy will all deposit the same energy in passing through. Lawrence Livermore National Laboratory has published work (see "Other Resources") on a simple modification to commercial ion chambers to greatly improve the beta response. By filling most of the chamber volume with inert plastic foam, the sensitive volume is reduced to a thin layer of air just inside the window, in effect creating an extrapolation chamber without the micrometer adjustment of the real thing. Now the correction factor becomes 1.0 within a few percent independent of beta energy or distance from the detector. Of course, the disadvantage of this modification is that the instrument sensitivity is reduced by decreasing the sensitive volume of the detector.

Mixed Beta-Gamma Radiation Monitoring

A common problem is to have surface contamination consisting of both beta and gamma ray emitters. The first problem for the technologist is to decide whether the survey is trying to measure the concentration of the contaminant (dpm/100 sq cm) using MARSSIM procedures or is the radiation field near the contamination (millirads/hr) the desired result. If the answer is the concentration, then a pancake GM probe or a beta proportional counter would be appropriate. The probe would have to be calibrated uniquely for the beta-gamma emitters found in the contamination. Once

the efficiency was determined, in counts/disintegration, the scan MDC and scan speed are calculated as previously discussed in the beta monitoring.

Since the ionization chamber is the only common instrument which reads energy deposited, it is the instrument of choice for beta-gamma field monitoring. As a practical problem, the gamma reading must be significantly smaller than the beta + gamma reading or else large uncertainties are introduced. The basic procedure is to take two readings – one with the window closed and the other with the window open. When the beta shield is in place (window closed), it shields the chamber volume from any beta component in the field. Thus, the ion chamber reads the gamma ray exposure rate associated with the contamination (mR/hr at the effective center). When the beta shield is removed (window open) then ionization is produced in the chamber by both photons and beta particles. Therefore, subtracting the two readings gives the beta component. Remember that the ion chamber is calibrated for uniform gamma exposure rate. Thus, the beta correction factor, CF, must be applied to the beta component to get the beta dose rate. See Sample Problems 6 and 7. In summary:

- **Window Closed = Gamma mR/hr @ effective center of chamber**
- **(Window Open minus Window Closed) X CF = Beta mrad/hr, contact.**

Sample Problem 6

GIVEN:

Eberline RO-20 readings in contact with a hot cell floor show 20 mR/hr window closed and 50 mR/hr window open.

FIND:

How are these survey results correctly reported?

SOLUTION:

The window closed reading gives the gamma component and is reported as 20 mR/hr at 3 cm (the approximate effective center distance of the RO-20). The beta component is obtained from the difference, 50 - 20 = 30 mR/hr. This fraction of the reading must then be multiplied by the correction factor. The approximate CF is 3.3 for the RO-20, so, the second reported value is 3.3 mrad/mR x 30 mR/hr = 99 mrad/hr, contact of beta dose rate.

Neutron Radiation Fields

Radiation surveys of fields with a neutron component are usually the most difficult problem facing the technologist. Neutrons are only indirectly ionizing. Most existing commercial neutron survey instruments (including “rem-meters”) have a significant energy dependence. Due to scatter from the air and surrounding walls or ground, the neutron field is never monoenergetic so that a simple correction factor cannot be applied for the energy dependence. In addition, neutrons are almost always accompanied by a photon field as well. In terms of the radiation protection technologist, neutron field monitoring presents only an external hazard.

Sample Problem 7

GIVEN:

A Cutie Pie reads 50 mR/hr window closed and 100 mR/hr window open in contact with a one inch diameter hot spot. The measured CF is 35 for a 1" spot, in contact.

FIND:

How are these readings interpreted?

SOLUTION:

The gamma results are just the window closed reading, at the effective center (about 5 cm in a Cutie Pie), i.e., 50 mR/hr at 5 cm, gamma. The beta reading is again obtained from the difference, 100 – 50 = 50 mR/hr in this case. Since the CF for a 1" spot in contact is 35 mrad/mR, the beta field component is reported as 50 mR/hr x 35 mrad/mR = 1750 mrad/hr, contact, beta.

Different instruments and techniques are used for monitoring different energy neutrons. In the case of thermal and slow neutrons (energies up to a few eV), a boron trifluoride proportional counter, BF_3 tube, is frequently used. This detector exhibits rejection of the associated gamma ray fields up to exposure rates of several 10s of mSv/hr (a few R/hr). The detector tube is exposed bare to the field. It can be calibrated to read directly in slow neutron flux (n/cm^2-sec). As previously discussed in Chapter 5, the flux can be converted into a dose equivalent rate in mSv/hr through use of the conversion factor of 27,200 n/cm^2-sec per mSv/hr. In very high fields, the proportional detector might experience significant losses due to the dead time, so foil activation techniques are used. Gold or indium metal foils can be placed in the field and the induced radioactivity is then counted to determine the neutron flux.

In the case of fast neutrons (broadly meaning a few eV to about 10 MeV), use is made of a moderated thermal neutron detector. The common moderators include wax and polyethylene in the form of cylinders or spheres. The fast/slow neutron survey meter makes use of a boron trifluoride proportional counter inside a cylindrical moderator with an outside thermal neutron shield. It was pictured in Chapter 7, Figure 25. The instrument is unusually energy dependent and so provides more of an indication of the presence of fast neutrons than actual flux values. The "remball" type of neutron survey instrument uses a 9" to 12" diameter spherical moderator with a BF_3 tube or thermal neutron scintillator at the center. (See Figure 22). It is designed to give reasonably correct dose equivalent response independent of energy from thermal to about 10 MeV. In the intermediate neutron energy range (a few keV to a few hundred keV) it overresponds by 100% to 300% of the actual rate. Unfortunately, this is the usual neutron field average energy in the vicinity of a nuclear reactor or an accelerator. Measurements made very close (to reduce the effect of scatter) to an isotopic neutron source would not suffer from this limitation. In the case of high fields or pulsed fields (such as neutrons from most nuclear accelerators) activation techniques are again used. A dosimeter using indium or gold activation foils inside a six inch diameter polyethylene spherical moderator with a cadmium thermal neutron shield measures the flux with a flat energy response from about 1 keV to 1 MeV.

Courtesy, Ludlum Measurements, Inc.

Fig. 22 - A neutron remball instrument

Finally, for relativistic neutrons over 20 MeV in energy, use is made of plastic scintillators in which the stable carbon-12 nucleus in the plastic is activated through a nuclear reaction. This leads to a radioactive daughter, carbon-11, which produces light flashes in the plastic scintillator. The energy response of the system is flat from 20 MeV to many GeV. The instrument is easily sensitive enough to use for radiation protection measurements in uncontrolled areas.

Calibration sources with a known distribution of neutron energies should be used. Some of the more often chosen sources include californium-252 and isotopic sources in which an alpha emitter is mixed with a target material. This later category includes Pu-Be, Am-Be and Am-B sources. Energy spectra are included for these, and some less common sources, in Appendix A-3. Flux to dose equivalent rate conversion factors are also included there.

To completely characterize a neutron survey instrument, the response to thermal, intermediate and fast neutrons is measured. A research reactor can usually provide a thermal neutron column for calibration purposes. Also, some facilities use an isotopic neutron source inside of a small moderating cavity (such as concrete blocks) and shadow shield to obtain an accurate thermal flux for instrument calibration. In the intermediate energy range, the most popular choice is a Cf-252 source moderated with heavy water, D_2O. Depending on the amount of moderator, average energies in the few hundred keV range are produced. The fast neutron response of a meter is measured with unmoderated Cf-252 or one of the isotopic sources mentioned earlier.

One note of caution when using a Pu-Be source. Both Pu-238 (86 year half-life) and Pu-239 (24,000 year half-life) are available. At first guess, the longer half-life isotope would seem to be the one of choice. This is not the case. Pu-239 available in the USA contains up to 3% of a contaminant, Pu-241, which decays to Am-241. The americium is, of course, an alpha emitter, and so the actual neutron output rate of a Pu-239:Be source increases with the passage of time for the first 70 years of the source life. The increase is as high as 2% per year. Corrections for Am-241 ingrowth should be made if the source is being used for neutron calibrations. (Pu-238 sources don't have this complication.)

One other minor detail should be considered in using any isotopic source if high accuracy is desired. These sources produce an anisotropic output, i.e., at a fixed distance, the number of neutrons emitted per second depends on the orientation of the source. The variation with orientation is typically 4 to 6%.

Probably the biggest problem in achieving a reasonable neutron calibration is taking proper account of the scattered field. One of the easiest ways to check for this problem is to measure the deviation of the dose equivalent rate from the predictions of the inverse square law. If scattered neutrons are present, the dose rate will fall off more slowly than calculated. Neutrons are readily scattered by the air in the room, room walls and ceiling, the ground beneath the calibration room, and equipment and personnel in the vicinity. In the fast neutron energy range, a rule of thumb is to choose a room large enough that the distance to the closest wall is at least two times the distance between the neutron source and the instrument under calibration.

To make a correction for scatter, the shadow-cone method is often used. This involves making a measurement first with and then without a solid plastic conical shield between the source and neutron detector. The shield blocks the primary, direct beam so in the first reading, only scattered neutrons reach the detector. The difference between the two readings allows the scattered contribution to be determined. Because this method is so sensitive to design details, the NCRP recommends an alternate method, placing the source so close that scatter is negligible. Then, a non-uniformity correction is applied. Details are given in Appendix A-3.

Due to the multitude of difficulties in calibrating neutron survey meters, the NCRP recommends that a neutron instrument be considered calibrated if it reads within ± 20% of the "known" field at dose equivalent rates over 0.02 mSv/hr (2 mR/hr). At lower dose equivalent rates, the allowed uncertainty increases to ±35% of the "known" field. They also point out that even if the meter is calibrated to these criteria, under field conditions, the uncertainty in the makeup of the neutron energies reaching the detector will probably cause a measurement accuracy to be considerably worse than the ±20 or ±35% figures.

Wipe Testing for Surface Contamination

The smear, swipe or wipe test is universally used for removable surface contamination monitoring. In the preceding sections dealing with alpha, beta and/or gamma surface contamination measurement, the quantity being measured with a survey meter probe always included both the fixed contamination and the removable contamination. The fixed + removable is called the "total contamination." With the

implementation of Subpart E, of Title 10 Part 20 of the Code of Federal Regulations, decommissioning activities make much less use of the wipe test than in former times. It is now only necessary to establish from a few samples that the removable contamination is less than 10% of the total contamination.

Usually, a cloth, paper, plastic foam or fiberglass disk is wiped over a surface area of 100 square centimeters (4" by 4" square) with a gloved hand. There is one exception to this rule, the DOT provision of 300 sq cm for package smear samples taken prior to transport. The U.S. NRC Regulatory Guides specify that the wipe should be taken with a dry medium using moderate pressure. In the case of very low energy beta emitters such as ^{3}H or ^{14}C, the wipe is usually counted by a liquid scintillation counter. Fiberglass disks are often used since they become transparent in a liquid scintillation counter vial. The plastic foam wipes are even better since they dissolve in the liquid scintillation fluid. A wetting agent is sometimes recommended for tritium wipes. The agent used should be the same chemical polarity as the contamination. The DOE Rad Con Manual specifies a wet swipe or styrofoam for tritium tests.

Wipe samples must be separately packaged to avoid cross-contamination of samples. Each should bear a label with the date and location for the sample. Nu-Con® Smears from D.A. Services Inc. are individually attached to folding covers where sample information can be written. Counting for alpha contamination and higher energy betas is usually done with a gas flow proportional counter. Gamma contamination on wipes can be measured with a scintillation or semiconductor counter.

A closely related procedure is the leak testing of sealed sources using the wipe test. (Note that leak testing is only one aspect of managing sealed sources. They should still be periodically inspected for cracks, deformities or broken windows.) The usual regulatory position is that sealed sources containing an activity above specified limits should be tested at intervals not exceeding six months. Although Part 20 of Title 10 of the Code of Federal Regulations does not explicitly state the requirements for leak testing, all NRC licensees have the necessary information incorporated into their licenses as a license condition and many Agreement States have written requirements into their state code. Leak test limits vary depending on the relative hazard of the radionuclide. The wipe test sampling medium is usually smeared over the entire surface of the source capsule. Remember to put on protective gloves first! Any source which is found to be leaking over 5 nanocuries of removable contamination must be immediately taken out of service and then repaired or disposed of as rad waste.

On high activity sources, some procedure must be worked out to minimize radiation exposure to the technologist performing the leak test. The wipe sample might be mounted on the end of a rod so that a reasonable distance can be maintained while wiping the external surface. As an alternative, wipes can be taken by hand at the closest accessible surface to the source while it is safely stored in its normal shield. On sources with a large physical volume, the immersion test is sometimes used. The source is placed in a fluid bath and the pressure is reduced with a vacuum pump. The presence of a stream of bubbles indicates a leak. Sealed radium sources are usually leak tested differently. Since some of the daughter activities from radium decays are gaseous, the pressure gradually builds up inside the source with age. Individual radium sources can be sealed up overnight in a tightly capped vial containing a small amount of activated charcoal. After a day or so, the radium source is removed and the vial is counted. A leaking source releases radon which is trapped by the activated

charcoal. Radioactivity in the charcoal indicates which of the sources are leaking.

Finally, an old National Bureau of Standards technique is still sometimes used. The container in which the source is stored is lined with absorbent filter paper so that the exterior of the source contacts and is "wiped" by the source capsule when it is removed or replaced. The absorbent material is then counted periodically to catch leaking sources.

Instruments for Homeland Security Operations

Following the September 11th terrorist attacks, numerous agencies with first responder oversight expressed a need for information on suitable radiation safety survey meters. While the list of favorite instruments in the radiation protection technologist's arsenal would generally be applicable, there are some unique aspects to first responder instrumentation. For homeland security monitoring situations, the survey meters need to meet the requirements in Figure 23.

Portable	**Very simple to operate and interpret**
Very rugged	**Wide ranging in dose rate**
Alarming	
Operational over temperature/humidity extremes	

Fig. 23 - Special requirements for Homeland Security instruments

As a result of their special needs, the American National Standards Institute quickly formulated a set of four standards specifically for Homeland Security applications. All four were released in early 2004. Through an agreement with the federal Department of Homeland Security, DHS, these standards have been made available on the internet (See "Other Resources" at the end of this Chapter for more information.) The specific instrument types covered are as follows:

- ANSI N42.32-2006 Alarming Personal Detectors
- ANSI N42.33-2006 Portable Radiation Detectors
- ANSI N42.34-2006 Hand-held Meters for Radionuclide ID
- ANSI N42.35-2006 Portal Monitors for Pedestrians, Vehicles & Packages

The alarming dosimeters must be readable in the dark, weigh <400 grams, survive 6 consecutive drops from 1.5 meters onto a concrete floor, alarm at a sound level of 85 dBA at one foot, and alarm within 2 seconds to a step increase of 50 μR/hr. Portable survey meters must weigh less than 6 pounds, read exposure rate, read from 100 μR/hr to 1,000 R/hr, operate at least 24 hours on a set of batteries, be accurate to ±30%, cover gamma energies from 60 keV to 1.33 MeV and work in 40 mph wind driven rain falling at a rate of 5 inches per hour! (Any volunteers to test this?)

The third document covers portable Multichannel Analyzers, i.e. radionuclide identifiers. They must be stand-alone devices (no laptop computers allowed), cover the 25 keV to 3 MeV energy range, display a gamma spectrum, have an audible alarm on dose rate, read dose rate to ±30%, contain a radionuclide library of at least 26 common radioisotopes, identify a list of 18 common sources both bare and also shielded by 5 mm of steel, and simultaneously identify both Ba-133 and plutonium. The portal

monitors for homeland security applications must respond to pedestrians walking at 1.2 m/sec, trucks & trains traveling 8 km/hr, gamma rays from 60 keV to 2.6 MeV, and must correctly alarm in 59 out of 60 test trials to gamma sources from 4 μCi of Co-60 to 462 μCi of Am-241 and to neutrons from Cf-252.

Instrument manufacturers responded rapidly to the new ANSI documents. Battelle Northwest Laboratory was enlisted to carry out the specific tests according to the ANSI protocols to see which of the multitude of different commercial instruments submitted met the criteria. Then the games began! The submission by manufacturers for this first round of testing brought in an overwhelming 175 meters and 241 probes. There were a total of 28 companies that participated and met the 2004 deadline.

It was decided that other national laboratories were needed to share the workload. In the end, Oak Ridge National Lab took over the radionuclide identifiers (ANSI N42.34). Los Alamos National Lab ran the portal monitors (ANSI N42.35) through their paces. Lawrence Livermore National Lab assisted with portal monitors and tested the portable survey meters (ANSI N42.33). Finally, Battelle kept the alarming dosimeters (ANSI N42.32) and some of the portable survey meters. The test series was completed in February 2005 and the results submitted to DHS. The lengthy time for first round testing was blamed on late submissions by some manufacturers. The second round deadline for submission also fell in February 2005. The costs for this second round of instrument testing were billed back to the participating companies. The first round tests were done at no charge to the manufacturers.

The Department of Homeland Security released the results to the instrument companies in March 2005. A number of disappointments became evident. Although it had been originally planned to list the various meter/probe combinations which had "passed," that idea never reached fruition. Instead, for each separate mechanical, electrical, electronic, radiological and environmental criterion, the results for each meter/probe were published. For example, this includes 34 separate tests for the ANSI Portable Radiation Detectors category! Few instrument combinations were able to survive the entire test array intact. Many "off the shelf" meters that are commonly used with confidence by radiation protection technologists failed one or more of the more exotic test criteria. Minor design changes would have to be introduced in numerous cases, probably resulting in across the board price increases with only questionable increase in value.

It was originally hoped that some sort of numerical rating system, similar to certain consumer product testing labs, would result. This would then give the purchaser/user a firm basis for choice between all the competitors. Alas, this too fell by the wayside. Neither an overall pass/fail score nor a numerical ranking was presented in the final product. (At least they didn't charge for round one!)

Apparently the numerical ranking was dropped because there were no standards in place to verify the accuracy and capabilities of the national labs that did the testing. (Who was going to test the testers?) The next step was to involve the National Institute of Standards and Technology, NIST, which conducts the Laboratory Accreditation Program for personnel dosimetry service companies. This would then put the testers on a firmer legal base so they presumably could then issue numerical rankings for Homeland Security instrumentation in the future.

Finally, some manufacturers were astounded when they learned

that the first round of testing did not strictly follow the four ANSI Homeland Security standards. The test labs apparently changed some criteria and perhaps added others. When pressed by manufacturers on what these changes were, so that the companies could fix the deficiencies in their instruments, they were told that these changes could not be revealed for national security reasons!

A second round of testing then followed. This time, manufacturers, were required to pay all of the costs involved. The tests were conducted between May 2005 and July 2006, under direction of NIST. In contrast to the first round, each instrument submitted was assigned a score for each test category. A final score was then released as a percentage of the maximum possible points. Although the instrument companies sent 175 different meters to round one, only a total of 30 different meters were submitted and scored for the second round. NIST did not establish a "passing score" for any of the four categories. If it is assumed that users would want to have a meter that scored at least 75% of the possible points, then, the round 2 results were not encouraging. Only 6 out of 15 personal alarming devices, 1 out of 5 radionuclide identifiers, 1 out of 2 portal monitors and 2 out of 8 portable survey meters reached this 75% level.

As of 2011, three new ANSI standards have been released. One standard dictates the format that homeland security instruments use to output their data. The second standard deals with portal monitors that can identify radionuclides and the third covers pocket size radionuclide identifiers (must be < 400 grams and able to work after being dropped 6 times from a 5 foot height onto a concrete floor). However, the traditional health physics instrument companies seem to be boycotting the whole process. The cost of altering proven designs to meet the new ANSI standards appears to exceed the expected return on that investment. In response, the Department of Homeland Security has formed a new committee that is working on more instrument standards. It has yet to begin testing rounds.

Finally, before leaving the area of radiation survey meters, see Figure 24. This full page Table summarizes most of Chapter 12 by suggesting appropriate instruments for field surveys under a wide variety of field conditions. Alpha, beta, gamma, tritium and neutron surveys are listed for various expected radiation dose rate levels. Some comments on operational use are also included for the recommended meters.

Decommissioning Nuclear Facilities

Introduction

As a result of ever tightening restrictions on radioactive material use and aging nuclear facilities, the need for surveys in connection with decontamination and decommissioning ("D and D") activities leading to closure of licensed operations is on the increase. As mentioned earlier in this Chapter, Subpart E, **Radiological Criteria for License Termination** of 10 CFR 20 establishes standard procedures which are codified in **The Multi-Agency Radiation Survey and Site Investigation Manual**, MARSSIM. The MARSSIM has been accepted by the U.S. Nuclear Regulatory Commission, The Environmental Protection Agency, The Department of Energy and The

Radiation Type	Expected Radiation Level	Survey Meter Type	Operational Comments
Alpha Contamination	All levels	Alpha Scintillator, ZnS(Ag)	Check for window light leaks before use. Check background rate often as probe gets contaminated easily. Be alert to gamma field interference.
Alpha Contamination	All	Alpha Proportional - Air or Gas	Check often for probe contamination. If using gas, keep an eye on the gas gauge. If using air, avoid surveys in humid areas.
Beta Contamination	All	Pancake GM	Calibrate with expected beta energy as probe quite energy dependent.
Beta Contamination	All	Gas proportional	Calibrate with expected beta energy as probe quite energy dependent. Calibration changes with air pressure.
Beta Contamination	All	Plastic scintillator	Check for window light leaks before use. Calibrate with expected beta energy as probe quite energy dependent.
Gamma Field	Low to Medium	Side wall energy compensated geiger counter	Won't work for gammas below about 50 keV, i.e., Co-57, I-125. Underestimates gammas above 1.5 MeV, i.e. N-16
Gamma Field	Low to Medium	Solid scintillator, NaI or tissue equivalent plastic	NaI micro R meter very energy sensitive.
Gamma Field	Low to Medium	Pressurized ion chamber	Verify chamber is still pressurized before use.
Gamma Field	High	Ion chamber	Allow warm-up time. Zero meter in low background area. Calibration depends on ambient air pressure.
Gamma Field	Extreme	High range telescoping GM	Instruments such as Teletector and Ludlum Stretch Scope.
Tritium Contamination	All	Windowless gas flow proportional	Steep learning curve, sensitive to barometric pressure, humidity, dust, gas flow rate and static electricity. H-3 calibration source fragile.
Neutron Field	All	Rem Ball, 9 inch	Reasonable accuracy for energies between thermal and 10 MeV. Overresponds to intermediate neutrons. High gamma field rejection.

Fig. 24 - Selection Table for radiation survey instruments

1) Introduction
2) Overview of Radiation Survey and Site Investigation Process
3) Historical Site Assessment
4) Preliminary Survey Considerations
5) Survey Planning and Design
6) Field Measurement Methods and Instrumentation
7) Sampling and Preparation for Laboratory Measurements
8) Interpretation of Survey Results
9) Quality Assurance and Quality Control

Fig. 25 - The MARSSIM Table of Contents main headings

Department of Defense. A listing of the topics covered in the MARSSIM Table of Contents is shown in Figure 25.

The MARSSIM is a 658 page document that contains detailed D and D instructions to cover almost any contingency that would be found in cleaning up a large multi-building nuclear facility spread over hundreds of acres of land. As such, it appears intimidating at first glance. Many technologists will never be involved in a maximum size MARSSIM project. On the other hand, the basic principles and procedures apply even to small remediation jobs. A small one room radiation lab can be remediated and have its license terminated using MARSSIM surveys. The complex flow charts of a large scale MARSSIM project can be pared down to the essentials. For persons tasked with small MARSSIM jobs, Supplemental Chapter S-3 in this book provides a "Twelve-Step MARSSIM Process" along with a checklist and comments based on Pacific Radiation's MARSSIM D and D experiences.

Technicians working in the decommissioning sector should also be familiar with the Multi-Agency Radiological Laboratory Analytical Protocols, MARLAP. This document provides complementary guidance to the MARSSIM. The MARLAP was issued in two parts and covers all aspects of obtaining credible analytical lab results to support the MARSSIM decommissioning project. Part I addresses the needs of project planners regarding identifying the required analytical data, evaluating the radiation lab, and validating the data quality. Part II is devoted to operational aspects of the radioanalytical lab itself, and discusses preserving samples, preparation and separation procedures, lab instrumentation, and quality control issues. Free copies of MARLAP can be downloaded - see "Other Resources" at the end of this Chapter for details.

The MARSSIM Process

Following a decision to decommission, a licensee must notify the appropriate state authorities or the NRC (depending on which agency issued the license) and then dispose of any remaining radioactive material at the site. The next step involves creation and carrying out of a plan to clean up contaminated areas. Contamination levels must be reduced below acceptable release guidelines. NRC licensees are allowed to use a generic computer code, "DandD," which provides guideline values for surfaces in dpm/100 sq cm and soils in pCi/gram. The objective of Subpart E of 10 CFR 20 is

to have licensees remediate a site so that residual radioactive contamination "results in a TEDE to an average member of the critical group that does not exceed 25 mrem (0.25 mSv) per year." Surface contamination levels are entered into the DandD code which outputs the annual TEDE dose. The concentration of surface contamination that would just produce 25 mrem/yr dose under the scenario modeled by the computer is called the "Derived Concentration Guideline Level" or DCGL.

The surveys conducted to support license termination, as with most surveys, represent legal data which must be complete, legible (in INK), and identifiable, i.e., the survey data sheets must be dated and signed by the radiation protection technologist making the measurements. If it is necessary to change data after recording, draw a single line through the error and initial the changes.

It is important, during sampling, to prevent contamination of other samples. For example, wipe samples should be isolated from each other with envelopes and the surveyor must wear protective gloves when collecting the sample.

In surveying inside buildings, begin by removing uncontaminated equipment and furnishings so that walls and floor can be accessed. In some cases, wall coverings and floor coverings will need to be removed. Covers to floor drains, electrical panels and ventilation ducts need to be opened. In the case of potential alpha contamination, great care must be taken to remove wax, grease, and oil or water films that would prevent detection. A reference grid is then laid out on the walls and floor, and scale drawings showing the grid relative to building features are prepared.

Moving to the building exterior, remove uncontaminated equipment and heavy ground cover. The reference grid can be laid out with wood or metal stakes. Again a scale drawing is made and gamma scans are carried out. If the soil is paved over, total surface activity is measured in those areas. The direct radiation level is measured at 1 meter above ground height. A movable tripod arrangement set at 1 meter is convenient for this purpose.

For soil samples, check first with the analysis lab for any special conditions. (The MARLAP can be helpful when communicating with the chosen radioanalytical laboratory.) Usually about 1 kg of soil is collected down to 15 cm depth if gamma ray spectroscopy is anticipated. For only chemical analysis of alpha and beta activity, 0.1 kg suffices. Remove grass, sticks and rocks from the sample before packaging it.

The MARSSIM scoping survey consists of direct readings, wipe samples and soil or construction material samples. It covers the entire site. The objective is to determine what radionuclides are present above guideline values and to segregate the site into contaminated versus non-contaminated areas. High accuracy is not necessary at this stage. The MARSSIM characterization survey then follows up in the contaminated areas by making more quantitative measurements, particularly in areas that are close to the release guidelines. Ratios of contaminants are determined.

The last survey that is the responsibility of the licensee is the MARSSIM final status survey. Surface contamination, direct radiation and radioactivity in soils or building materials are accurately measured using a grid survey technique. Values are compared against the DCGL guidelines, using specified statistical tests, to demonstrate that cleanup was adequate.

The final status survey is the justification for the regulatory authorities to terminate the license. As part of their evaluation of the decontaminated site, they make a small number of measurements to show that the licensee's survey data are valid.

A report of actions taken is finally filed with the authorities, who, upon satisfactory review and confirming measurements, release the site to the licensee with no further restrictions on its use.

Counting Statistics For Data Analysis

Introduction

Recall that the process of radioactive decay is random in time. This means that we can never know exactly when a given radioactive atom will decay. If we have a large collection of such atoms, the decays will follow the laws of chance and we will be able to make a prediction about how many of the sample atoms will decay but not which ones. Although our prediction about the number decaying per minute from a sample may be quite accurate, it is never exact. There is always an uncertainty or error associated with the prediction. This finally means that we cannot calculate the TRUE decay rate without any uncertainty. Further, we cannot measure the TRUE decay rate without uncertainty due to the randomness of the decay process. We thus must infer the true rate by "sampling" the count rate for some limited time interval. We then determine our best guess as to the true rate, and the uncertainty in that rate, by using the laws of counting statistics.

Count Rate And Its Error

By measuring the number of disintegrations in a sample over a time period such as 1 minute, we make a guess as to the true decay rate by calculating:

Decay Rate = # Decays ÷ Time **[Eqn. 2]**

Note that, as discussed in Chapter 7, the number of decays is not usually the number of counts recorded by the detector system. Most detectors have a counting efficiency which is less than 100%. Therefore, the number of decays of the sample is given by:

 # Decays = #Counts ÷ Detector Efficiency, ε **[Eqn. 3]**

The question of how close this "guess" is to the true decay rate is harder to answer. If a large set of 1 minute counting runs is made, the "spread" in values that will likely be obtained is shown graphically by Figure 26. The vertical axis represents the probability of getting a particular number of counts, "n," in a given run, and the horizontal axis is a plot of the number of counts. The bell-shaped curve results from the laws of chance. The central value (indicated by the dashed vertical line) represents the best guess as to the true rate. But, as the curve shows, other values of "n" are also possible. As the value of "n" gets farther from the center value, the chances of that value being counted decrease.

The error or uncertainty in our best guess is closely related to the width of the bell curve. Figure 27 illustrates two hypothetical counting experiments to determine the count rate. Since the results of experiment B give a much narrower curve, the error or uncertainty in our best guess is smaller in experiment B compared to experiment A. This general idea of the narrowness meaning less error can be made more

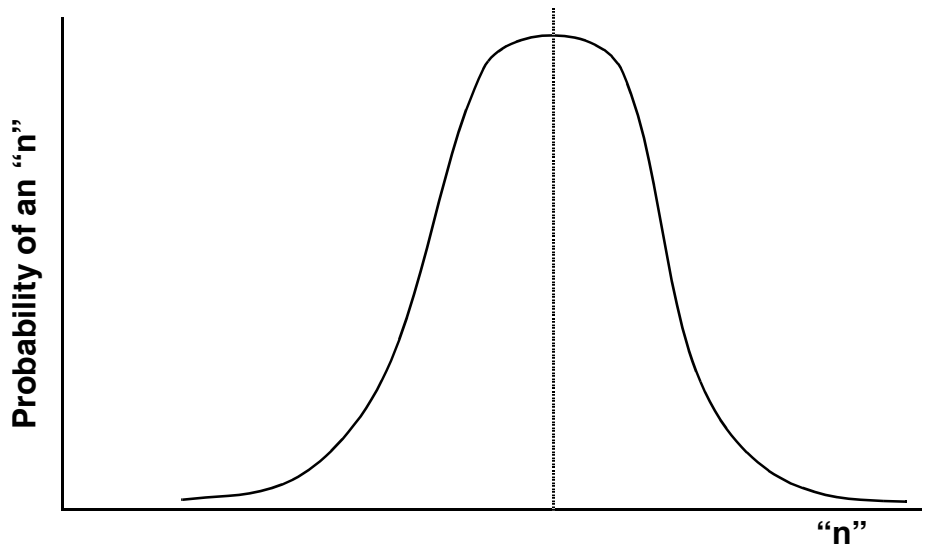


Fig. 26 - Probability of obtaining some particular value, "n"

quantitative by using the theory of statistics. The width of the bell curve is related to the standard deviation which is defined by Figure 28. The small symbol σ is a Greek lower case sigma. It is universally used to represent the standard deviation. One standard deviation is the distance from the peak (the dashed vertical line) out to a vertical line which encloses 34.15% of the total area under the curve. The total area under the curve equals 100% which just means that it includes all the counting results for a series of measurements. The center line value is called the mean value or average or best guess of the true value. The symbol used to represent the mean is usually a letter

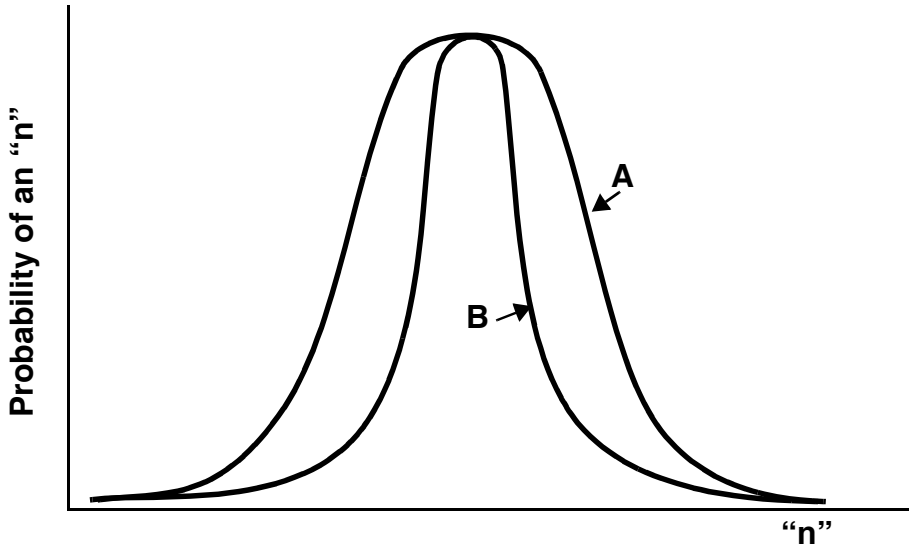


Fig. 27 - Two hypothetical counting experiments to measure "n"

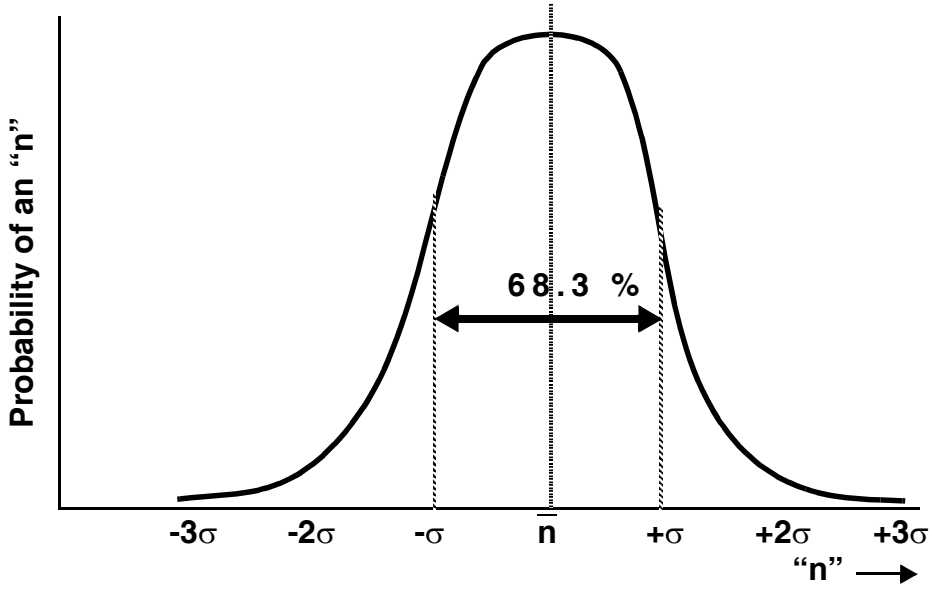


Fig. 28 - A graphical definition of the standard deviation

with a horizontal bar above, e.g., $\bar{n}$. (It is read "n bar"). As shown by the curve in Figure 28, if we include values of "n" from one standard deviation below $\bar{n}$ to one standard deviation above $\bar{n}$, this will enclose 68.3% of the total area under the curve. If the range of "n" values is extended to ±2 standard deviations from the mean, these limits would enclose 95% of the total area under the curve. The size of the standard deviation gives the error or uncertainty in the mean value.

We need to be able to calculate the value of the standard deviation so we can report the uncertainty in our guess of the count rate. It turns out that, in the special case of counting nuclear decays which occur purely randomly in time, the standard deviation for a single count is given by the square root of the number of counts.

Standard Deviation of a Single Count = $\sigma = \sqrt{n}$ **[Eqn. 4]**

Since the count rate is the number of counts divided by the counting time, we report the rate and its uncertainty as follows. Sample Problem 8 illustrates this.

Count Rate = R = n/t **[Eqn. 5]**

Standard Deviation in Rate = $\sigma/t = (\sqrt{n})/t = \sqrt{\frac{R}{t}}$ **[Eqn. 6]**

Reported Count Rate = R = n/t $\pm (\sqrt{n})/t$ **[Eqn. 7]**

The relative error and percent error are two other terms that we can now calculate. These are defined as:

Relative Error = Error ÷ Reported Value and **[Eqn. 8]**

% Relative Error = (100% x Error) ÷ Reported Value. **[Eqn. 9]**

Since the error increases only as the square root of the counts, the percent error and relative error will get smaller as the number of counts becomes larger. Sample Problem 9 gives some examples. Note that increasing the number of counts by 10 times (from 100 to 1000) decreases the percent uncertainty from 10% to 3.2%.

Sample Problem 8

GIVEN:
A single count of a sample yields 100 counts in 2 minutes.
FIND:
What is the error and how is the rate reported?
SOLUTION:
The "error" is the standard deviation = $\sigma = \sqrt{n} = \sqrt{100} = 10$ from Eqn. 3.
The count rate = R = n/t = 100 counts/2 min = 50 counts/minute.
The uncertainty in the rate is σ/t = 10 counts/2 min = 5 counts/minute.
The reported rate = R = 50 ± 5 counts per minute from Eqn. 7.

Sample Problem 9

GIVEN:
2 minute counts of two different sources give 100 counts and 1000 counts.
FIND:
Find the relative and % errors in the two rates.
SOLUTION:
For n = 100 counts, Count Rate = R = 50 ± 5 cpm from Sample Problem 8.
Relative error = Error/ Value = 5/50 = 0.10 from Eqn. 7.
The % relative error = 5/50 x 100% = 10% from Eqn. 8.
For n = 1000 counts, $\sigma = \sqrt{1000}$ = 32 counts. Thus, R = 1000/2 ± 32/2 = 500 ± 16 cpm
Now, the relative error = 16/500 = 0.032 and the % relative error = 16/500 x 100% = 3.2%

Confidence Level

Returning to Sample Problem 8, how much faith can we have that the next count that we would make would show a rate falling within the range 50 ± 5 cpm? In other words, how confident are we that the rate would be between 45 and 55 cpm? Remembering that the uncertainty represents ± one standard deviation, and that the ±1 σ limits enclose 68.3% of the area under the curve, it can be concluded that we are 68.3% confident that the next count would produce a rate between 45 and 55 cpm. That is, we have about a two-thirds chance of falling in that range or conversely, about a one-third chance that the next measured rate will be above 55 cpm or below 45 cpm. In statistics theory, we say that this count rate is being reported at the 68% confidence level.

Another way to look at this concept is that if we repeated the count rate measurement 100 times in a row (assuming the source half-life is relatively long), 68 of the measured rates would be between 45 and 55 cpm, 16 would be <45 cpm and the

remaining 16 would be >55 cpm. It is common practice in the fields of engineering and physics to report results to the 68% confidence level, i.e., the number ± one standard deviation. In the field of radiation protection, where our measurement results are often directly utilized to decide on the safety of some operation or facility location, it is felt that "being right" only two-thirds of the time is not enough. Using only "one sigma" means that, on the average, the dose rate or contamination level is actually higher than the stated range 16% of the time. Generally, radiation protection measurements are reported at the 95% confidence level – now we are "right" 95 times out of 100 (much to the chagrin of weather forecasters). To be in this envious position, we are forced to give something up. The RANGE of values in the reported results must be doubled. Recall that when discussing the bell curve earlier it was stated that 95% of the total area under the curve is enclosed by limits at ± 2σ (± 2 standard deviations). By reporting results with an uncertainty of ± 2σ, we can be 95% confident that the actual true rate is within the specified range. The downside is that the percent and relative errors are doubled. We are more confident of our stated result but it is less precisely stated. Sample Problem 10 illustrates the arithmetic.

Sample Problem 10

GIVEN:
Let n = 100 counts and t = 2 minutes.
FIND:
What is the count rate, to 95% confidence, and the relative and % errors?
SOLUTION:
To 68% confidence, R = 50 ± 5 cpm, since σ = 10 and t = 2.
To achieve 95% confidence, the range must be extended to ± 2 σ = 20. Now, R = 50 ± 10 cpm to 95% confidence as 2 σ/t = 20/2 = 10 cpm.
The new Relative Error is 10/50 = 0.20 and the % Error = 10/50 x 100% = 20%.

This brings up a new problem. If a measurement result is stated to be 30 ± 6 units, how do we know what confidence level is being used? From the stated results, the confidence level cannot be determined. The raw numbers would have to be given to be able to figure out how many sigmas are represented by the stated uncertainty. In cases like this, we fall back on the rules of tradition. By convention, we accept the stated uncertainty of a measurement result as being one standard deviation unless stated otherwise. Thus, the 30 ± 6 result means that 6 is ± 1σ for this measurement. The two common ways of indicating use of a different confidence level are as shown:

To 95% confidence, X = 30 ± 12 or X = 30 ± 12 (2 σ).

Confidence Level	# of Standard Deviations
68.3%	1.0
90%	1.65
95%	2.0
99%	2.6

Fig. 29 - Confidence level values

Although 95% is commonly used in radiation protection technology, other levels of confidence are occasionally used. The table in Figure 29 shows the number of standard deviations that correspond to various confidence levels.

Background Corrections

In many counting situations, the sample has so little activity that it seems to count at the background rate. This is, of course, the norm for counting environmental samples. Thus, we need to resort again to the laws of statistics to show how to properly report the sample count rate. This is perhaps best done with an example.

Assume that you have been hired as a nuclear safety consultant by a local city council concerned about radioactivity in their drinking water supply. You are asked to make measurements to determine if the water is radioactive. Assume that the sample, including background, counts at a rate of 1040 counts in 10 minutes. The background count, with a clean "dummy" sample in place is counted for 60 minutes and gives 6000 counts. The basic arithmetic is as shown in Sample Problem 11A.

Sample Problem 11A

GIVEN:

The city water supply shows 1040 counts in 10 minutes. Background is 6000 counts in 60 minutes.

FIND:

Is the water radioactive?

SOLUTION:

The gross rate is R_{S+B} = 1040/10 = 104 cpm where R_{S+B} means sample + background rate. Since the background rate is R_B = 6000/60 = 100 cpm, the net sample rate becomes $R_S = R_{S+B} - R_B$ = 104 - 100 cpm = 4 cpm.

So, YES, the water is radioactive!

At this stage, it would appear that the water is radioactive by an amount corresponding to 4 cpm. (This could be converted to pCi/liter if the counter efficiency is known along with the sample volume). However, having just learned some statistics skills, we realize that this 4 cpm is subject to some uncertainty. The equation used to calculate the uncertainty in the difference of two nuclear count rates is given in Equation 10 along with the calculation for this example in Sample Problem 11B.

$$\sigma_{R_S} = \sqrt{\frac{R_{S+B}}{t_{S+B}} + \frac{R_B}{t_B}} \qquad \textbf{[Eqn. 10]}$$

Sample Problem 11B

Continuing, plugging in the values into Eqn. 10 gives:

$$\sigma_{R_S} = \sqrt{\frac{104}{10} + \frac{100}{60}} = \sqrt{10.4 + 1.67} = 3.5 \text{ cpm.}$$

So, R_S = 4 ± 3.5 cpm.

So, the sample is still radioactive, at least to 68% confidence. But, we usually specify 95% confidence for reporting radiation protection measurements. To 95% confidence (2 σ) the result is:

$$\mathbf{R_S = 4 \pm 7 \text{ cpm } (2\sigma).}$$

This time, the water cannot be said to be radioactive since the uncertainty associated with the net 4 cpm carries the range below 0. Statistically, the 4 cpm is too low to be definitely caused by radioactivity in the water supply. Another way to put this is that the net 4 cpm obtained in the single measurement might have been caused by a slight random increase in background during the 10 minutes that the sample was in place. It would be proper to report to the city council that, at the 95% confidence level, the water is not radioactive.

Considering, for a moment, the public concern for radiation matters, it would probably not be responsible to leave this issue without further comment. After all, the drinking water did show a net positive count rate of 4 cpm. The best approach is to look closely at the calculations. The reason why the water goes from being radioactive to not being radioactive is basically caused by the large uncertainty in the difference of two almost identical count rates. (Again, please note that this situation is not a theoretical textbook case but occurs for almost every environmental sample counted.) The question of the radioactivity would be much clearer if the standard deviation of the difference (the 3.5 cpm) were much smaller. In Sample Problem 11B note that the 3.5 cpm resulted from taking the square root of the sum of two numbers, the individual rates divided by the counting times. Also, note that one of these numbers is much larger than the other. Tracing this back, it can be seen that the gross sample + background term is so much larger than the background term because the sample was only counted for 10 minutes. The general rule of thumb is that when counting near background rates, the sample should be counted for the same time as the background count to minimize the uncertainty in the net rate. For example, if the sample had been counted for 60 minutes, (the same as the background), and had still given a rate of 104 cpm, the standard deviation of the net rate would have been 1.8 cpm. Now the water is, to 68% confidence and 95% confidence, radioactive.

There is a general formula used to calculate the relative counting time for the sample and the background so that the error in the net difference is minimized. It is:

$$\frac{t_{S+B}}{t_B} = \sqrt{\frac{R_{S+B}}{R_B}} \qquad \textbf{[Eqn. 11]}$$

This formula would be used for a counting situation in which a single background run and sample run were to be made, and the total counting time is limited.

The calculations discussed just above were for the case of samples counting near the background rate. There are occasions when a technologist will encounter the opposite problem – very high count rates. Under these conditions, it is necessary to introduce a correction factor to take into account the loss of counts which happen to enter the detector during the dead time. The following formula is used to calculate the true count rate under these circumstances.

$$R_{true}\,(\text{cpm}) = \frac{R_{meas}(\text{cpm})}{1 - R_{meas}\tau} \qquad \textbf{[Eqn. 12]}$$

where R_{meas} = Measured count rate (cpm)
τ = Detector dead time (min)

One other practical problem is sometimes faced. If the activity of a short-lived isotope is being measured, it becomes necessary to correct for the loss of activity during the count (i.e., the sample activity is <u>not</u> a constant). This is done using the formula given below. A realistic case where this occurs is in counting induced radioactivity in activation foils used to measure neutrons.

$$\text{cpm at time } t_1 = \text{Counts } (t_1 \text{ to } t_2) \times \frac{\lambda}{1 - e^{-\lambda(t_2 - t_1)}} \qquad \textbf{[Eqn. 13]}$$

Detection Sensitivity (MDC and MDA)

The last statistical concept to be covered here is detection sensitivity. This is measured using the Minimum Detectable Concentration, the MDC. Note that in the recent past, this parameter was called the Minimum Detectable Activity or MDA. Since the introduction of MARSSIM in 1997, federal agencies are encouraging everyone to use MDC as the preferred terminology. The MDC (or MDA) is a measure of the least amount of radioactivity that must be present to enable the technologist to state that the sample was radioactive to some confidence level. The concept of MDC is intended to be used as a "figure of merit" or "seal of approval" to demonstrate that <u>a counting system and accompanying laboratory analysis procedures are sufficiently sensitive to generally perform in the desired manner</u>. Many technologists mistakenly treat the MDC as another parameter to be calculated for each sample along with the standard deviation. This misuse has resulted in a statement by the Nuclear Regulatory Commission saying, "It should be recognized that the MDC is defined as an *a priori* (before the fact) limit representing the capability of a measurement system and not as an *a posteriori* (after the fact) limit for a particular measurement."

Technically, the MDC is "the smallest concentration of radioactive material in a sample that will yield a net count (above system background) that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a 'real' signal." In other words, the MDC amount of radioactivity on a sample will yield a net count rate in the system that just barely exceeds 2 standard deviations (95% confidence) for the net count rate. Therefore, that sample will be reported as containing radioactivity.

As discussed earlier in this Chapter, the Scan MDC amount of activity on a concrete floor would stand a 95% chance of being noticed by a technologist during a scan. Note, however, that the Static and Scan MDCs <u>are numerically different</u>! Even with the identical detector, the scanning survey will be less sensitive than the fixed survey since the detector has a full minute or more to distinguish activity above background when held in place rather than moving over the surface.

Based on Equation 10 for the standard deviation of the difference of two count rates, it should be clear that the MDC will strongly depend on the background rate in the counter. In theory, even if the gross sample + background is counted for an infinite time, the net rate will still have an uncertainty which is the square root of the

Static Minimum Detectable Concentration = Static MDC

$$\text{Static MDC (pCi/gm)} = \frac{3 + 4.65\sqrt{R_B \cdot t_B}}{2.22 \cdot \varepsilon \cdot M \cdot Y \cdot t_B} \qquad \text{[Eqn. 14]}$$

where R_B = Background rate (cpm)
t_B = Background counting time (min) & gross count time (min)
ε = Counter efficiency (cts/dis)
M = Sample size (grams)
2.22 = dpm per pCi of activity
Y = Fractional chemical yield, if applicable

$$\text{Static MDC (dpm/100 sq cm)} = \frac{100 \times (3 + 4.65\sqrt{R_B \cdot t_B})}{\varepsilon \cdot A \cdot t_B} \qquad \text{[Eqn. 15]}$$

where A = Probe area (sq cm)

Fig. 30 - The calculation of the minimum detectable concentration, MDC

background rate divided by the counting time for the background. The equations used by MARSSIM for calculating the Static MDC of a laboratory analysis system and the Static MDC for a fixed survey meter reading are given in Figure 30. Two different formulas are included to cover the cases of volumetric contamination (e.g., contaminated soil) measured in picocuries per gram and surface contamination, measured in dpm per 100 square centimeters. A sample calculation is then given (Sample Problem 12) to illustrate the Static MDC formula.

The other case where the MDC concept is useful is the Scan MDC for a moving survey meter that was discussed earlier in this Chapter in the Beta Monitoring section and was illustrated in Sample Problem 5. For the sake of completeness, the equation is listed again, in Figure 31. The ratemeter time constant is NOT included in the

Sample Problem 12

GIVEN:

A lab counter has a background rate of 100 cpm, count time of 1 hour, 0.1 ct/dis efficiency and a 100 gram sample capacity.

FIND:

Calculate the MDC for this system.

SOLUTION:

1 hr = 60 min, so $\sqrt{R_B \times t_B} = \sqrt{(100 \times 60)} = 77.46$ **cts**

With no radiochemical separation, and an immediate count, Y = 1. Thus, we have, from equation 14:

MDC = (3 + 4.65 x 77.46 cts)/(2.22 dpm/pCi x 0.1 ct/dis x 100 gm x 60 m)
= 0.27 pCi/gm

Scanning Meter:

$$\text{MDC (dpm/100 cm}^2\text{)} = \frac{1512 \cdot \sqrt{R_B}}{\sqrt{i} \cdot \varepsilon \cdot A} \qquad \text{[Eqn. 16]}$$

where R_B = Background rate in cpm

ε = Probe efficiency in counts/disintegration

A = Probe area in sq cm

i = Time interval moving probe spends over any point, in seconds

Fig. 31 - MDC formula for portable field instruments

formula as the ability of the technologist to distinguish an elevated area is based on the increased audio click rate, not the meter needle fluctuations, and as mentioned earlier, the time constant of the audio output is zero! PLEASE NOTE: The MDC formulas given in both Figure 30 and Figure 31 are valid only if the counting time is identical for both the gross sample count and for the background count.

Technologists sometimes ask for MDC formulas where the background is counted longer than the sample. For unequal counting times, the following formulas are used.

For Equation 14, the Static MDC in pCi/gm is:

$$\text{MDC} = \left\{3 + 3.29 \sqrt{R_B t_{S+B}(1 + t_{S+B}/t_B)}\right\} / [2.22 \bullet \varepsilon \bullet M \bullet Y \bullet t_{S+B}]$$

For Equation 15, the MDC_{Static} in dpm/100 sq cm is:

$$\text{MDC} = \left\{3 + 3.29 \sqrt{R_B t_{S+B}(1 + t_{S+B}/t_B)}\right\} / [\varepsilon \bullet (A/100) \bullet t_{S+B}]$$

Problem Set

1. What is the difference between a full characterization calibration and a routine calibration for a field survey meter? What percent error is allowed in a survey meter calibration before the instrument is no longer considered calibrated?

2. Why does the NRC prohibit the calibration of ranges routinely used on radiation survey meters with a properly calibrated electronic pulse generator?

3. Justify the text statement that alpha contamination does not present an external hazard to the monitoring technologist.

4. Make a chart to compare the efficiency, gamma rejection, operating convenience, and special limitations of the three types of alpha survey instruments discussed, air proportional, propane proportional and ZnS(Ag) scintillation.

5. Calculate the equivalent thickness of air in cm that corresponds to the typical density thickness of the aluminized mylar entrance window of an alpha proportional counter.

6. Why should the probe be lifted 10 cm away from the surface when a spot of alpha contamination is found?

7. Discuss the circumstances when it is proper to use the audio output of a survey meter and when it is proper to use the meter reading on an alpha survey instrument.

8. The transuranic radionuclide ^{241}Am is an alpha emitter with a half-life of 432 years. Calculate the number of atoms that would be present in a square cm of surface contaminated to the DOE permitted release limits of Figure 4. (Hint: Review the last section of Chapter 2).

9. Why is it particularly important to move the probe slowly and in close proximity to a surface when surveying for alpha contamination? What problem is posed by a surface layer of oil? How can such a surface be monitored?

10. Why is it important to label an alpha survey meter as to whether the instrument had a 2π or 4π calibration?

11. Which two instrument types are commonly used for gamma ray field measurements? Compare their operating characteristics and limitations.

12. Why is a Geiger counter survey meter more appropriate than an ion chamber for measuring the exposure rate leaking through a crack in a shield surrounding a large Co-60 source? Would the Cutie Pie overrespond or underrespond in this situation?

13. What problem would result in using a Geiger counter with the rotating beta shield open to survey a source emitting low energy photons?

14. What is the problem in calibrating the high ranges of an RO-20 ion chamber instrument by placing it a few cm from a point gamma ray source?

15. A cylindrical ion chamber with a diameter of 10 cm is to be calibrated on the 500 R/hr scale. Approximately what activity Cs-137 source would be

needed to perform this calibration without use of correction factors? Show any assumptions made.

16. Why is a Geiger counter practically useless for quantitative surveys of beta fields?

17. A conventional Cutie Pie ion chamber is placed close to a 5 cm diameter spot of contamination. The "window open" reading is 47 mR/hr and the "window closed" reading is 15 mR/hr. Correctly report the radiation levels for this spot if the correction factor is 9.3 mrad/mR.

18. Why is a boron trifluoride tube able to detect thermal neutrons in the presence of a strong gamma ray background?

19. What is the chief limitation on the usefulness of a "fast/slow" neutron survey meter in reading the flux of fast neutrons?

20. A wipe sample from the outside of a sealed Fe-55 source gives a net count of 155 cpm on a detector that reads 1,850 cpm with a 0.8 μCi ^{55}Fe standard. Is this source leaking in terms of current standards?

21. What is the specified frequency for leak testing a large sealed radioactive source?

22. What is the difference between a MARSSIM scoping survey and a MARSSIM final status survey conducted during the decommissioning of a nuclear facility?

23. A 100 sq cm beta proportional probe is to be used for scanning a P-32 spill on a laboratory floor. The measured probe efficiency, including backscatter, is 22% and the background rate is 300 cpm. How fast should the probe be moved to have a good chance of detecting residual contamination spots of 1000 dpm?

24. Define the term "standard deviation." If a nuclear counter receives 400 counts in one minute, what is the count rate and its standard deviation? How long would the sample have to be counted to show a percent error of 2.5%, to 68% confidence?

25. Derive the formula for the standard deviation of the count rate given in Equation 6.

26. Calculate the total number of counts that would have to be recorded in a single counting measurement in order for the percent error (1 σ) to equal exactly 1%.

27. A single count of 1000 is recorded in one minute by a Geiger counter and scaler using a Cs-137 source. What is the expected range of values that

another count should show with the same counting conditions: a) to 68% confidence? b) to 90% confidence? c) to 95% confidence?

28. If the counter setup in Problem 27 has a measured background rate of 100 cpm (measured during a counting time of one minute), calculate the net count rate and its uncertainty: a) to 68% confidence and b) to 95% confidence. What percent errors do these results have?

29. Define the term "minimum detectable concentration." Why should it NOT be calculated for every sample counted?

30. A laboratory procedure has been developed to count I-131 in lake water. Samples are counted in a shielded cave using a NaI(Tl) crystal and pulse height analyzer. The MDC is calculated to be 30 pCi/l for the system. What are some changes that might be made so that this system meets the NRC guidelines on MDCs given in Chapter 10, Figure 13?

31. A weak Co-60 standard is being assayed. The gross count rate of the source is 65 cpm in a system with a 10 cpm background. If one hour is to be allowed for counting, calculate the optimum counting times for the background and the source.

32. A bench model Geiger counter records 200,000 cpm on a wipe test sample. The counter has a dead time of 125 μsec. What is the true count rate? If the counter efficiency is 16%, what is the actual activity on the sample in Bq?

S-1. By measurement, it is determined that the exposure at the wall of a patient waiting room is 0.004 mR for each routine chest x-ray taken in the x-ray room behind the wall. Under what conditions would this facility meet current radiation protection standards (1 mSv per year to members of the general public) if it is assumed that the occupancy factor for the waiting room is 1/16 and that only chest films are taken in the x-ray room?

S-2. Following a criticality accident, a technician records 3487 counts in 5 minutes from copper-62 ($T_{1/2}$ = 9.7 minutes) activity in a copper foil. Calculate the count rate at the beginning of the counting time.

S-3. A portable beta meter with scaler readout and 1 minute timer is used for total surface contamination measurements. The background has been measured for 10 minutes to be 26 cpm. If the efficiency is 32% and the probe area is 20 cm^2 what is the MDC in dpm/100 cm^2?

S-4. A 9" rem ball is being calibrated with an Am:Be source with an output of 2 x 10^7 n/sec. What value should the meter be adjusted to read if the source center is held 2 cm from the surface of the rem ball? (See Appendix A-3)

Other Resources

1. “Calibration of Survey Instruments Used in Radiation Protection for the Assessment of Ionizing Radiation Fields and Radioactive Surface Contamination,” NCRP Report No. 112, Bethesda, MD, 1991.

2. “Health Physics Surveys for By-product Material at NRC-licensed Processing and Manufacturing Plants,” U.S. NRC Regulatory Guide 8.21, Revision 1, Washington, 1979.

3. “Radiation Surveys at Medical Institutions,” U.S. NRC Regulatory Guide 8.23, Washington, Revision 1, 1981.

4. “Compendium of Neutron Spectra and Detector Responses for Radiation Protection Purposes,” Griffith, Palfalvi and Madhvanath, International Atomic Energy Agency, Technical Report Series No. 318, Vienna, Austria, 1990.

5. J. Wade and G. Cunningham, “Radiation Monitoring, A Programmed Instruction Book,” Dept. of Energy Technical Information Center, Oak Ridge, 1967.

6. “Radiological Health Handbook - Revised Edition,” U.S. Department of Health, Education and Welfare, Public Health Service, Document Number PB230846, January 1970. (Still available, as of 2/2011 from National Technical Information Service, Springfield, VA. Go to http://www.ntis.gov/ and enter the Document Number in the Search Field.)

7. C. Graham, “A Survey Instrument Design for Accurate β Dosimetry,” in Health Physics 52, pp. 485-489, 1987.

8. P. Frame and E. Abelquist, “Use of Smears for Assessing Removable Contamination,” Operational Radiation Safety, pp. S57-S66, May, 1999, (Supplement to Health Physics 76.

9. “Multi-Agency Radiation Survey and Site Investigation Manual,” U.S. Nuclear Regulatory Commission NUREG 1575, Washington, DC, December, 1997.

10. American National Standards ANSI N42.32 and 33 and 34 and 35 are available for download at http://standards.ieee.org/getN42/index.html.

11. “Multi-Agency Radiological Laboratory Analytical Protocols,” MARLAP, available for free download at http://www.epa.gov/rpdweb00/marlap/manual.html.

13

Radioactive Waste Management

Outline of this Chapter

Chapter Summary

In order to deal effectively with the management of radioactive waste, it is necessary to have some understanding of its sources and forms. The chapter begins with a discussion of the types and quantities of rad waste that originate from the use of radioactive materials in medical, university and industrial applications. The nuclear fuel cycle accounts for a substantial fraction of civilian high level and low level waste so it is treated in detail. Then, the disposal sites that have functioned over the years within the civilian and DOE sectors are described.

The actual management principles can be summarized very briefly. For high level waste, the principle is "concentrate and contain." For low level waste, the principle is "dilute and disperse." The next two sections of the chapter focus on the various technologies that have evolved to concentrate, contain, dilute or disperse radioactive waste.

Compaction and incineration are two practical techniques for concentrating solid rad waste. High level liquid waste is usually solidified after an initial period of tank storage. Much of the high level liquids resulting from nuclear weapons production has been converted to powdered calcine. The DOE has some capacity for converting calcine to a vitrified (glass) product, and large scale plants are in operation in South Carolina and New York.

Shallow-land burial is the method of choice for low level waste in both the civilian and DOE sectors. Details of the construction and operation of such facilities are presented. With cancellation of the high level repository at Yucca Mountain, Nevada, the U.S. has no current solution for storage of power reactor spent fuel. The DOE's WIPP deep underground geologic repository for transuranic waste has been operating since 1999 in New Mexico.

In the U.S., the management of low level waste from the civilian sector is now the province of a group of Nuclear Waste Compacts – groupings of states that have agreed to establish disposal capacity for the Compact members. These compacts have been responsible for taking the steps needed to open new low level burial facilities. Several new sites have been proposed but most are on hold for various reasons at this time. The West Texas site operated by Waste Control Specialists will be the first new site to actually be open for Class A through Class C waste in 39 years!

The chapter closes with a brief discussion of the "retirement" process for shallow-land burial sites. Financial guarantees and long term maintenance commitments are the keys to success in finally closing these facilities.

Sources and Disposition of Radioactive Waste

Radioisotope Use

Radioisotopes whether of natural, reactor or accelerator origin ultimately all end up in forms or activity levels which are no longer useful to the licensee. These then become radioactive waste or "radwaste." Not all radwaste sources are well documented as to the volumes and activities contained therein. As a result of increased interest in the overall radwaste situation (encouraged in part by the growing public concern over the "nuclear issue"), information is becoming available on some radwaste generators.

A survey was conducted in 1973 of all U.S. universities relative to volumes and composition of their radwaste. The results indicated that much of the volume was in the form of liquid scintillation fluids containing small concentrations of tritium and carbon-14. The typical volumes for an "average size" university were about 1,000 cubic feet per year of compressible solid radwaste and 500 gallons per year of liquids. As a result of more effort being expended to reduce volumes due to increased disposal costs, these averages are undoubtedly higher than current practice.

In 1978, the state of California conducted an extensive radwaste survey of medical licensees. Most of the hospital radwaste originates in the nuclear medicine department and from medical research activities. In this regard, it should be noted that federal regulations state, "Excreta from individuals undergoing medical diagnosis or therapy with radioactive material shall be exempt from any limitations contained in this section" (Disposal by release into sanitary sewerage systems). The normal forms for medical radwaste include paper products, disposable gloves, glassware and syringes. The California survey found that the overall volume generated per medical licensee was about 5 barrels (36 cubic feet) per year. Looking over the data, it became clear that the volume distribution for radwaste generated by California medical licensees was severely skewed. In other words, most of the volume was actually coming from just a few licensees. When the licensees generating less than 5 cubic feet per year were excluded, the average jumped to 15 barrels per year per hospital.

A 1981 study was conducted by the NRC as part of the environmental impact statement written for establishing requirements for land disposal of radioactive waste. In that year, there were about 20,000 U.S. licensees generating some amount of radwaste. Commercial generators of low-level waste (LLW) put out 85,000 m^3 (3 million cubic feet) per year. The study projected figures for radwaste generation for the years 1980 - 2000 to total 3.62 X 10^6 cubic meters. Of this total, the percentage breakdown indicated the following sources:

- 65% from the nuclear fuel cycle
- 6.5% from institutions
- 28.5% from other sources

It might also be noted that the civilian nuclear power industry in this country accounts for only 0.8% of the total high level waste (HLW) produced to date. The

remaining 99.2% came from military uses. Civilian sources account for 24% of the low level waste buried in this country. Although low level radwaste constitutes about 85% of the total volume of radioactive waste, it contains only 1% of the total radioactivity.

The Nuclear Fuel Cycle

Since such a large fraction of non-military radwaste comes from the use of nuclear energy to generate electricity, this source will be explored in some detail. The basic steps in the uranium fuel cycle are illustrated in Figure 1. At each step, radioactive wastes are generated. MINING AND MILLING involves those operations whereby the U_3O_8 ("yellow cake") is extracted from the ore. This leaves low grade crushed rock and sand as byproducts. In addition, hundreds of gallons of water are used for each ton of ore processed. This wastewater contains significant concentrations of radium and uranium. The waste products of the mining and milling steps are termed "mine tailings." In 1977, there were 17 operating uranium mills and 26 sites that were inactive or on standby conditions. Altogether, there were 138 million tons of solid tailings covering about 3,000 acres of land. In 2011 there was only a single operating mill in

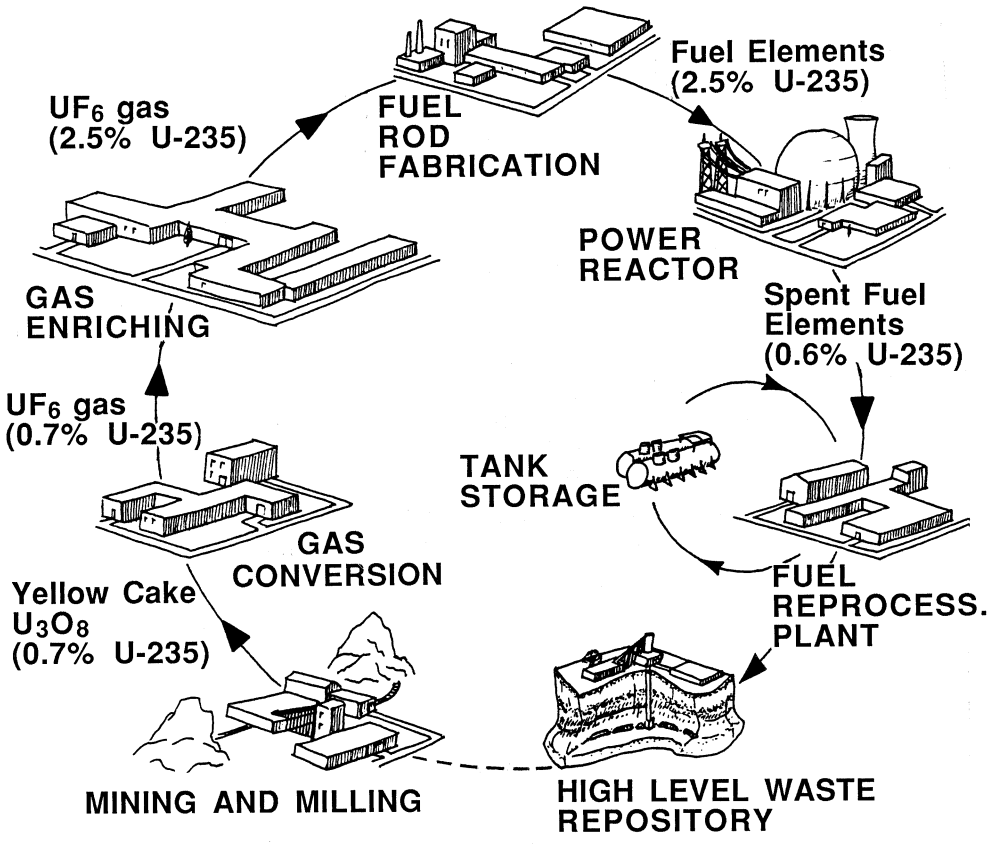


Fig. 1 - The uranium fuel cycle

the U.S. Three other mills are in standby mode. They could be placed back in service if the support for new power reactor construction continues.

In 1982, the U.S. Congress funded the Uranium Mill Tailings Remedial Action Program, or UMTRAP, under the direction of the Department of Energy. The purpose was to clean up the mill sites, reduce radon emanation from the tailings to acceptable levels by burying them and to restore the land to unrestricted use. The UMTRAP sites processed and supplied ore from domestic mines to the former Atomic Energy Commission. More recently, two other categories of sites have been identified – Formerly Utilized Sites, FUS, and Surplus Facilities, SF. Remedial Action Programs have been underway for them as well. The FUS were contractor facilities processing uranium ore from Africa for the Manhattan Project and the SF are sites currently under the supervision of the DOE or its contractors but are being restored for unconditional release.

One of the earliest remedial action projects (RAP) was the Grand Junction RAP. From 1951 – 1970 the Climax Uranium Co. operated a uranium mill in Grand Junction, Colorado. Between 1952 and 1966 several hundred thousand tons of mine tailings were used in the area for construction and landfilling. In 1972, federal guidelines were established and Congress authorized a RAP for 600 properties that exceeded the guidelines for radiation levels. The federal government paid 75% of the cost and Colorado picked up the balance. Generally, contaminated soil was excavated from underneath structures and replaced with uncontaminated material (see Figure 2). The project was completed in 1987.

The map in Figure 3 shows the locations of the uranium mill sites that were part of the UMTRAP. The affected states paid 10% of the cost with the federal government paying the balance. In addition to the 24 mill sites, an additional 4,800 individual properties had been identified near the sites which required decontamination. The tailings were stabilized and

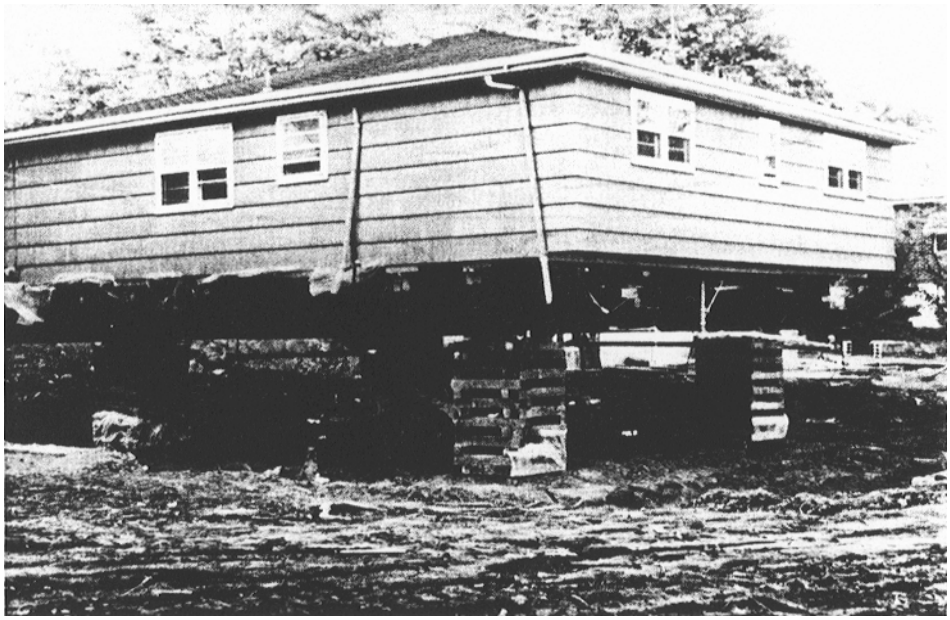
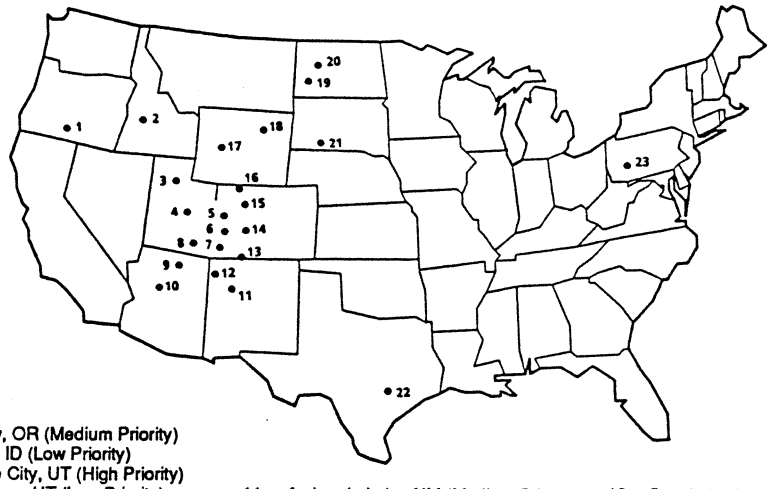


Fig. 2 - Example of soil excavation to remove tailings



1 Lakeview, OR (Medium Priority)
2 Lowman, ID (Low Priority)
3 Salt Lake City, UT (High Priority)
4 Green River, UT (Low Priority)
5 Grand Junction, CO (High Priority)
6 Naturita, CO (Medium Priority)
7 Slickrock, CO (Low Priority) (2 sites)
8 Mexican Hat, UT (Medium Priority)
9 Monument, AZ (Low Priority)
10 Tuba City, AZ (Medium Priority)
11 Ambrosia Lake, NM (Medium Priority)
12 Shiprock, NM (High Priority)
13 Durango, CO (High Priority)
14 Gunnison, CO (High Priority)
15 Rifle, CO (High Priority) (2 sites)
16 Maybell, CO (Low Priority)
17 Riverton, WY (High Priority)
18 Spook, WY (Low Priority)
19 Bowman, ND (Low Priority)
20 Belfield, ND (Low Priority)
21 Edgemont, SD (High Priority) (Vicinity Properties Only)
22 Falls City, TX (Medium Priority)
23 Canonsburg, PA (High Priority)

Fig. 3 - Locations of UMTRAP sites

then buried under several layers of top covering materials to reduce radon emissions and limit erosion (see Figure 4). Upon conclusion of the RAP, the tailings piles had less than 5 pCi/g of radium and thorium in the topmost soil layer. They produced radon levels less than 0.02 Working Level in any habitable building in the vicinity and did not raise the background gamma ray level more than 20 μR/hr inside any nearby habitable structure.

For the sake of interest, one ton of uranium ore usually yields between 2 and 3 kilograms of yellow cake. Yellow cake has 95% of the uranium contained in the original ore but only 14% of the total radioactivity.

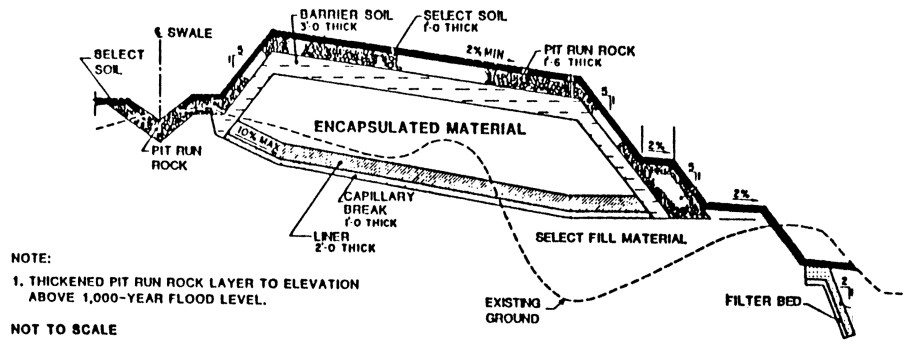


Fig. 4 - Encapsulation scheme for mine tailings

The POWER REACTOR stage in the overall fuel cycle is the largest civilian generator of radioactive waste at the present time. A nuclear reactor produces radwaste in all three physical forms, solid, liquid and gas, and it produces both high level and low level waste relative to contained radioactivity. Figure 5 shows the average volume and activity of LOW LEVEL radwastes shipped off-site or released to the environment by all operating PWRs and BWRs in the U.S. during 1985 (a year chosen to include HTGR data). The PWR data includes 62 plants and the BWR averages are based on 33 plants. Also shown is data for the single High Temperature Gas Cooled Reactor, HTGR, that was operating in this country. It was permanently shut down in 1989.

From Figure 5 it is evident that there are generic differences in the radwaste production between the two major plant types. BWRs generate much more low level trash due to the fact that primary coolant water flows throughout the entire loop including the turbine. In the PWR, primary coolant flows only in the first loop. PWRs generate much more tritium as a result of the boric acid solution (chemical shim) added to the primary coolant to adjust reactor reactivity. As boron nuclei pass through the core, they capture neutrons. This sometimes leads to the formation of a tritium nucleus following emission of two alpha particles.

SOLID RADWASTE

PLANT TYPE	VOLUME(m^3)	ACTIVITY(CI)	# TRUCKLOADS
PWR	292	1056	23
BWR	949	15739	71
HTGR	110	419	6

LIQUID RADWASTE

PLANT TYPE	Activity Released (Ci) [not including H-3]	H-3 Activity Released (Ci)
PWR	1.6	457
BWR	0.9	10
HTGR	0.002	15

AIRBORNE RADWASTE

PLANT TYPE	Gaseous (Ci)	Particulates & I-131(Ci)
PWR	3308	0.123
BWR	10400	0.211
HTGR	2	0.000

Fig. 5 - Average LLW generated at U.S. nuclear power stations

There has been a consistent trend in recent years of reduced annual volumes of solid low-level waste from both PWRs and BWRs. Figure 6 shows the data from 1980 through 1998.

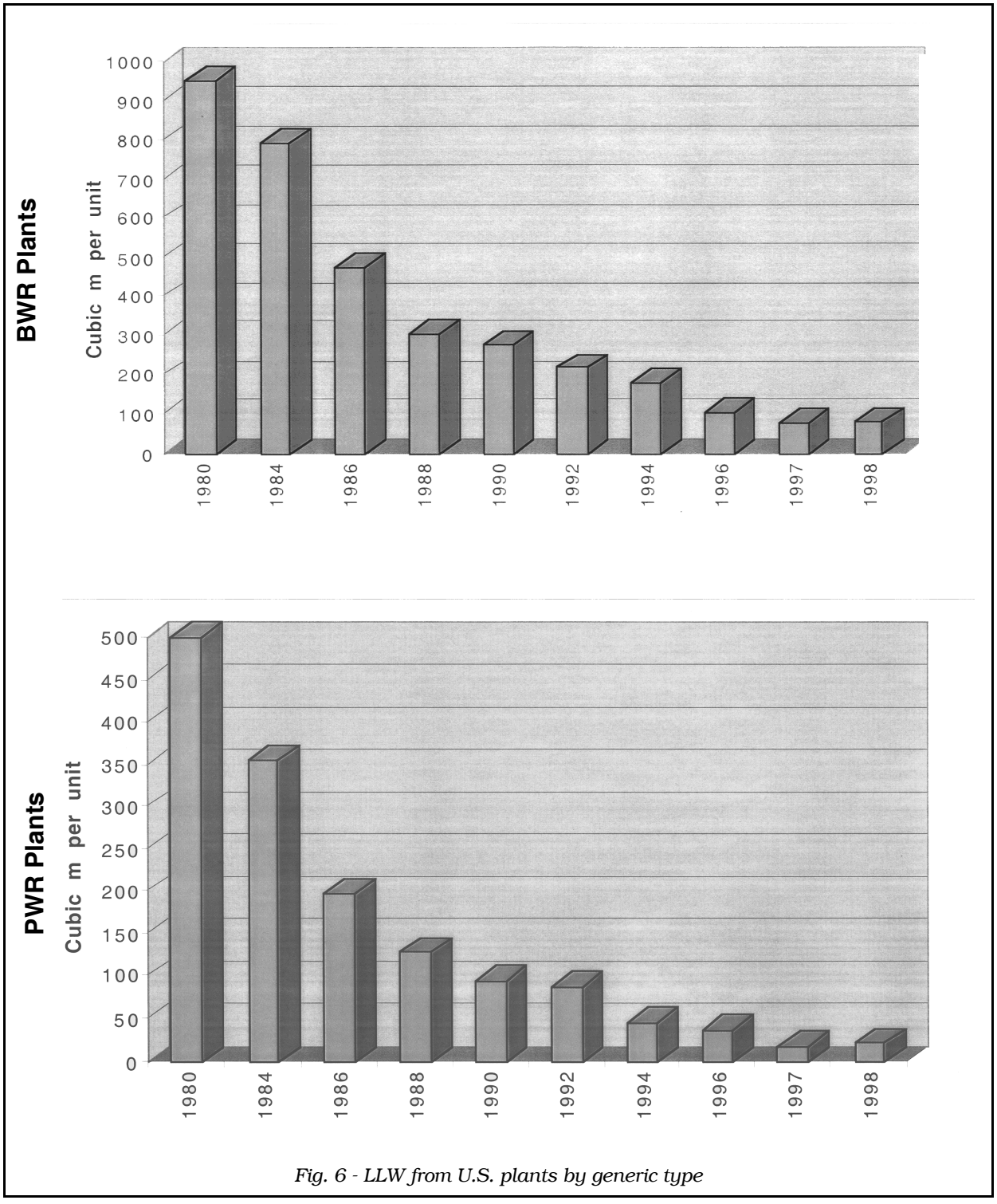


Fig. 6 - LLW from U.S. plants by generic type

To put the power reactor radwaste generation into perspective, it is an interesting exercise to calculate per capita volume. The high level fission product waste from a "standard" 1,000 megawatt electric nuclear plant amounts to 2.7 ft^3 per year in unprocessed form (contained within the fuel pellets). When this waste is finally solidified into a glass product, the final annual volume per plant amounts to 70 ft^3. If the complete total lifetime electrical needs of each person in the USA were met exclusively by nuclear generated electricity, then the total volume of fully processed high level radwaste in glassified form would amount to 150 cubic cm, about 1/2 cup per person. Actually, in this country, nuclear energy accounts for only about 20% of the total U.S. electrical capacity.

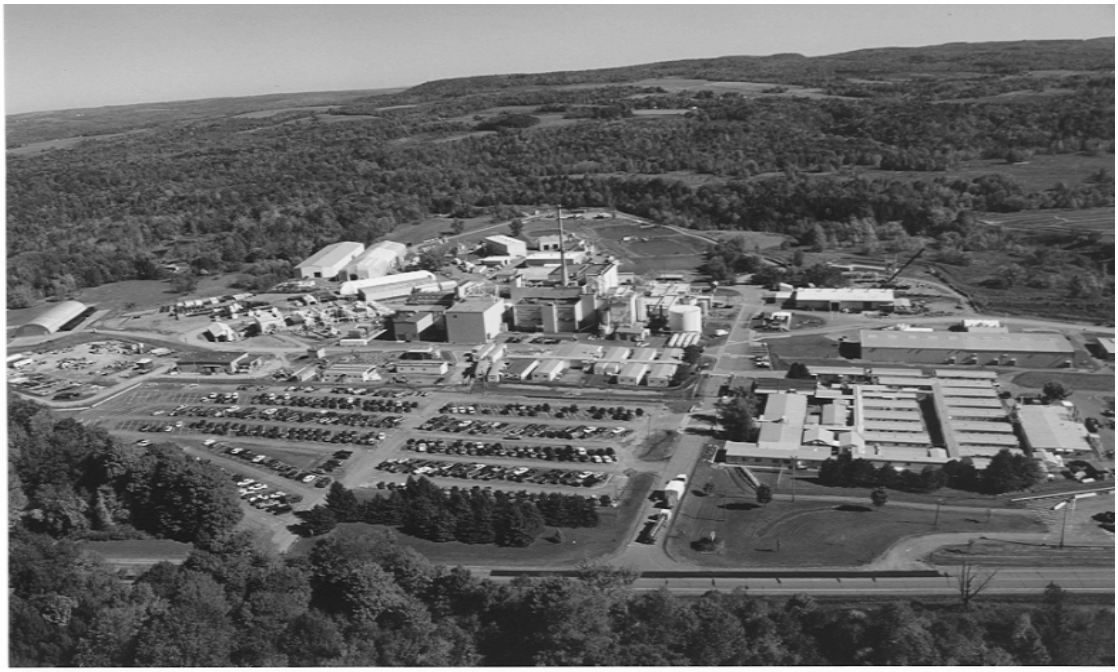
A second way to look at the reactor radwaste problem is to realize that if all of the high level waste ever produced by all of the U.S. commercial nuclear power stations was processed into glassified final form, the combined volume stacked up on a standard football field would be less than 6 feet high.

The final stage of the fuel cycle to be considered here is fuel reprocessing. This involves a series of steps. First, the fuel element (a bundle of individual fuel rods or "pins") is disassembled into rods. Then the rods are mechanically chopped into short sections. Next, the outer Zircaloy cladding is removed, followed by placement into a nitric acid bath which dissolves the uranium. The final steps in reprocessing involve recovering the useful elements from the acid solution. Four separate recovery operations are conducted. Each uses the chemical process of solvent extraction in which the desired elements are separated into layers using tributyl phosphate solutions. The four extractions remove uranium, plutonium, the transuranic elements ("actinides") and the fission products as separate solutions. The last two fractions are the high level liquid waste. The first two are recycled into useful materials. The liquid waste is produced at a rate of about 400 liters per ton of reprocessed fuel. The 400 liters contain between 40 kg and 90 kg of waste radionuclides at a radioactivity concentration of between 3,000 and 5,000 Ci/liter.

To date, there has been only one commercially operated spent fuel reprocessing facility in the U.S., located near West Valley, New York. This high level waste (HLW) plant was in production from 1966 through 1972 and processed 640 tons of spent fuel. At the conclusion of operations, about 560,000 gallons of high level liquid waste remained at the site, and was stored in a carbon steel tank inside a concrete vault. Then, in 1980, Congress approved the West Valley Demonstration Project, Figure 7, to fund the solidification of the waste, transportation to a federal repository and decontamination of the site. The liquid waste was to be incorporated into borosilicate glass and sealed in special canisters, 10 feet long by 2 feet in diameter as shown in Figure 8.

Between 1985-89, the glass-making system was tested with a non-radioactive product. In addition, the liquid phase from the storage tank was run through an ion exchange process to remove the majority of the radioactivity. The cleaned liquid was blended with cement and put into 71 gallon steel drums for disposal as low level waste. This produced about 20,000 drums of cemented waste

The second phase of the project involved mixing the ion exchange materials and the sludge from the tank bottom and then feeding the mixture into a 52 ton ceramic melter for conversion to glass. This equipment



Courtesy, U.S. Dept. of Energy

Fig. 7 - View of the West Valley Demonstration Project site

Courtesy, U.S. Dept. of Energy

Fig. 8 - West Valley canister for vitrified high level radwaste

was tested in 1994 and began processing high level radioactive waste in 1996. It was anticipated that the second phase would take 3 years to finish. As of 2005, all of the sludge phase, including tank bottom scrapings, has been converted to glass, producing 250 canisters. The vitrification cell has been decommissioned and decontaminated, the first such project in the U.S.

Final plans for the site have not yet been determined. Considerations range from complete demolishment of all site facilities to a less drastic approach in which the buildings can be decontaminated and converted to some other use. More details of the West Valley project will be covered later in this Chapter.

Existing LLW Disposal Sites

As of 2000, the latest year for which figures were available, the total volume of LLW buried at commercial United States sites was approximately 56.7 million cubic feet. The U.S. NRC established a classification system for LLW a number of years ago. The system defines three classes - A, B and C. Class A has the least radioactivity while Class B and Class C have progressively higher activity. Further details and class limits will be discussed later in this Chapter.

The annual buried LLW volumes showed a steady decline from the 1980s until the late '90s as illustrated by Figure 9. At that point, decommissioning activities at several large sites pushed the annual totals to several million cubic feet. Figure 10 shows the activities buried at commercial sites over the same time frame. See Sample

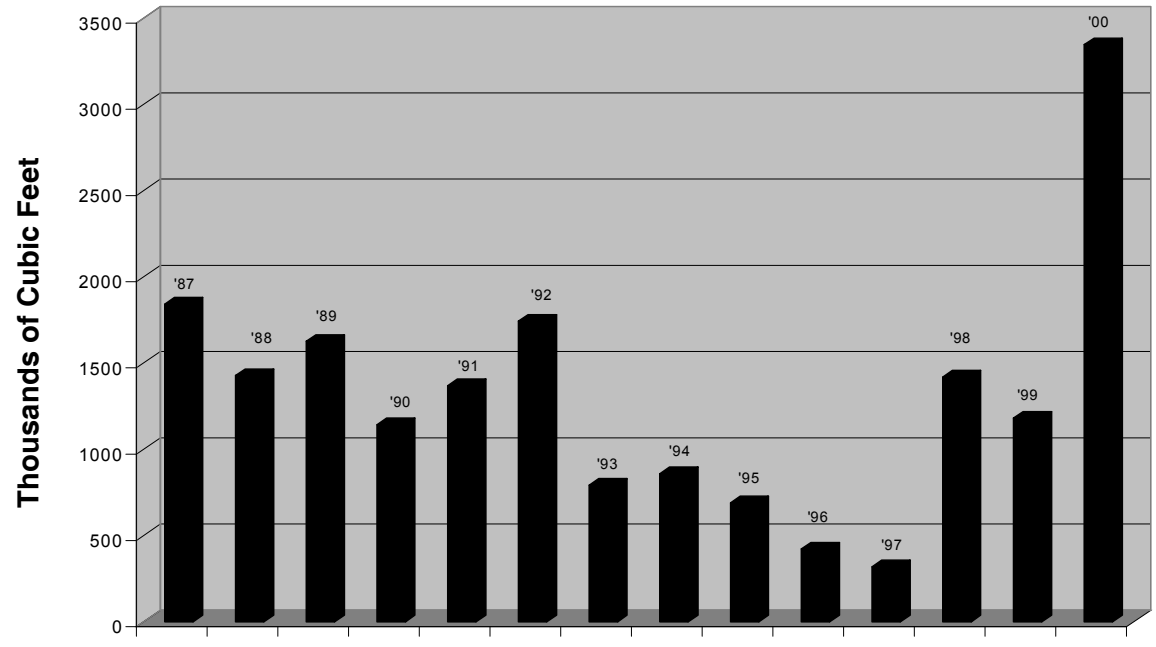


Fig. 9 - Buried volumes of commercial low level waste (LLW)

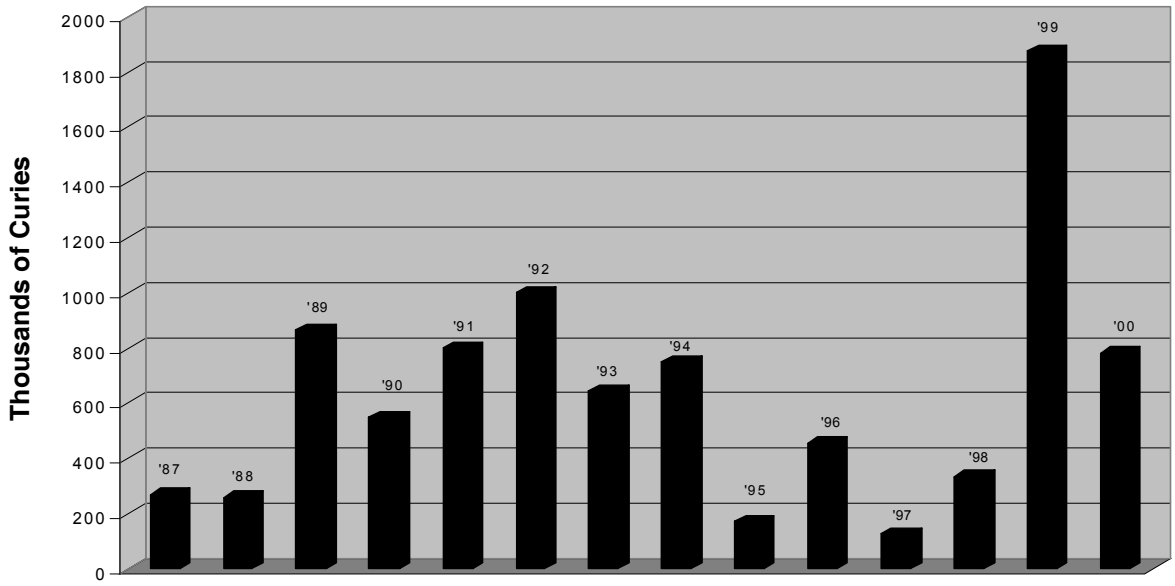


Fig. 10 - Activities buried at commercial LLW sites in the U.S.

Problem 1. As of 2010, the NRC states that current LLW waste volumes continue in the range of several million cubic feet per year.

After a 38 year drought, a major breakthrough occurred in 2009. A new commercial U.S. low level waste site was licensed for all three classes of LLW. See Figure 11. The Waste Control Specialists site is located near Andrews, Texas, on a 1,338 acre

Fig. 11 - The Waste Control Specialists LLW site in West Texas

Sample Problem 1

GIVEN:

Assume that all commercial LLW was buried in 55 gallon drums in 1997.

FIND:

What was the average activity per drum buried that year? How did it compare to 1987?

SOLUTION:

The activity/drum = Activity per 55 gal x 0.1337 cu ft/gal = Activity/ 7.4 ft^3. From Figs. 9 and 10, in 1997 the activity was 1.0 x 10^5 Ci and the volume was 3.5 x 10^5 ft^3. So the average activity/drum in 1997 was 1.0 x 10^5 Ci x 7.4 ft^3/3.5x 10^5 ft^3 or 2.1 curies. In 1987 the number was 2.3 x10^5 Ci x 7.4 ft^3 / 1.8 x 10^6 ft^3 = only 0.95 curies. Compaction pays, folks!

parcel. It was licensed to hold up to 2.3 million cubic feet of LLW, with the condition that 20% of that volume was reserved for the state of Vermont. The site is accessible by both trucks and rail cars. Geographically, this West Texas region receives only 9 inches of precipitation annually. Groundwater is at a minimum depth of 800 feet, and the soil is red clay. As of 2011, construction was proceeding on schedule, with an expected opening date late that year.

COMMERCIAL SITE	OPERATOR	YR LICENSED	2011 STATUS
Barnwell, So. Carolina	Chem-Nuclear	1971	Open
Beatty, Nevada	US Ecology	1962	Closed 1/1/93
Hanford, Washington	US Ecology	1965	Open to some
Maxey Flats, Kentucky	US Ecology	1962	Closed
Sheffield, Illinois	US Ecology	1967	Closed
West Valley, New York	Nuclear Fuel Services, Inc.	1963	Closed
Clive, Utah	Energy Solutions	1991	Class A only
Ward Valley, Calif.	US Ecology	1993	Terminated!
Andrews, Texas	Waste Control Specialists	2009	Opening Fall 2011

Fig. 12 - Past and present commercial U.S. LLW sites

The earliest commercial disposal site licensed for LLW in the United States opened in 1963. Within the next eight years, five additional sites in the U.S. were licensed and opened. (This does not include the Energy Solutions, UT site which only takes Class A waste.) All commercial sites and their current status are shown in Figure 12. The Department of Energy also maintains both high level waste and low level waste sites. Some information on the DOE sites that have operated in the past or are currently accepting waste is given in Figure 13.

In 2010, there were only three active commercial LLW sites and no active HLW sites in the U.S. Since 2008, the Barnwell, SC facility has accepted waste only from the Atlantic Compact States. It can receive all three classes of LLW. The Hanford, WA site is only accepting waste from the Northwest Compact and the Rocky Mountain

LOW LEVEL SITES

Hanford, Washington	**Oak Ridge National Lab, TN**
Idaho Falls, Idaho	**Sandia Laboratory, New Mexico**
Los Alamos, New Mexico	**Savannah River Site, South Carolina**
Nevada Test Site	**7 smaller sites, various locations**

HIGH LEVEL SITES

Location	**Waste Forms**
Hanford, Washington	**Liquid, salt, sludge**
Idaho Falls, Idaho	**Liquid, calcine**
Savannah River, So. Carolina	**Liquid, salt, sludge**

Fig. 13 - DOE radwaste sites

Compact. It, too, can receive all three waste classes. In contrast, the Clive, UT site is open to all U.S. licensees, but can only accept Class A radioactive material. Late in 2011, the West Texas facility, discussed above, will hopefully be added to the list.

Radioactive Waste Management Principles

The basic principles of dealing with radioactive waste are relatively simple. The application of these principles involves very complex technology and difficult political decisions. Generally, radwaste having an intermediate level of radioactivity (roughly, anything with an activity higher than microcuries/gallon and lower than curies/gallon) cannot be economically dealt with. This intermediate level radwaste is either

concentrated to produce high level or diluted to produce low level. The basic principles are then stated as follows:

HIGH LEVEL: CONCENTRATE & CONTAIN

LOW LEVEL: DILUTE & DISPERSE

PRINCIPLE	PHYSICAL STATE			CATEGORY
	SOLID	LIQUID	GAS	
CONCEN-TRATE:	Crush	Evaporate	Adsorb	HIGH Level Waste
	Incinerate	Ion Exchange	Pressurize	
		Precipitate	Cryo. Separate	
			Freon Absorption	
CONTAIN:	Rock Burial	Concrete Tank	Cylinder	
	Salt Burial	Solidify: calcine glass ceramic metal matrix		
	Rocket to Sun			
	Nuclear Test Hole			
DILUTE:	Hold for Decay	Hold for Decay	Holdup Tank	LOW Level Waste
	Bioremediation	River Dilute		
	Phytoremediation	Ocean Dilute		
DISPERSE:	Ocean Burial	Sanitary Sewer	Stack Release	
	Shallow Land Burial	Ground Release		

Fig. 14 - Basic principles of radwaste management

The basic objective is to prevent human contact with dangerous concentrations of radioactivity or unsafe radiation levels. In the case of the high level radwaste, this means isolating the material from the biosphere. In the case of low level radwaste, the concentration is reduced to a "safe" level and the material can then be released. Figure 14 is a chart which lists a variety of techniques in use to concentrate, contain, dilute or disperse radioactive waste products in the form of solids, liquids and gases. Some of these technologies will be discussed in the next section.

After many years of tossing virtually anything contaminated with radioactivity into the same container and sending it off to the burial ground, the realization finally struck that "business as usual" was no longer a viable alternative. Both from a moral/environmental point of view (we are running out of land area for shallow-land burial sites) and an economic one (everyone knows about the skyrocketing costs of LLW burial) it now makes sense to perform some kind of waste separation into different categories right at the point of generation. After much thought, the U.S. NRC came up with a classification scheme that was incorporated into the 10 CFR Part 61 regulations **Licensing Requirements for Land Disposal of Radioactive Waste**.

Class A segregated waste is usually kept separate from other classes at the burial site. It has the lowest allowed concentrations of radioactivity (see Figure 15). It cannot contain more than 1% freestanding liquid, is not accepted in cardboard or fiberboard boxes, it must be nonexplosive, not biologically hazardous and not pyrophoric. According to the NRC, about 96% of all LLW is Class A. Class B stable waste can contain higher concentrations of radioactivity (B is higher than A but less than C) and must meet special conditions as to form "to ensure that the waste does not structurally degrade and affect overall stability of the site through slumping, collapse or other failure of the disposal unit and thereby lead to water infiltration." The necessary structural integrity can be assured by the waste form itself (e.g., concreted liquid waste), by further processing of the waste or by placement inside a structurally approved container.

Finally, Class C intruder waste has the highest allowed concentrations. It must meet the stability requirements of Class B but extra precautions must be taken by the disposal site to "protect against inadvertent intrusion." The NRC was concerned about the possibility of future generations establishing home sites on an ancient abandoned

Isotope	Class A Max. Ci/m^3	Class C Min. Ci/m^3
Half life < 5 years	700	70,000
H-3	40	1 E 08
C-14	0.8	0.8
Co-60	700	70,000
Sr-90	0.04	150
Tc-99	0.3	0.3
Cs-137	1	44
Enriched Uranium	0.04	0.04
U_{nat} or Depleted U	0.05	0.05

Fig. 15 - 10 CFR Part 61 classification of rad waste

low level waste facility. At the time of burial of Class C waste, various barriers are installed to discourage future digging up of the waste.

Applied Processing Techniques

Concentration of Solids

Two techniques have found use in increasing the concentration of solid radioactive waste. The first one, COMPACTION involves crushing and baling under compression. This process has been found to produce a typical volume reduction of 3 to 7 times for the usual forms of contaminated trash. A common compactor receptacle is a standard 55 gallon DOT approved steel drum used to transport the waste. By compacting directly inside the shipping drum, one additional handling step is eliminated,

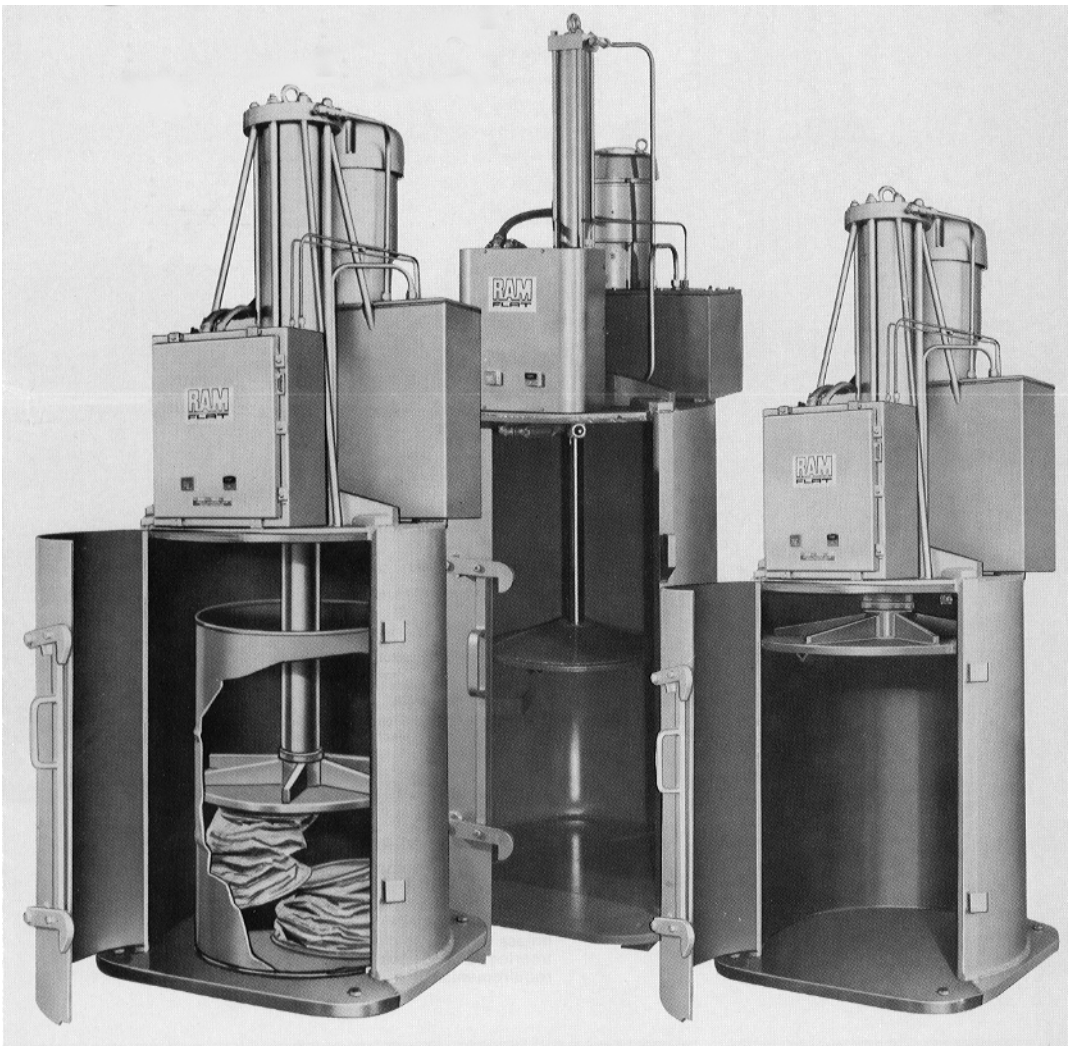


Fig. 16 - Radwaste compactors

Courtesy, S&G Enterprises

thus reducing doses to the technologists involved. Figure 16 shows different commercially available models designed to work into shipping drums. The two models on the left use a compaction force of 42 tons while the model on the right uses 10 tons. They operate with a hydraulic ram which takes about one minute per cycle.

A new class of compactor has become commercially available – the "super compactor." Figure 17 shows such a machine which is capable of applying a force of 1,500 metric tons! The world's first such machine was installed in the Netherlands in 1978. The first super compactor installed permanently in the U.S. (designed and manufactured by Machinefabriek A. Fontijne B.V. and Stock Equipment Company) was operated by the Scientific Ecology Group in Pennsylvania. It went into operation in 1986. The press can process thirty 55 gallon drums per hour. All operations are computer controlled and fully automated. The press working area and the auxiliary mechanical equipment used to handle drums are contained in a negative pressure cell. Following compression, the collapsed drums are loaded into a DOT approved "overpack" container for shipment to a burial site. The computer individually measures the height of each compressed drum and sorts them into one of six overpacks so as to maximize

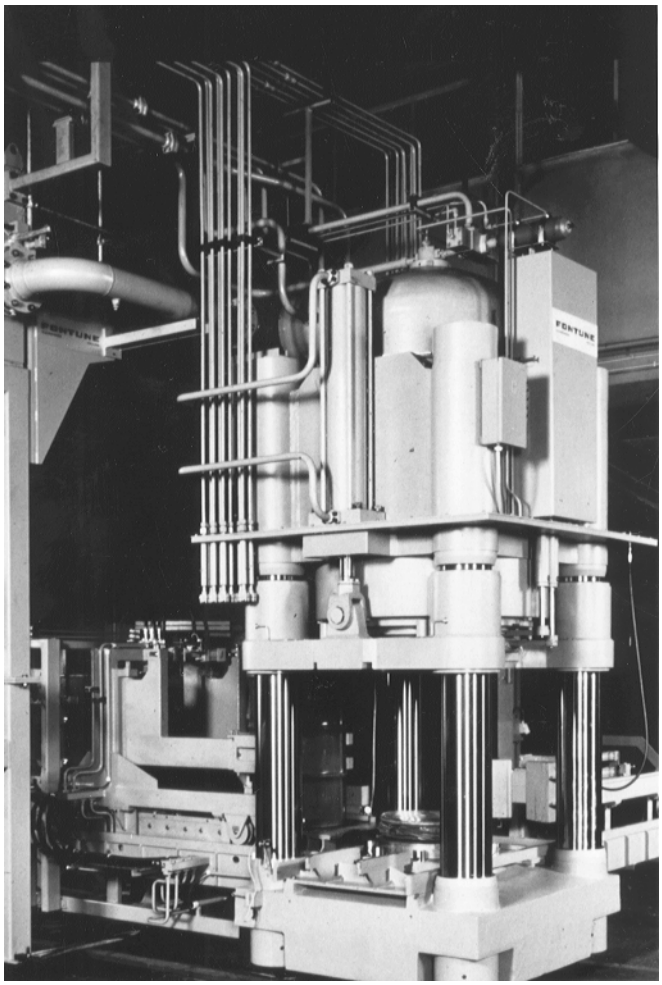


Fig. 17 - A "super compactor"

Courtesy, Stock Equipment Company

the volume before shipment. Figures 18 and 19 show a 55 gallon drum before and after super compaction.

The radiation protection problem posed by compactors is the production of airborne activity as a result of the crushing operation. Thus, compactors must be fitted with an air suction and filter assembly to remove the generated particulates. In addition, consideration should be given to the increased external radiation level associated with compacted waste, particularly when using a super compactor. The act of compressing the waste means that the specific activity (becquerels per gram) is increased, thus, increasing the exposure rate from the final package.

The technique of INCINERATION is the second method used to concentrate solids. This method is particularly useful with animal carcasses which are radioactive. Burning waste normally gives a better volume reduction than ordinary compaction – a 10 to 15 times reduction is common. However, this process produces radioactive gases and particulates in addition to the solid ash residue. Extensive gas cleanup and filter systems are needed with incinerators. Although air quality standards and public pressure have made incinerator operation difficult in the past, the rapidly escalating costs for burial services and the shrinking available land area at authorized burial sites will probably cause more use of this option in the future. See Figure 20 for a sketch of a radwaste incinerator.

One novel scheme which was proposed in 1983 by a European consortium to get around some of these problems was to mount incinerators

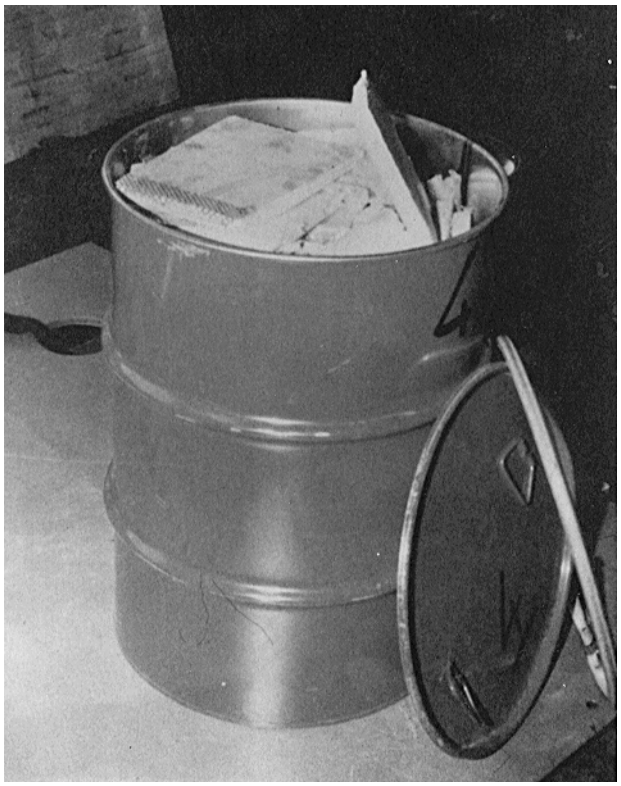
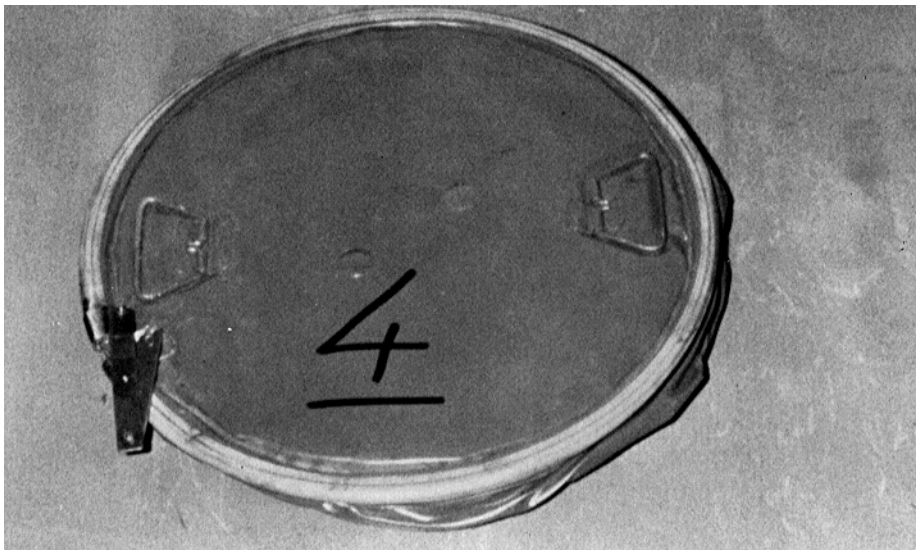


Fig. 18 - A waste drum filled with wallboard

Courtesy, Stock Equipment Company



Courtesy, Stock Equipment Company

Fig. 19 - Same drum as Fig. 18 following supercompaction

for radwaste aboard oceangoing vessels. The actual burning operations can then be conducted on the high seas in international waters!

Volume reduction techniques have proven to be very successful. Commercial generators buried over 3.7 million cubic feet of low level rad waste in 1980. Ten years

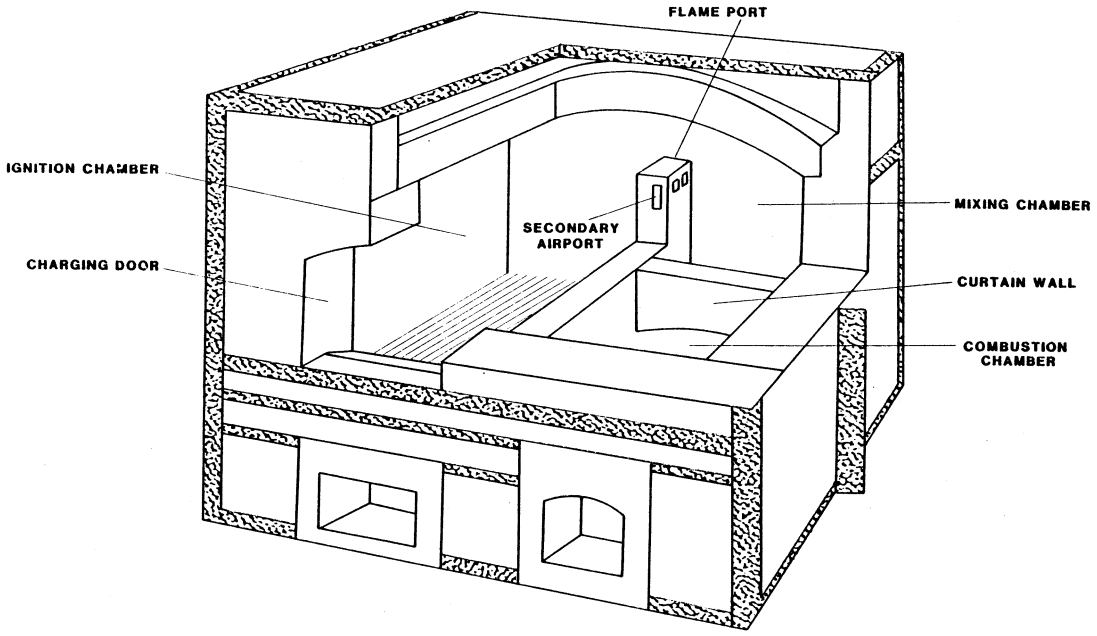


Fig. 20 - An incinerator for burning radwaste

later, the annual commercial volume had shrunk to 1.14 million cubic feet. In 1997 it was down to 0.32 million cubic feet. As detailed earlier, nuclear power production accounts for the majority of this waste. The nuclear utilities managed to reduce their annual volume by 60% while the number of licensed plants increased by two-thirds over the ten-year time span 1980 to 1990.

Solidification of High Level Liquids

Dealing with the high level liquid waste from uranium fuel reprocessing involves a series of steps. These steps are summarized in Figure 21. They will then be covered individually below.

An initial VOLUME REDUCTION is needed to reduce the overall magnitude of the storage problem to manageable proportions. The construction and operation of storage "tank farms" is very costly, so any initial reduction of volume is cost-effective. Evaporators are commonly used for this step. Based on actual plant experience, the acidic waste solutions from reprocessing can normally be concentrated between 10 and 50 times by evaporating in a large heated vessel. The practical limit is reached when solids begin to precipitate out of solution.

A typical evaporator plant vessel has a volume of around 10 cubic meters. The evaporator is located inside heavy shielding walls (see Figure 22). The solution is heated by steam coils (visible in the drawing near the bottom of the vessel) to a temperature of about 115° C. The vessel is usually operated at reduced pressure which helps reduce vessel corrosion. In addition, the residual tributyl phosphate from the reprocessing solvent extraction stage forms an explosive gas at temperatures above 130° C. Also, most evaporators have the capability for cooling the solution if that becomes necessary. The vessels are constructed of stainless steel or titanium. Design features allow for remote maintenance operations.

The next step, TANK STORAGE, is utilized in the overall management of the waste to allow for substantial radioactive decay before the solidification is attempted. If a solid is formed too early, the actual temperature in the center of the solid product will be so high that fractures will probably occur. This will increase the surface area and thus increase the chances of the radioisotopes being leached and transported

Initial Volume Reduction
Tank Storage
Solidification
Evaporation
Nitrate Removal
Calcination
Vitrification
or
Ceramic Formation

Fig. 21 - Solidification steps for high level liquids

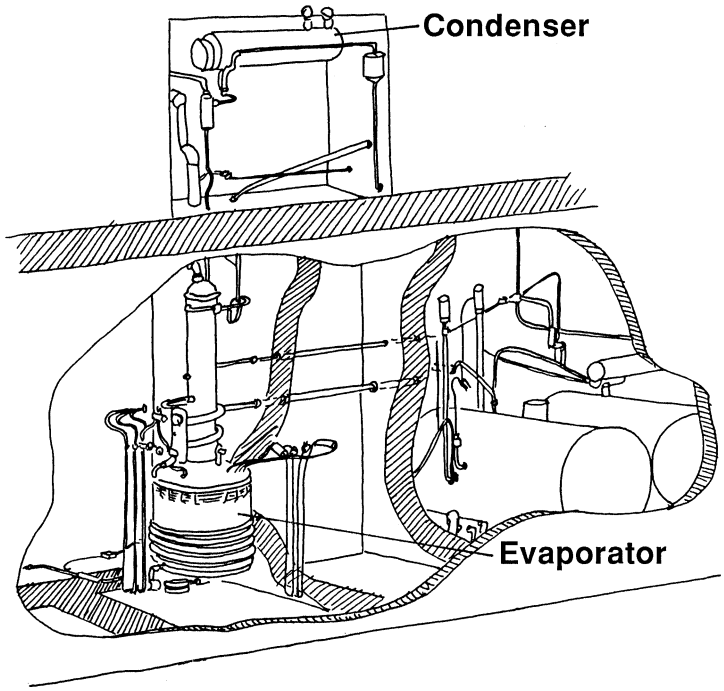


Fig. 22 - Sketch of a European evaporator plant

away by groundwater (assuming a geological depository as the final resting place). This temperature rise is, of course, produced by the decay heat of the radionuclides, i.e., the energy deposited in the solid product by the absorption of some of the decay radiations. Figure 23 is a graph of the decay heat produced in high level fission product waste. After 5 years, about 0.5% of the initial heat production rate is still present. The curve then reaches a slight plateau until the strontium and cesium isotopes decay away. In the United States, federal law requires that any liquid waste from commercial reactor fuel reprocessing must be converted to solid form within 5 years after reprocessing. This period is spent in a tank farm.

Storage tanks for high level liquid radwaste have several uncommon features. They must be equipped with a cooling system to maintain the solution at an acceptable temperature. The decay heat produces enough thermal energy to boil the liquid if it were not cooled. The higher the temperature, the higher the corrosion rate in the tank wall. As a compromise, the tanks are usually held between 60 and 65° C. A 100 m^3 (25,000 gallon) storage tank produces up to 2 megawatts of thermal power (over 7,000,000 BTU/hr) from radioisotope decay heat. From a cost perspective, the optimum tank size is about 1,000 m^3 in size and is constructed of stainless steel. Individual tanks are raised off the ground and actually contain redundant internal cooling coils as a backup in case of failure of one set. Tank farms are sometimes built underground to save on shielding costs. Figure 24 is an artist's sketch of a portion of a tank farm.

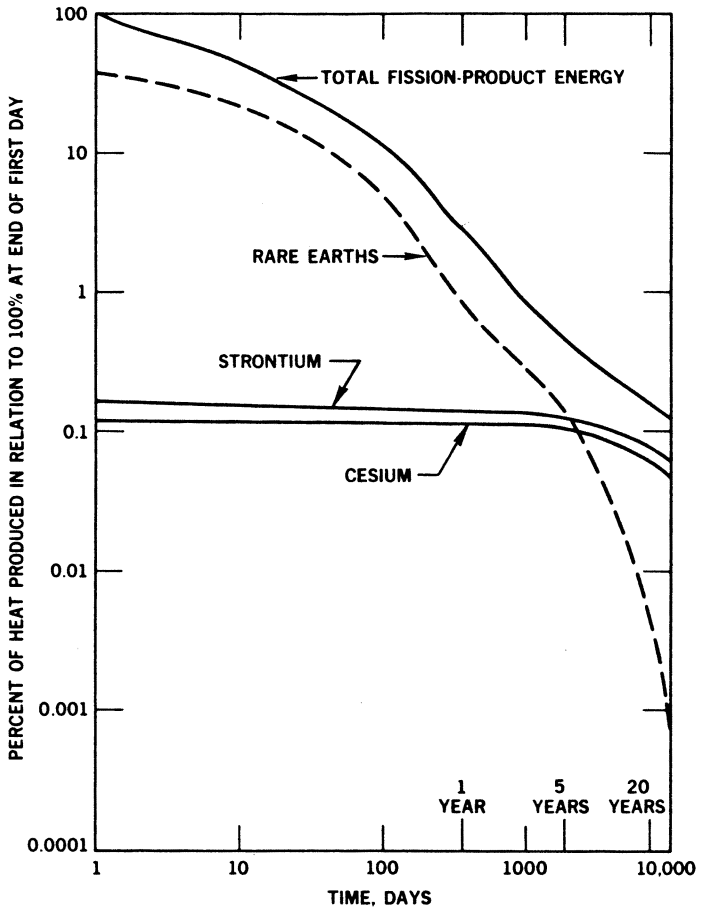


Fig. 23 - The decay heat from high level reactor radwaste

The liquid must be kept in continual motion to prevent settling out of precipitates. This is done by a "sparging system" which uses compressed air to create a circulating current in the tank. The control room for the tank farm contains instruments to monitor tank temperatures, liquid levels, and radioactivity monitors for sump liquids, cooling loops and effluent releases.

The SOLIDIFICATION steps are then carried out after the decay period in the tank farm. The basic process steps were listed earlier in Figure 21. The liquid waste is evaporated and then the nitrates are removed chemically. Next is the calcination stage. Calcine is the name given to the mixture of fission product oxides that result when the liquid waste is heated in the presence of oxygen to about 400° C. The powdered calcine represents an interim solid form for the high level waste (see Figure 25). It occupies only 1/8 the storage space of the liquid waste it replaces.

The world's first full-scale plant for this process was the Waste Calcining Facility located near Idaho Falls on the Idaho National Engineering Laboratory DOE site. It went into operation in 1964 and was finally "retired" in 1981. The reprocessing plant was operated by Westinghouse Idaho Nuclear Co., WINCO. The actual calciner vessel was about 1.2 meters

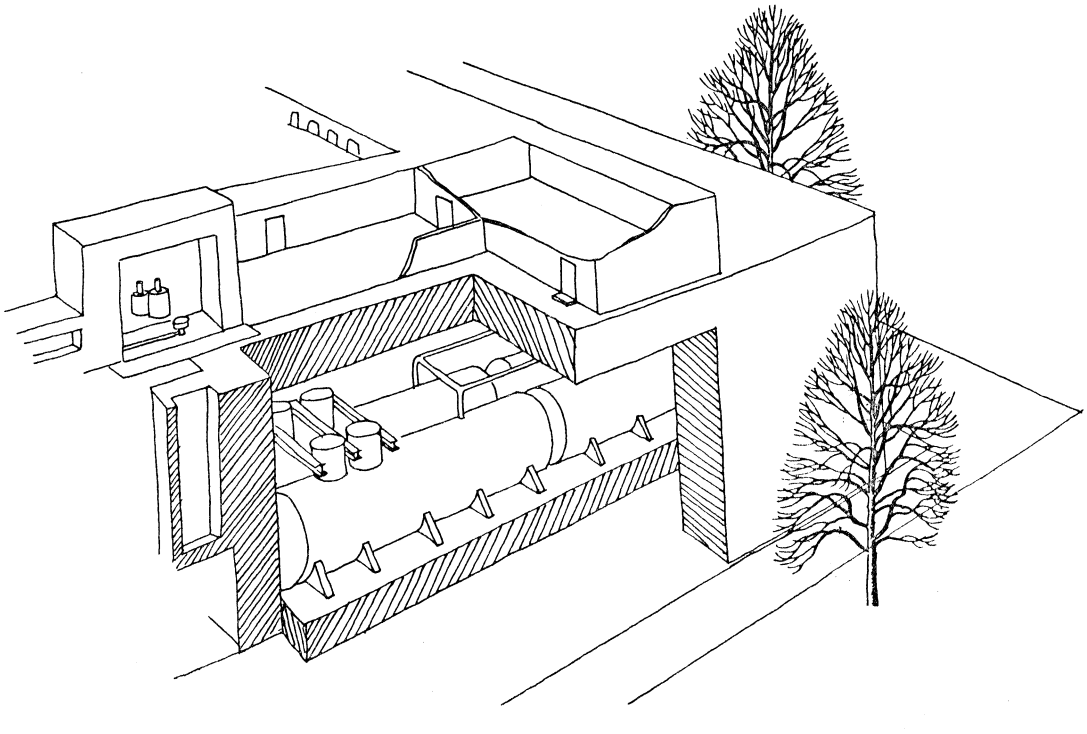


Fig. 24 - Sketch of a tank farm for high level liquids

in diameter and was able to process 1400 gallons of high level liquid waste into calcine each day.

The New Waste Calcining Facility opened in 1982, incorporating many design improvements based on knowledge learned at the old plant. It processes 3,000 gallons per day and employs nearly 1,500 WINCO personnel.

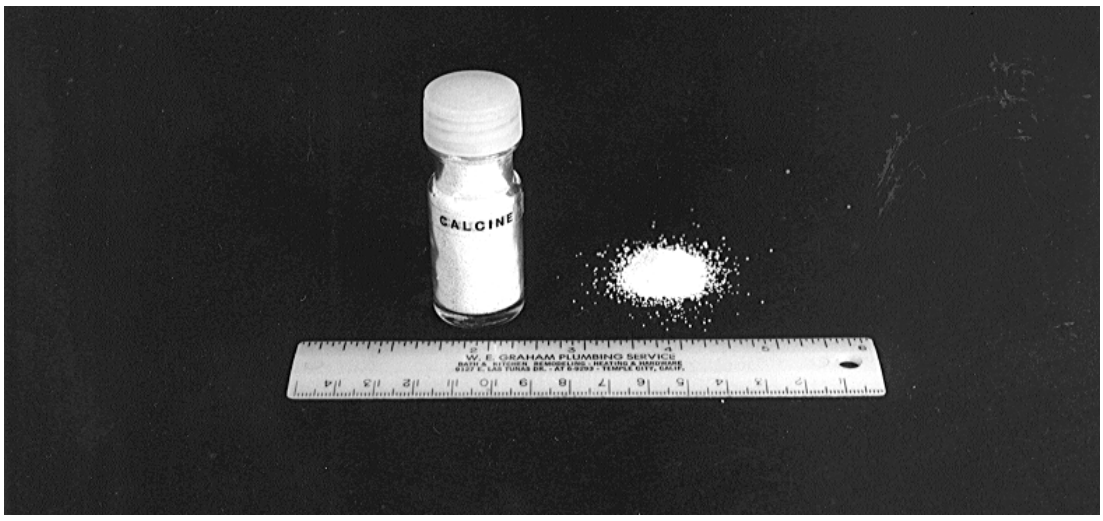
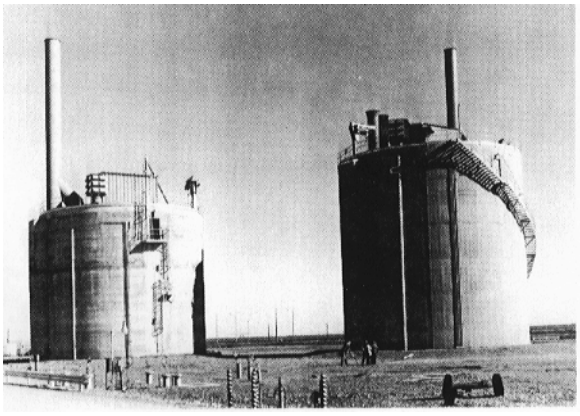


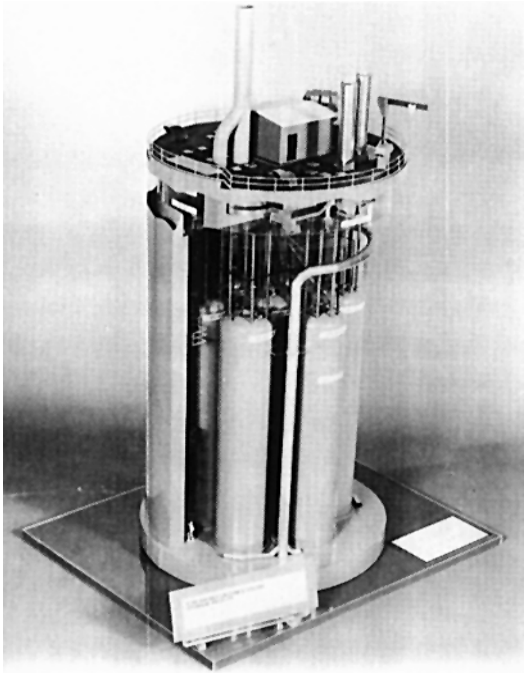
Fig. 25 - A sample of powdered calcine

Courtesy U.S. DOE

Fig. 26 - Processing equipment at the New Waste Calcining Facility in Idaho

Many of the routine and maintenance operations have been converted to remote control. This greatly reduces decontamination time and increases the safety for operators. Figure 26 shows some of this equipment at the new calcining





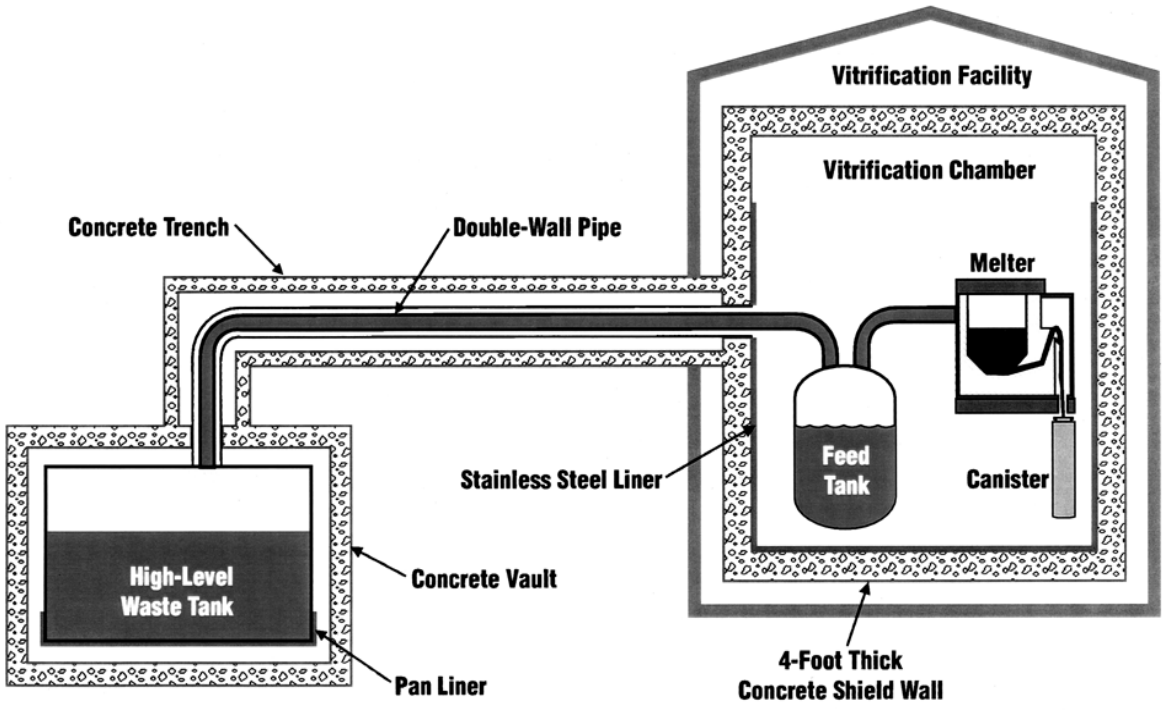
Courtesy U.S. DOE

Fig. 27 - Calcine storage tanks at the Idaho National Engineering Laboratory

facility. Figure 27 illustrates some of the above-ground calcine storage tanks at the Idaho site. It also shows a cut-away model of a typical tank.

The calcine is only an intermediate step. It does not possess the mechanical properties desired in a solid product designed to withstand environmental forces for thousands to millions of years. On the other hand, vitrified products (glasses), do have some desirable properties. They can dissipate decay heat fairly well and have a rather high resistance to leaching (dissolving away of the surface by outside groundwater). Measured leach rates for glasses run around a microgram per day over a one square cm surface area. The vitrified end product typically contains 15% to 30% waste oxides by weight. The vitrification process involves heating the calcine to between 1,000 and 1,200° C. in the presence of glass chips. The final glass solid produced is a black, opaque material with a density of between 2.5 and 6 gm/cubic cm.

As mentioned earlier in this Chapter, the West Valley Demonstration Project in New York state was the first successful large scale high level waste vitrification project to be completed. Figure 28 shows the basic layout. The high level waste tank is 30 feet high, 70 feet in diameter and made of carbon steel. It originally held 600,000 gallons. Over time, most of the fission products decay so the waste is primarily Cs-137 and Sr-90. The waste was pumped into the Vitrification Facility where it was chemically assayed, pH adjusted and then added to the molten glass in the Melter, Figure 29. The 52 ton ceramic lined Melter heats the glass and waste to 2000° F and over the course of 36 hours, fills the stainless steel canister with 2.5 tons of product. A photograph of the Melter and accessories is shown in Figure 30. The Melter is a 10 foot cube inside the hot cell. The off gas system handles the gases, particulates and moisture produced before release. Operators view the process through 4 foot thick shield windows and 19 video cameras. After filling, the canisters have a top welded on, are



Courtesy, West Valley Demonstration Project

Fig. 28 - West Valley Demonstration Project basic flow diagram

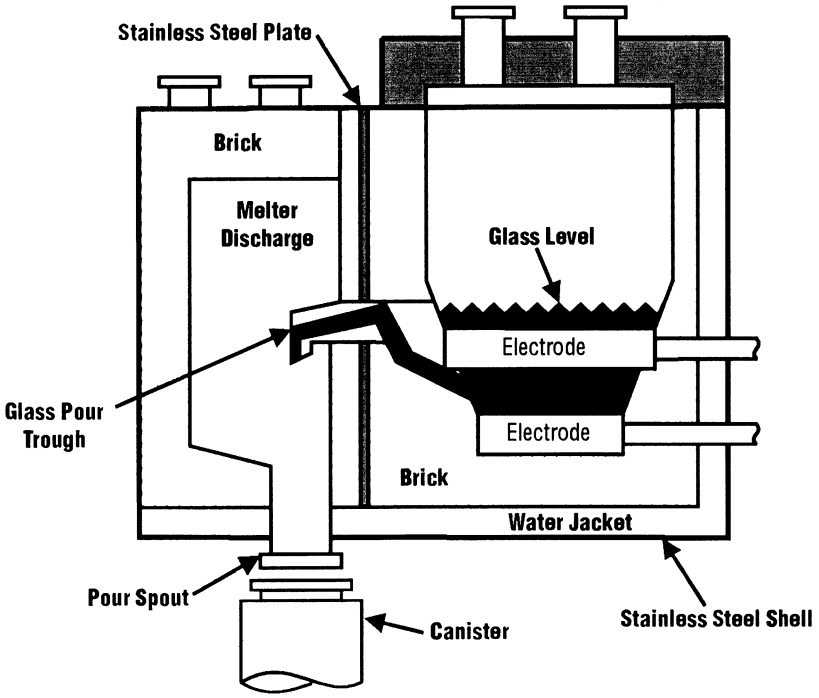


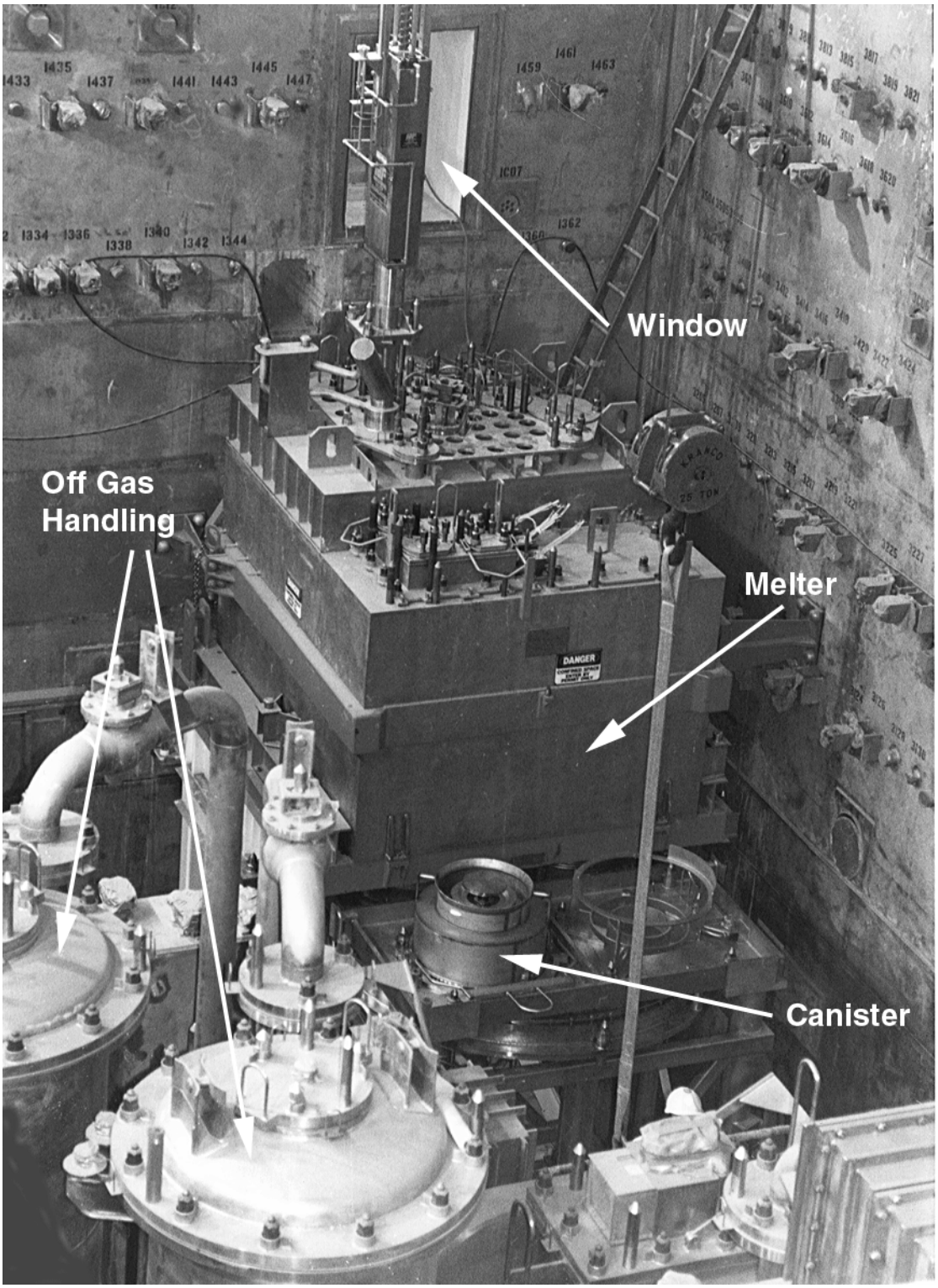
Fig. 29 - The 52 ton West Valley Melter

inspected (remotely due to a 7500 rem per hour surface dose rate) and placed in temporary storage in a hot cell at the West Valley Facility, Figure 31.

Construction commenced in April 1992 at the Hanford Waste Vitrification Plant near Richland, Washington. Plans called for construction to be completed in 1998 and for the glass melter to start-up in 1999. Unfortunately, the project was cancelled following concerns regarding storage of the low level waste stream product in grouted vaults at the Hanford Site. The building has been converted to store spent unprocessed fuel from the Hanford reactors.

As of 2000, BNFL has taken over vitrification projects at Hanford with funding coming mostly from the private sector. Upon successful performance, the DOE will pay BNFL for completed vitrified product. Plans call for vitrifying 10 to 15% of the radwaste stored in 177 underground tanks at Hanford. A 1/3 scale pilot test melter was successfully operated by GTS Duratek, a BNFL partner on the Hanford River Protection Project, in 1999. The pilot melter produced 5 tons of product per day with an average melter availability of over 97%.

The Defense Waste Processing Facility, DWPF, at the DOE Savannah River Site was actually the first (8 months ahead of West Valley) to produce high level vitrified waste in the U.S. This $2.4 billion plant began operations in April, 1998 and is the world's largest vitrification facility. It is operated by the Department of Energy. It is planned that 34 million gallons of high level waste containing Sr-90, Cs-137, plutonium and uranium and presently stored in 51 tanks, will be vitrified into some 6,000 stainless steel canisters 10 feet high by 2 foot in diameter over a 25 year period. In the DWPF project, about 94 pounds of radionuclides are mixed with 3700 pounds of glass



Courtesy, West Valley Demonstration Project

Fig. 30 - Inside the vitrification chamber at West Valley

Courtesy, West Valley Demonstration Project

Fig. 31 - Stored vitrified canisters at West Valley

in each canister. The decay heat output averages about 400 watts per canister in the Savannah River project. As of January 2011, 3059 canisters had been filled with 12,000,000 pounds of glass containing over 34 million curies of high level waste.

Recently, discussion has centered on the question of what is the best final end product. Many investigators believe that the vitrified form just discussed can be improved upon. One of the leading suggestions is to form a true micro-crystalline material, i.e., a ceramic. This form is expected to show greater resistance to fracturing into small pieces. Other suggestions involve placing chunks of ceramic in a container which is then filled with a metal alloy to increase the overall thermal conductivity. A final decision on the ultimate form is probably many years away.

Mixed Waste

There is one other category that should be mentioned. Federal regulations define "Mixed Low-Level Radioactive and Hazardous Waste" as "waste that satisfies the definition of LLW in the Low-Level Radioactive Waste Policy Amendments Act of 1985 and contains hazardous waste that either (1) is listed as a hazardous waste in Subpart D of 40 CFR 261 or (2) causes the LLW to exhibit any of the hazardous waste characteristics of Subpart C of 40 CFR 261." In the past, such mixed waste has been controversial since it comes under regulatory authority of both the NRC and the EPA. The regulations of the two agencies are contradictory and incompatible. Hazardous waste processors refuse it because it is radioactive and the LLW burial sites cannot accept it because it is "hazardous" under EPA Resource Conservation and Recovery

Act, RCRA, rules. Common examples include organic liquid scintillation fluids, slightly radioactive cleaning solvents and discarded lead shielding containers with radioactive contamination.

The NRC and the EPA cooperated in the development of regulations to ease the bottleneck created by mixed wastes. Nuclear Sources and Services in Texas was the first company to successfully negotiate the regulatory minefield and receive, in 1989, the licenses needed to handle mixed waste. Similar licenses were issued in 1990 and 1991 to Envirocare of Utah, Inc., for a facility about 75 miles west of Salt Lake City. Then in 2002, the EPA issued a final rule granting a conditional exemption from RCRA of low level rad waste during storage, treatment, manifesting, transporting and disposal.

The trenches at Envirocare (now operated by Energy Solutions) are filled and compacted in 12 inch slices using a continuous "cut and cover" process. Upon completion, the trench is covered with a 7 foot thick clay radon barrier, an 80 mil high density polyethylene seal and 2 feet of rock. The liner beneath the trenches uses multiple drainage nets to collect leachate. To emplace mixed waste, a client must furnish a mixed waste profile form, a physical properties form, a radiological evaluation form, a certified laboratory analysis and then deliver 60 pounds of representative samples to the site.

Biological Methods of Remediation

A recent development applicable to low level waste management is an array of biological techniques for contaminated soils and tank sludges. As will be discussed in the next section, radioactively contaminated soils traditionally are dug up and hauled away to an approved burial site. This can be expensive if large volumes are involved and disruptive of the contaminated site. Also, it may not be the best solution in terms of "using up" burial site capacity. Two classes of biological methods show particular promise. These are phytoremediation and bioremediation. The radionuclides that have been identified as possible candidates include Cs-137, Sr-90, Ra-226, Tc-99, plutonium, uranium and thorium.

Phytoremediation is the use of plants to remove and contain radioactive material. Through their root systems, they can efficiently extract certain radiocontaminants by natural processes. Specific plants are chosen dependent on the radionuclide and its chemical form. They are grown and then harvested and incinerated to concentrate the radioactive material that was removed from the contaminated soil. A closely related technique, called rhizofilteration, grows plants off-site with the roots submerged in radioactively contaminated water transferred from the site. Once the plants are "trained" to the contaminated water, they are taken to the site and planted in the soil. The roots continue to extract contaminated water from the ground and the plants can later be incinerated to again concentrate the radiocontaminant in a small volume of waste.

The DOE has sponsored research into the DNA mapping of Geobacter metallireducens. This bacterium, through redox reactions, can alter the valence state of uranium allowing easier removal. Genes have been identified that adapt the organism to changing soil conditions, allowing it

to grow a whip-like tail that enables it to move to a higher uranium concentration area to continue feeding.

Bioremediation is the term applied to using living microorganisms to trap certain radiocontaminants or to change them chemically so that they are less of a hazard. Several techniques are being investigated. Microbes can be selected which particularly accumulate certain radiocontaminants and then they are added to the soil at the site. Fertilizers and/or increasing the oxygen supply stimulate the microorganisms to even greater activity. Some microbes can chelate specific radionuclides, particularly metals, which will bind them up so that they cannot migrate as readily into groundwater. Other microbe processes can convert organic contaminants into less toxic inorganic compounds.

Biological methods of remediation are most suited to large land areas with low concentrations of rad waste. One downside is that they need longer times to act than the traditional "dig it out" approach. On the upside, the methods can be cheaper and have less of an impact on the site. To be most efficient and cost-effective, a site must be carefully evaluated before recommending biological methods. The soil must have nutrients available to sustain the plants or microbes or be receptive to the addition of fertilizers, etc. Specific plants or microbes have to be matched up to the known radiocontaminants in terms of radionuclides and chemical form. More information on this topic is listed in the Reading List.

Long-Term Storage Methods

Low Level Waste Burial

The only technique developed to date for LLW involves ground burial. The slit trench approach has been popular in this country. At an approved burial site, a trench is cut in the ground, typically 6 meters wide by 6 meters deep by 150 meters long. Then, 55 gallon shipping barrels or radwaste boxes are stacked into the trench in layers, up to within about a meter of ground level. The trench is next back-filled and additional earth is formed into a mound above the trench to aid in rainwater runoff. Figure 32 shows some details for this method. Figures 33 through 35 are aerial photos of the Richland, Washington and Beatty, Nevada sites along with a close-up view of a trench. Although the Beatty site was closed in 1993, it was still owned and licensed by the operating company the same as all the other closed sites. Thus, history was made on December 30, 1997 when the site was officially transferred to the State of Nevada for long-term custodial care. It was the first time any commercial operator in the U.S. was able to meet all the regulatory paper work, set out in 10 CFR 61, satisfy site closure requirements and return a site to state government control. A seven foot soil cap was placed over the entire 26 acre site. Money for the long-term care was collected as a surcharge on site users.

In the case of high gamma ray exposure rates from the steel drums, the trenches are cut narrower to allow the earth sides to more effectively shield operating personnel. Remote connecting grapples on cranes are used to off-load in such cases. A frequent practice is to dig the trenches with a slope toward one side. This tends to

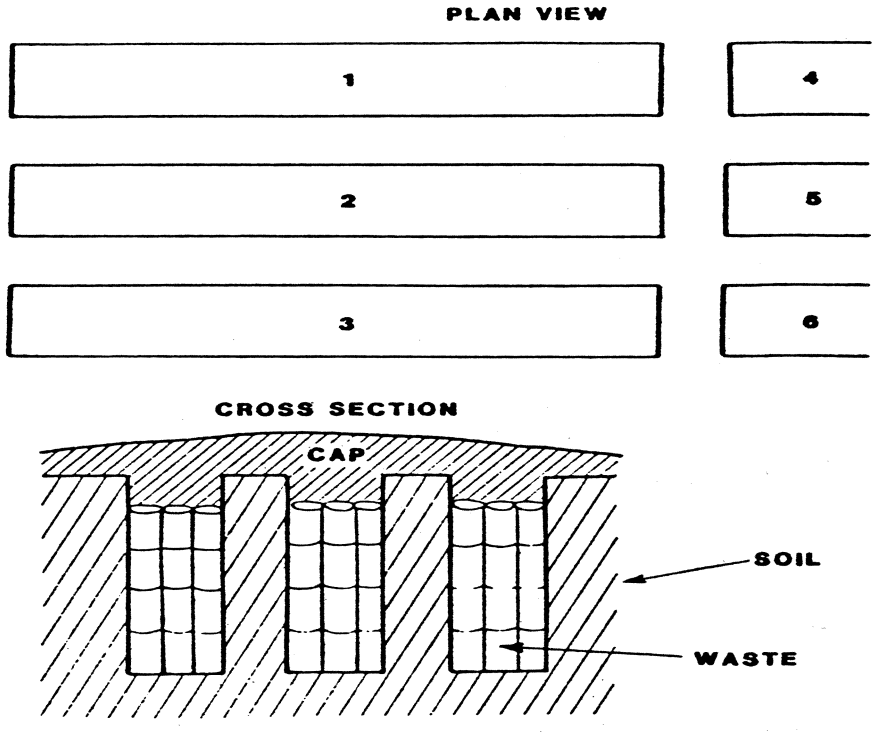


Fig. 32 - Details of shallow land burial trench

Courtesy, U.S. Ecology, Inc.

Fig. 33 - Richland, WA low level disposal facility

Fig. 34 - Beatty, NV low level disposal facility

Courtesy, U.S. Ecology, Inc.

Fig. 35 - Disposal trench at Beatty, NV

Courtesy, U.S. Ecology, Inc.

JOB DESCRIPTION	# PERSONNEL	AVE DOSE/QUARTER (rem)
Health Physics	10	0.331
Offloaders	26	0.452
Truck Drivers	8	0.028
Technical Services	5	0.013
Equipment Operators	6	0.521
Maintenance	12	0.000
Administrative	40	0.035
Contract Personnel	5	0.000

(Note: Activity handled for this quarter = 98,905 Ci)

Fig. 36 - Quarterly radiation doses at a LLW burial site

divert rainwater toward the edge of the trench where water sampling pipes and a trench drain had been installed before the trench went into use. The sampling pipes will provide some information on leakage rates in the future. Also, cement monuments (tombstones) must be affixed at each end of a filled trench at a shallow-land burial ground listing the curie contents and date of burial. Since the effective half-life of the mixture of buried radionuclides is unknown, the curie content is completely useless. However, the U.S. Congress mandated this placement.

The designed-in ability of a trench to contain the buried waste depends on the class of waste put into the trench. The Class A trench must provide containment for 100 years. The higher level Class B waste must be contained for 200 to 300 years and Class C trenches are designed for 500 year integrity. To give an idea of the amounts of personnel radiation exposure at an operating waste burial site, data is provided in Figure 36 for exposures at a commercial site in 1979. The average exposure for the burial site workers was177 mrem for the calendar quarter.

Before concluding with a discussion of high level storage techniques, it may be appropriate to mention the past practice of ocean dumping of drums of radwaste. Four different ocean sites have been used by the U.S. A summary of former ocean disposal practice is given in Figure 37.

High Level Waste Storage and Disposal

A discussion of long-term management for high level radwaste needs to be preceded by some comments on the types of radionuclides which might fall in this classification. As a general rule, transuranic wastes (wastes containing elements with a Z >92, the Z of uranium) have many extremely long-lived nuclides and often decay by alpha particle emission. The long half-life means they will maintain their radioactivity over a time span drawn out enough that movement from a geologic repository into the biosphere is possible. Once they reach the biosphere, their quality factor of 20 means they are unusually hazardous to life forms. On the other hand, fission products and reactor activation products have relatively short half-lives and are usually beta-gamma emitters with a quality factor of 1. As a consequence, storage times needed for

Site I.D.	Depth (m)	Distance from land (km)	# 55 gal. Drums Dumped
Atlantic "A" Site "A"	2800	190	14,300
Atlantic Site "B"	3800	320	14,500
Pacific Site "A"	900	60	3500
Pacific Site "B"	1700	77	44,000

Fig. 37 - Former U.S. ocean disposal sites

transuranic waste to reach background levels is of the order of 10 million years while fission and activation product waste requires a storage time of the order of 1,000 years. Much more stringent engineering safeguards are thus needed for the transuranic waste. Unless this waste form is separated from other waste, the entire volume will have to be treated as containing transuranics. Thus, it becomes important to segregate waste types at the point of generation as an overall management principle. This point was discussed in conjunction with the 10 CFR Part 61 federal regulations mentioned previously.

A variety of different techniques have been seriously proposed over the years for dealing with high level radioactive wastes over long time periods. Some of these are considered retrievable, i.e., the material could, in principle, be located and removed in the future. Other proposals involve non-recoverable methods. A short but representative list of some long-term methods is provided in Figure 38.

The polar ice sheet is that mass of ice which permanently covers the South Pole. Since the steel drums of high level radwaste would be thermally "hot," the proposal was to set them on top of the ice pack and leave. Presumably, they would melt their way down until they contacted the underlying bedrock of Antarctica. This proposal is not considered feasible for several reasons. International treaties prohibit dumping nuclear waste in the region. From the technical point of view, too little information is available to assess the proposal. The rates of movement and overall stability of the ice sheet over million year time spans is not known. Finally, conditions at the ice/rock interface have not been studied. The drums might be scraped along and ground up and dispersed into the ice.

Various methods involving the seabed have been suggested. One popular proposal is to core a hole in the seabed and place the waste in the hole. A more clever suggestion is to implant the waste into a river delta where a build-up of silt will cover it. Another recommendation is to place

NONRETRIEVABLE

Polar Ice Sheet	Sea Bed	Volcano
Deep Space	Transmutation	

RETRIEVABLE

Deep Geologic Hole	Salt Deposit	Nuclear Test Hole
Hard-rock Mined Tunnel		

Fig. 38 - Some suggested long-term radwaste storage ideas

the waste at an undersea junction between moving crustal plates. Then, over geologic time spans, the one plate will slide under the other which carries the waste further away from the biosphere. All of these are considered technically unfeasible at present. There are too many gaps in our knowledge of long-term effects in terms of the geology, marine biology and oceanography. We cannot guarantee that the material will not become waterborne.

The volcano proposition is interesting. An active volcano is located and the high level waste is to be transported to the site. It is then placed directly in the path of a flowing lava stream. The lava is supposed to flow over the waste, harden and seal it up forever. (The persons advancing this scheme failed to give details on how the material would actually be placed and by whom. Any volunteers??) The chief objection to this method is the fact that the radioactive waste is placed in an obviously volcanic region. This makes the chance of a future eruption directly underneath the deposited waste too high for comfort.

As with the seabed, several schemes have been advocated which involve locations off the globe as the final repository. Merely lifting the waste into an orbit around the earth is not satisfactory. The stability of orbits cannot be guaranteed for the necessary time spans (as both NASA and the Russians have found out from time to time). One truly final space site would be to shoot the waste into the sun. Unfortunately, the cost of solar impact vs. alternative scenarios is unattractive (see Figure 39). With the current availability of a working space shuttle, launching a waste package into a solar orbit is considered technically feasible and might be economically viable for high level transuranic wastes. The major difficulty is the possibility of a launch failure. The subsequent dispersal of the material would pose major radiological hazards.

The final nonretrievable method listed involves transmutation – changing one element into another. It has been suggested that the transuranic wastes be put back into a nuclear reactor where the high neutron flux would induce a large fraction of the nuclei to undergo fission. This process would then convert the long-lived alpha emitters into relatively

DESTINATION	$PER POUND	$PER CUBIC METER
High Earth Orbit	628	2,200,000
Solar Orbit		
Single-burn Earth Escape	628	2,200,000
Via Mars or Venus	794	2,800,000
Dble-burn Circular Orbit	800	2,800,000
Solar System Escape		
Via Jupiter	3,500	12,300,000
Direct	4,420	15,500,000
Solar Impact		
Via Jupiter	4,700	16,400,000

Fig. 39 - Estimated space shuttle disposal costs in 1972 dollars

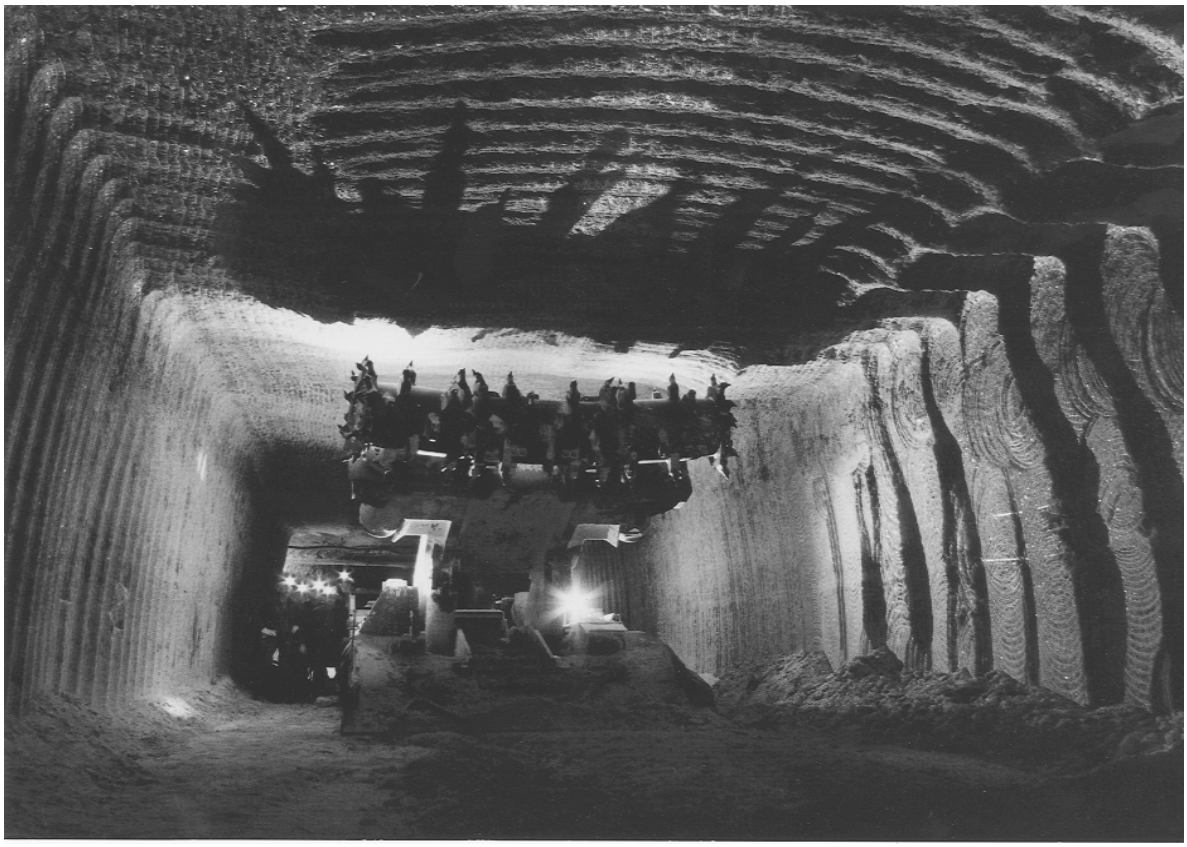
short-lived fission products. This technique is considered technically feasible and is being given further study.

Figure 38 suggests four possibilities for retrievable storage. The deep hole method involves boring a hole several miles deep into the ground and then dumping in the waste. This idea has been criticized as requiring the development of a whole new technology since existing drilling technology cannot go that deep. Furthermore, the reason for going an extra mile or two is not clear. If a future geologic disruption is going to take place at that location, a few miles deeper probably would not prevent dispersal into the biosphere.

Use of a test hole following an underground nuclear detonation has some interesting possibilities. This idea is currently receiving consideration at the Nevada Test Site. Although many of the test holes collapse shortly after the detonation, some do remain intact. These glassified walls appear to be an attractive container for long-term isolation from the environment. Hopefully, more definitive information will be available in the near future on the suitability of this technique.

Salt deposits and granitic rock have both received a fair amount of study as final repositories. The salt layer is actually a plastic material – fissures and cracks produced tend to close up over time as the salt mass flows. Salt also has good thermal conductivity which is necessary to dissipate decay heat. Finally, it is easy to mine. The disadvantages include the possibility of brine inclusions. Future groundwater changes could rapidly dissolve away the deposit. The atmosphere is corrosive to containers. Finally, the salt deposit might be considered useful to generations in the distant future who would then mine the area. The WIPP facility discussed below is the first of this type of repository to open in the USA.

Granitic deposits also have some advantages and disadvantages. The material is much more difficult to excavate. The granite contains little material with high ion exchange properties so it does not act as a natural trap. However, some deposits are relatively stable on a geologic time scale. The decay heat released does not appear to introduce major stress



Courtesy, U.S. Department of Energy

Fig. 40 - Salt deposit excavator in use at the WIPP facility in New Mexico

in the rock structure. The Yucca Mountain Project discussed later in this Chapter is the first U.S. entry into granite repositories.

The long awaited opening of the Waste Isolation Pilot Project, WIPP, finally occurred March 1999. This research and development facility of the U.S. Department of Energy is located 26 miles east of Carlsbad, New Mexico. It is the nation's first permanent geologic repository for defense generated radwaste. It consists of mined cavities in a salt deposit 2150 feet below ground level and is capable of disposing of transuranic wastes generated in weapons production. The underground tunnels are mined with carbide steel excavators (Figure 40). Some 16 kilometers of tunnels have been prepared at a depth of 655 meters below the surface. The overall layout of the site is shown schematically in Figure 41. The capacity of the WIPP site is expected to be 6.2 million cubic feet which will take about 35 years to fill. During 2005, the milestone of 1 million cubic feet had been reached. As of February, 2011, a total volume of 2.6 million cubic feet have been emplaced.

Transuranic waste is brought to the area by tractor-trailer rigs carrying 3 TRUPACT-II vessels that hold 6 standard waste boxes each of which holds the equivalent of six 55-gallon drums (Figure 42). (TRUPACT = Transuranic Package Transporter.) The drivers are in constant touch with the WIPP dispatcher by radio and the truck location is constantly monitored by a satellite tracking system.

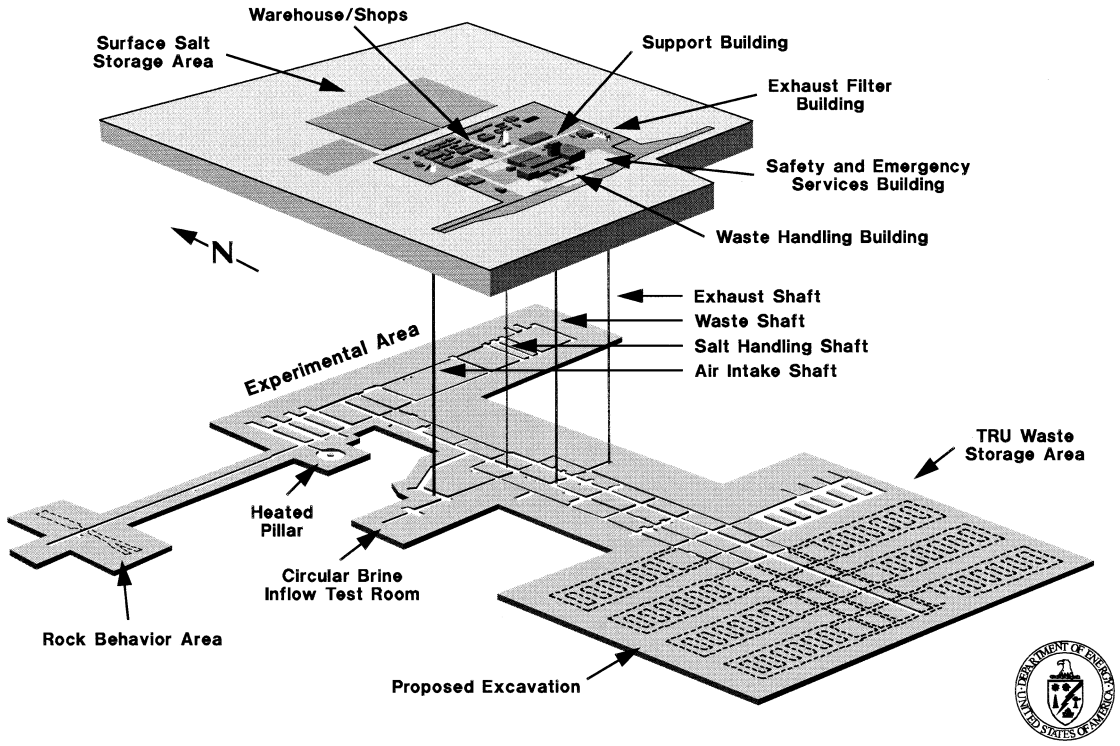


Fig. 41 - Overall layout of the WIPP transuranic disposal facility in New Mexico

Fig. 42 - Transport of TRUPACT-II transuranic waste containers

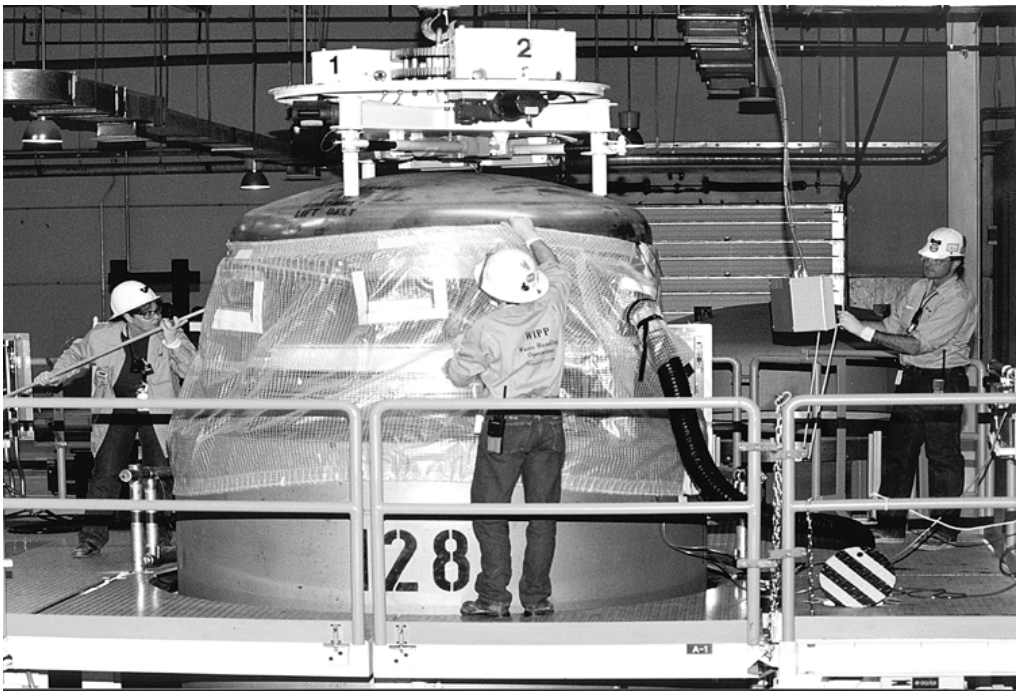
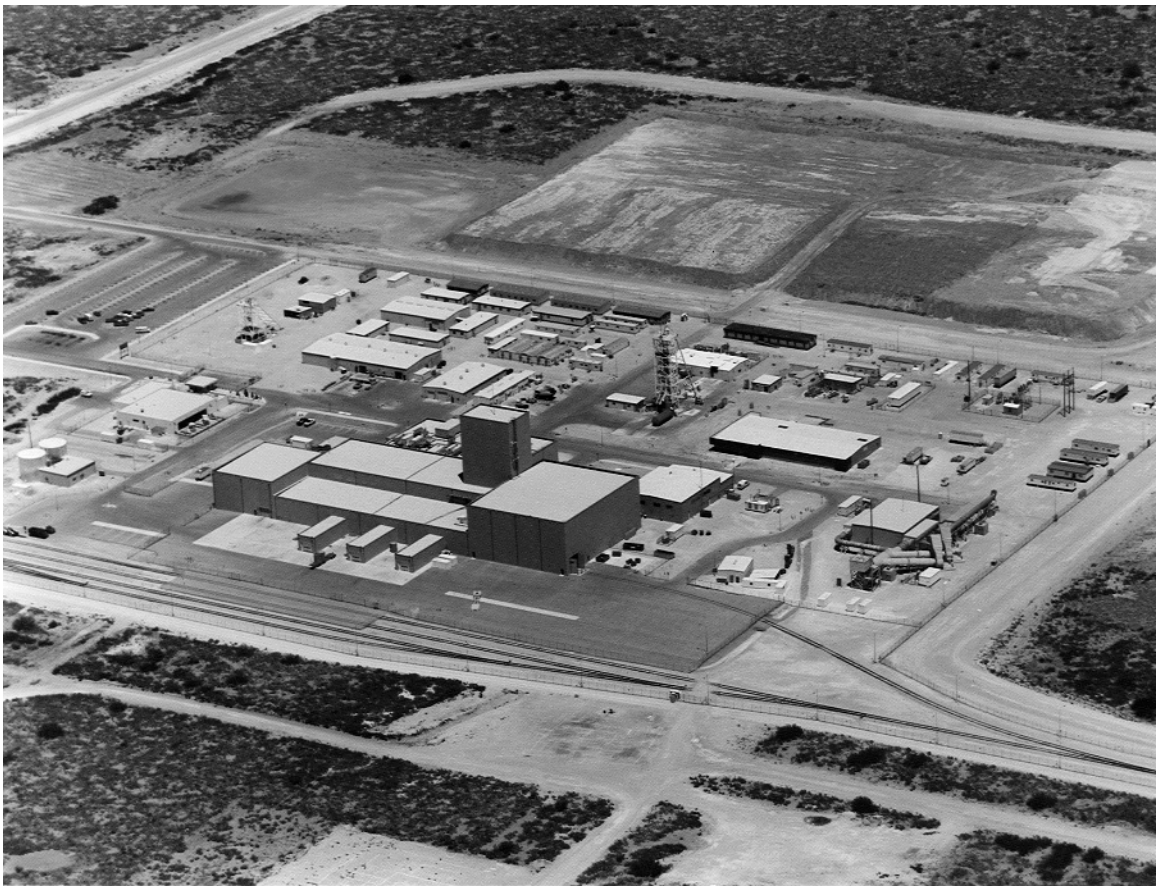


Fig. 43 - Wipe testing a TRUPACT container at WIPP

Courtesy, U.S. Department of Energy, WIPP Site

Fig. 44 - Final emplacement of transuranic waste in the WIPP salt deposit

Courtesy, U.S. Department of Energy, WIPP Site



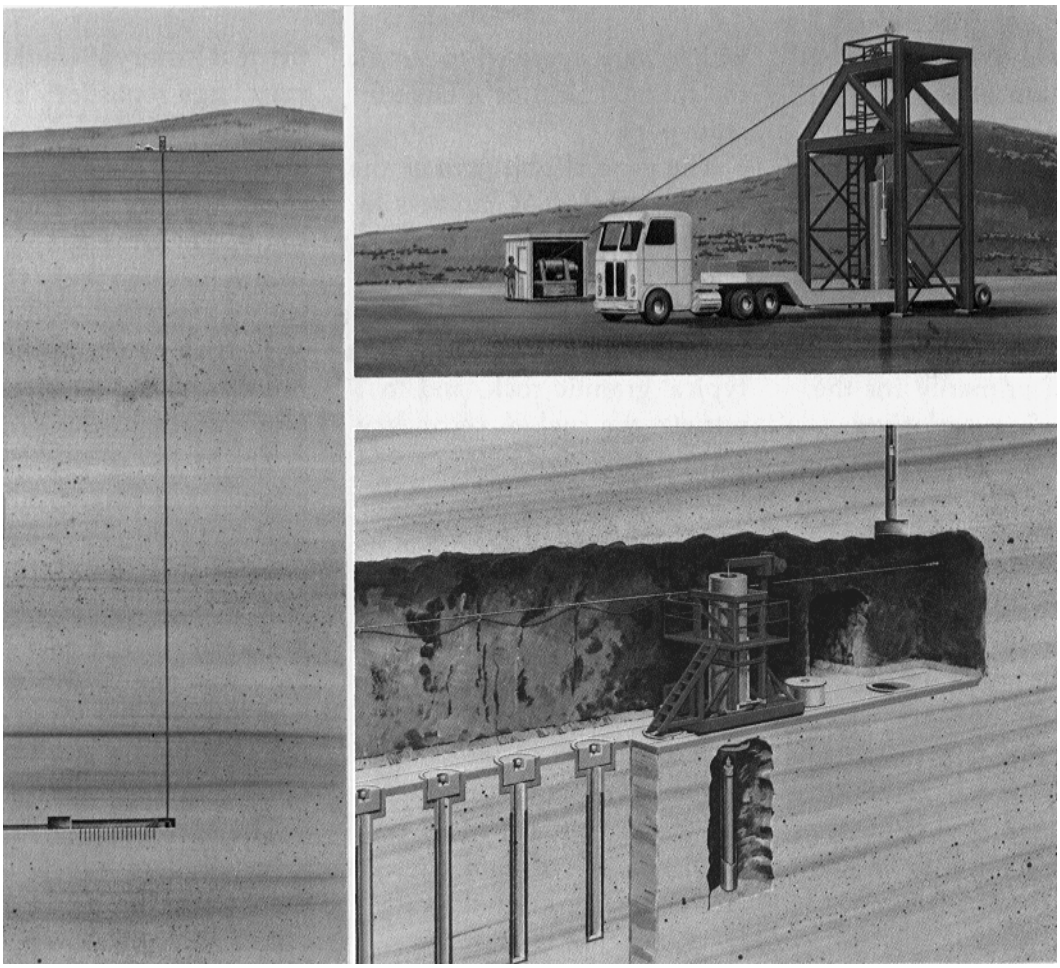
Courtesy, U.S. Department of Energy

Fig. 45 - Aerial view of the WIPP facility near Carlsbad, NM

After arrival, the TRUPACT-II is unloaded into the surface Waste Handling Building. After passing numerous radiological tests, (Figure 43), the waste boxes are removed and carried underground to their final resting place, (Figure 44).

The physical facilities were essentially complete (Figure 45) as of 1993. Political problems then delayed the opening until 1999. Eventually, both the DOE and the Environmental Protection Agency certified that the WIPP site met all federal regulations for transuranic waste disposal, and litigation brought by public interest groups was finally settled.

Several experiments have been conducted to study the feasibility of using mined cavities in granite as an interim storage location for spent fuel assemblies from power reactors. One such project was completed in the Climax Mine on the Nevada Test Site. Eleven canisters of spent fuel from a commercial nuclear station were emplaced during the spring of 1980. Adjacent cavities were loaded with electrical heaters to simulate the thermal characteristics of a large storage array. Temperatures and stresses produced in the granite were monitored over the three and one half year study at more than 750 locations. Figure 46 is an artist's sketch of the spent fuel test setup. Figure 47 shows the vehicle designed and used for underground remote transport and handling of spent fuel canisters.



Courtesy, Lawrence Livermore National Lab & DOE

Fig. 46 - Spent fuel storage test in Nevada

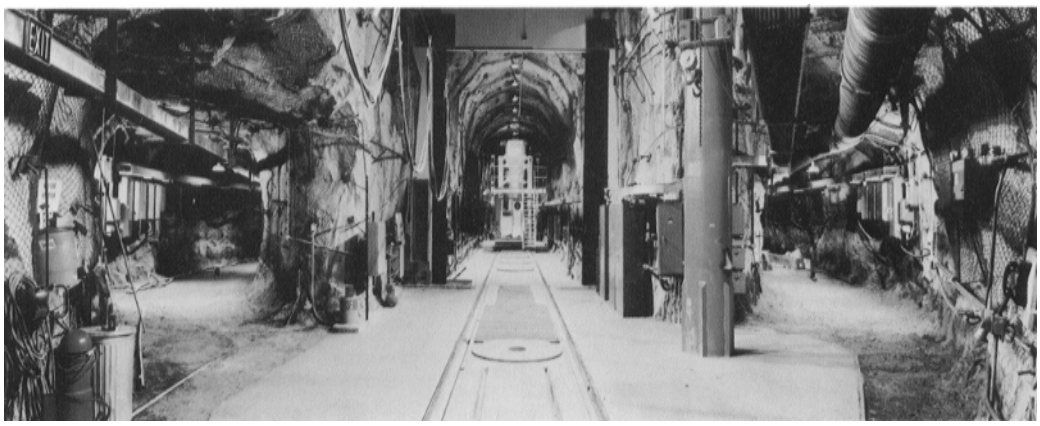


Fig. 47 - Underground storage and side drifts. Transporter in center drift

Courtesy, Lawrence Livermore National Lab & DOE

At the conclusion of the test, the spent fuel canisters were removed from the Climax Mine. Test results indicated that the maximum wall temperature reached by a canister emplaced in the granite was 140° C. The height of the drift (measured floor to ceiling) showed a maximum change of 3 mm due to the thermal load produced by the fuel. The average dose to personnel to handle one fuel bundle turned out to be 2 millirem (20 μSv). Finally, the researchers reported that the heating of the granite caused the radon concentration in the storage area to increase by about a factor of 10.

A legitimate question which arises in connection with long-term disposal proposals relates to the time factor. How can we know that a repository can contain radioactivity for a time span longer than recorded human history? A number of engineering arguments and calculations can be presented, but perhaps the best answer is shown by experience. A case exists where reactor fission products and transuranic waste have been successfully isolated from the environment in a geologic deposit for 1,800 million years, some 200 times longer than proposed by the most conservative environmentalists!

This amazing feat took place in the Republic of Gabon in western Africa. A high concentration of uranium, naturally enriched to 3.2%, collected in a river delta and formed a zone about 1/2 mile wide by 15 - 30 feet thick. The addition of river water moderated the zone sufficiently so that k_{eff} exceeded 1. This natural "Oklo Reactor," discovered in 1972, operated for about 150,000 years, at an average power level of around 100 kilowatts, before exhausting the ^{235}U. Measurements of isotopic ratios conducted in 2004 revealed that when active, the reactor cycled through 30 minutes of "ON" time which created enough heat to boil off the moderating river water followed by 2 1/2 hours "OFF" time that allowed the steam to condense enough to moderate the neutrons sufficiently to re-initiate a new cycle. During operation, it produced a total of 12,000 pounds of fission products and produced 4,000 pounds of plutonium through neutron capture reactions. Present-day tests demonstrate that the noble gases in the fission products, krypton and xenon, escaped along with some water-soluble products. However, the bulk of the fission products and virtually all the thorium, uranium, neptunium and plutonium have stayed in place, even though the zone was repeatedly flooded with river water.

The U.S. Congress passed the "Nuclear Waste Policy Act of 1982" in December of 1982. This law authorized the Department of Energy to design and construct two repositories in geologic formations. The first site was limited to a capacity of 77,000 tons of radioactive waste. The President was required to report on the location of the first site by 1987 and the second by 1990. These sites were to be used primarily for encapsulated spent fuel assemblies from commercial nuclear power stations. In 1984, the Department of Energy selected three candidate sites for the first high level repository – Hanford, Washington (3,000 foot deep basalt rock deposit), Deaf Smith County, Texas (2,500 foot deep salt deposit) and Yucca Mountain, Nevada (1,500 foot deep granite deposit). In May, 1986, the Secretary of Energy announced the postponement of development of the second repository. The volume of spent nuclear fuel at commercial plants was growing more slowly than originally estimated, and plans for the first site were on schedule for a 1998 opening. Then, in December 1987 Congress voted to

Fig. 48 - Yucca Mountain, Nevada

Courtesy, DOE and EG&G

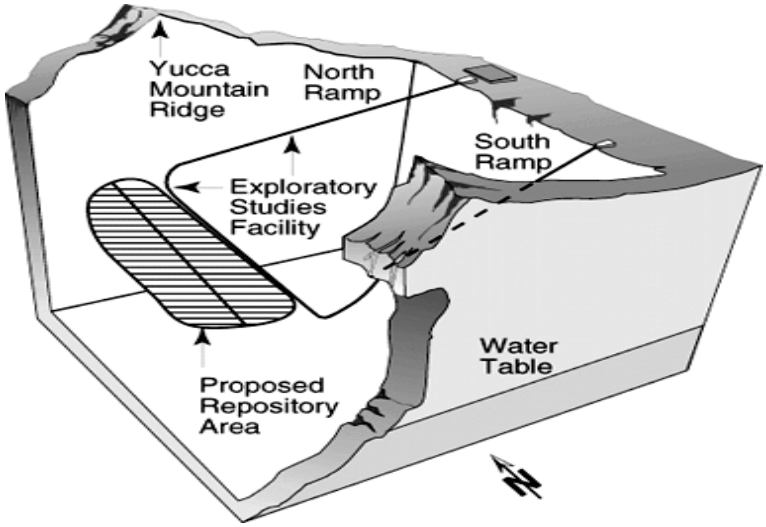


Fig. 49 - Relative location of the proposed Yucca Mountain HLW repository

Courtesy, U.S. Department of Energy

halt all studies on the Texas and Washington state candidate sites and proceed only with the Nevada site. This decision saved about $4 billion in testing costs at the two abandoned locations.

Yucca Mountain is located about 100 miles northwest of Las Vegas, NV. It lies on the border between a bombing range on Nellis Air Force Base and the Nevada Test Site where 689 announced underground nuclear tests have been conducted. An aerial view of the site (with a superimposed construction map) is given in Figure 48. Figure 49 shows the relative location of the repository with respect to the groundwater table depth and the top of the mountain.

The high level waste would be in the form of spent fuel assemblies encapsulated in canisters and placed into holes drilled in the floor of tunnels, called drifts, that would run for 116 miles throughout the site. Geologically, Yucca Mountain is formed from an igneous rock called tuff. Tuff contains the mineral zeolite which traps radioactive particles while allowing diffusion of water molecules. The mountain is riddled with earthquake faults which are of great concern to some scientists. Possible volcanic activity at the repository and the threat of rainwater seeping in and corroding canisters to the point where radioactivity could reach groundwater are other voiced concerns.

By 2000, site characterization was well underway. This project cost $6 billion and addressed the topics shown in Figure 50. The results were transmitted to the U.S. Nuclear Regulatory Commission for review. In February 2002 President Bush recommended, to the U.S. Congress, the Yucca Mountain location for a permanent spent fuel repository. Congress concurred in July 2002, appropriated $580 million and ordered the U.S. DOE to prepare a license application.

Then, the courts stepped in again. Under the Nuclear Waste Policy Act of 1982, the EPA is responsible for writing the environmental safety standard for Yucca Mountain. In September of 2004, the U.S. Court of Appeals ruled that the EPA proposed 10,000 year safety standard was invalid because it didn't meet the National Academy of Sciences recommendation of 100,000 years or more. Once that issue was clarified, in 2008, the DOE submitted an 8,600 page license application, The NRC was given three years to hold public hearings, review and act.

Politics again came to the forefront. The incoming Obama administration declared that Yucca Mountain would not be used for radioactive waste storage! The Department of Energy withdrew the Yucca Mountain license application in March, 2010. Thirty-nine years and billions of dollars later, the United State's geologic depository for spent fuel storage was shut down.

Geology	**Geoengineering**
Hydrology	**Geochemistry**
Climate	**Repository Design**
Waste Package	

Fig. 50 - Topics addressed by the Yucca Mountain Site Characterization Plan

Courtesy U.S. Department of Energy

Fig. 51 - A view inside the Exploratory Studies Facility tunnel

One of the chief sources of data for the characterization study was the Exploratory Studies Facility (ESF). Figures 51 and 52 show a view inside the ESF tunnel and a sketch of the 13.6 miles (22 km) of tunnels and drifts that constitute the ESF. The "TS North Ramp" runs from the surface to the Topopah Spring level. It will be 25 feet in diameter and 1¼ miles long. The North Ramp to the "Calico Hills level" (which lies about 170 meters below the Topopah Spring level) will be 16 feet in diameter and 3/4 of a mile long. The main 25 foot diameter tunnels will be serviced by a

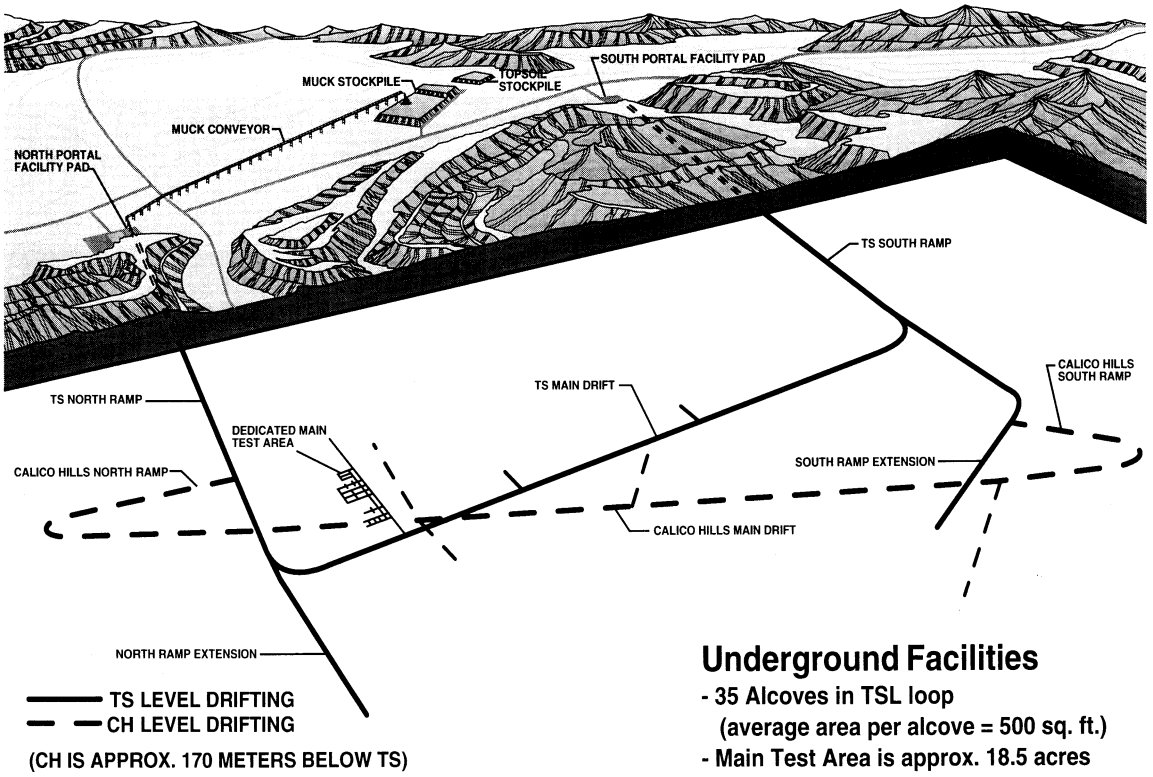


Fig. 52 - The design of the Exploratory Studies Facility at Yucca Mountain

two-way rail transportation system running on four rails attached to the twelve-foot wide tunnel floor. The location for the TS North Ramp was carefully chosen to pass through several geologic faults and other areas of geologic interest.

In April 1993, drilling and blasting began for the ESF "Starter Tunnel." Occasional air pockets were encountered which had been left by the volcanic gases that had been trapped in the cooling ash. In May 1993, a $13 million order was placed for the first tunnel boring machine (TBM). It arrived in 50 truck shipments during May 1994 and was reassembled on site. The TBM weighed in at 720 tons, was 220 feet long and was 25 feet in diameter. Its electrical power plant was rated at 3,800 horsepower. After assembly, it was moved by rail into the Starter Tunnel. It advanced by gripping the sides of the tunnel and extending its cutters into the rock face. While operating, workers can move around inside the TBM from compartment to compartment on different levels. The technology for tunnel boring is well established. The Yucca Mountain TBM is very similar to the units that were excavated under the English Channel and beneath Los Angeles for the new Metro Project. Figure 53 shows an example of a TBM similar to the model ordered for the project. Figure 54 shows the TBM emerging from the face of Yucca Mountain, on April 25, 1997, after completing five miles of ramps and the main drift loop of the ESF. It took three years to bore the 5 mile tunnel.

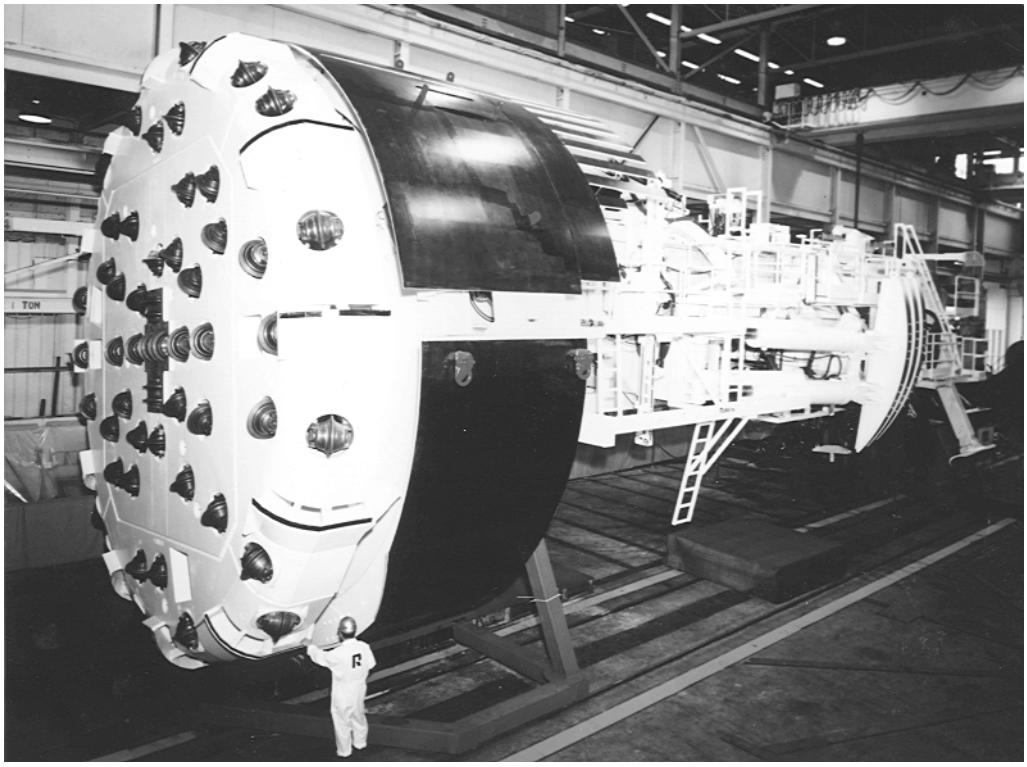


Fig. 53 - A tunnel boring machine similar to the one used in Yucca Mountain

Courtesy, U.S. Department of Energy

Fig. 54 - The Tunnel Boring Machine emerging from Yucca Mountain

U.S. Department of Energy

Dry Cask Storage of Spent Fuel

Before leaving the high level waste topic, more attention needs to be paid to commercial reactor spent fuel. As of 2004, some 37,650 tons were in storage at U.S. nuclear stations. All currently licensed plants are storing spent fuel in underwater pools. The storage capacity has been reached at some plants. In 1990, the NRC changed the law to allow on-site storage of spent fuel in above ground dry casks. As of 1999, seven different designs have been approved and 13 different sites have been licensed as "independent spent fuel storage installation sites," (ISFSI sites). See Figure 55.

A typical dry cask is16 feet high, 8 feet in diameter with 9 inch steel walls, weighs about 120 tons loaded and holds 24 spent fuel assemblies. Before loading, the spent fuel is allowed to radioactively decay for an extended period. This significantly reduces the decay heat output of the assemblies. Then, the cooled fuel is placed inside a special canister and the air and water removed. The canister is filled with inert helium gas and a lid is welded to the opening. Next, the canister undergoes rigorous leak testing. Finally, the sealed canister is placed inside the concrete cask and placed on an isolated concrete paved storage area at the ISFSI. Figure 56 illustrates the principle and Figure 57 shows loading operations at the Calvert Cliffs Nuclear Plant dry cask facility in Maryland.

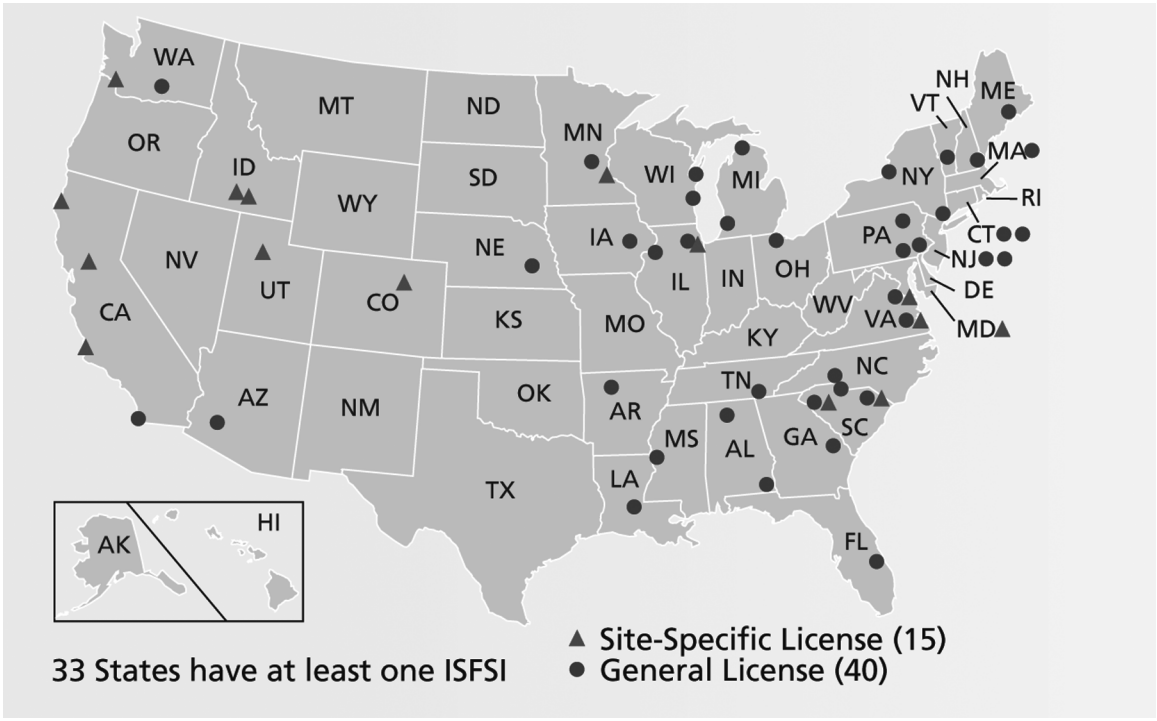


Fig. 55 - The Independent Spent Fuel Storage Installations for dry casks in the USA in 2010

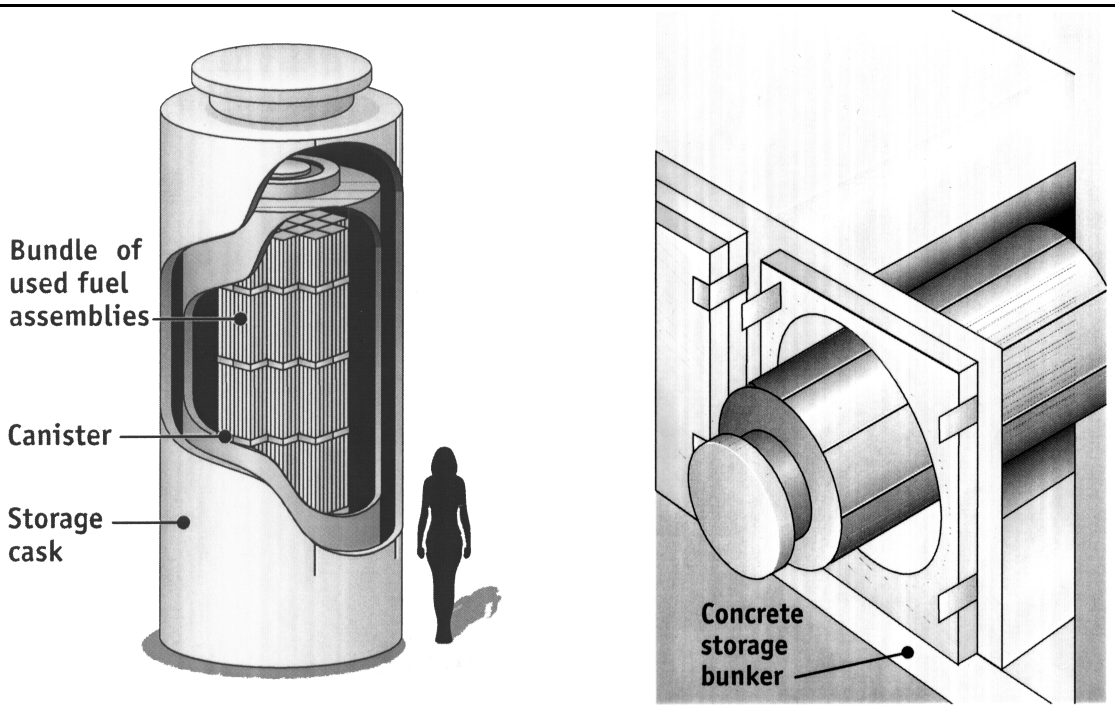


Fig. 56 - The dry storage cask concept

Courtesy Calvert Cliffs Nuclear Power Plant

Fig. 57 - Cask emplacement at the Calvert Cliffs facility

The Politics of Radioactive Waste

Nuclear Waste Compacts

As indicated earlier in this chapter, the last United States commercial low level disposal site was licensed in 1971. This situation persisted until September of 1993! The Ward Valley site in California was licensed but terminated before construction began. For the next decade or more after 1971, little thought was given to the long-term future of LLW sites. Waste generators went merrily about their business of generating radwaste without much regard for volume and the disposal sites continued to dig trenches. In the 1980s, it suddenly dawned on the involved parties that a potential problem was on the near horizon. The existing LLW sites began issuing projections showing only a few more years of capacity and the hosting states declared their opposition to enlarging any of these sites beyond the originally licensed acreage. Once the licensees realized what was happening, they began taking steps to reduce the volume of generated waste, realizing, however, this was only a temporary solution.

The breakthrough in removing the low level burial "log jam" came in December 1985 when the U.S. Congress passed the "Low-Level Radioactive Waste Amendments Act of 1985," a revision of the 1980 "Low-Level Radioactive Waste Policy Act." The original 1980 bill set up the procedures for forming a "Regional Waste Compact" – an agreement to mutually handle the waste generated within the included compact states by siting a new disposal area within the borders of the regional compact. The revision approved the first seven regional compacts involving 37 states and allowed existing burial sites to remain open to non-compact member states past the original Jan. 1, 1986 closure deadline. To obtain the extension on burial site operation, the non-compact states were penalized by imposition of numerous legislative deadlines (demonstrating their commitment to joining a compact, selecting a repository site and facilitating its opening), waste volume surcharges ($10 per ft^3 in 1986 and 1987, $20 in 1988-89 and $40 in 1990-92) to be used by the host state to develop new repositories and by penalties for noncompliance with the legislative deadlines. The bills provided a deadline of January 1, 1993 for new burial sites to be in full operation in all compacts. A map of the regional compacts approved as of 2010 is given in Figure 58.

When the January 1993 deadline for new sites was reached, the Southeast Compact, operator of the Barnwell facility, decided to continue accepting waste from outside the compact, but at a price! Large volume generators must pay "access fees" in advance, must project the volumes expected to be delivered and must pay a "premium rate" of 130% to 150% for volumes delivered in excess of the projections. The access fees were set at $1650 per 55 gallon drum and were in addition to the regular disposal costs plus the volume surcharges. In contrast to Barnwell, the Hanford LLW site has been reserved exclusively for use by the Northwest Compact and the Rocky Mountain Compact.

In the last few years, some political maneuvering has occurred among compact states and unaffiliated states. The state of New York filed a lawsuit against the federal government challenging a key provision of the low level waste acts – the "take title

Appalachian
Delaware
Maryland
Pennsylvania
West Virginia

Atlantic
Connecticut
New Jersey
South Carolina*

Central
Arkansas
Kansas
Louisiana
Oklahoma

Central Midwest
Illinois
Kentucky

Midwest
Indiana
Iowa
Minnesota
Missouri
Ohio
Wisconsin

Northwest
Alaska
Hawaii
Idaho
Montana
Oregon
Utah*
Washington*
Wyoming

Rocky Mountain
(Northwest accepts Rocky Mountain waste as agreed between compacts)
Colorado
Nevada
New Mexico

Southeast
Alabama
Florida
Georgia
Mississippi
Tennessee
Virginia

Southwestern
Arizona
California
North Dakota
South Dakota

Texas
Texas
Vermont

Unaffiliated
District of Columbia
Maine
Massachusetts
Michigan
Nebraska
New Hampshire
New York
North Carolina
Puerto Rico
Rhode Island

Note: Data as of June 2010.
*Site of an active LLW disposal facility.

Fig. 58 - A listing of regional compacts through 2010

provision." This provision required that every state that failed to have disposal site access by January 1, 1996 had to accept physical possession of the LLW generated in their state and also accept legal title to the waste. In June 1992 the U.S. Supreme Court struck down this "take title provision" as it applied to New York State but restated the validity of the remaining provisions of the acts. Thus, states were required to continue to move forward in their plans to open new sites. New York withdrew from its Regional Compact.

The state of Wyoming elected to switch compacts and has moved from the Rocky Mountain Compact to the Northwest Compact as shown in the compact list. Michigan resigned from the Midwest Compact and decided to go it alone. They subsequently agreed to refund to the compact about 6.9 million dollars that had been received as a result of their acceptance of host state designation (the host state agrees to construct a disposal facility within their boundary). Ohio was named as the new

host state. Maine has withdrawn from the Texas Compact. As of 2010, including Puerto Rico, there are a total of ten "unaffiliated states" as shown in Figure 58. As of 2000, over one billion dollars had been spent for new low level waste facilities but not a single cubic foot of capacity has actually opened. Clearly, things aren't working!

Two Compacts have been involved in lawsuits. The Central Compact sued the state of Nebraska for it's bad faith efforts to site a disposal facility. The Compact was awarded $140 million but no disposal site! The Southeast Compact has sued North Carolina for $90 million for their failure to develop a facility. This case is currently before the U.S. Supreme Court.

Proposed New LLW Disposal Facilities

For a while, there was the very real possibility of some new disposal facilities opening in the USA. The first license, in over two decades, was issued in September of 1993 to US Ecology to build and operate a new LLW facility in Ward Valley, California. The granting of the license by the California Department of Health Services was the culmination of over eleven difficult years of work. The actual license application was submitted by US Ecology in 1989. A number of outside groups then laid siege to the project in an attempt to overturn the efforts of the Southwest Compact to meet the federally mandated deadlines. Hurdles continued to appear. Litigation blocked the transfer of the land from the federal government to the State of California. In 2000, US Ecology filed a lawsuit against California to recover damages for the state abandoning their "contractual commitment to obtain the [Ward Valley] site for development..." and also wants to recover lost earnings for the projected 30 year operating life of the site - a total of $160 million. They lost the case but it is currently being appealed. In 2002, the California state assembly acted to close down all further development of the Ward Valley site.

The Ward Valley site appeared almost ideal for a shallow-land burial facility. The evapotranspiration rate in this Mojave Desert region is 130 inches per year. (See Figure 59.) This fancy word merely means that the top layers of soil have a demand for 130 inches of rainfall each year before excess water would be available to migrate down deeper, possibly into the water table. The upper layer demand for moisture is due to evaporation caused by high desert heat and to uptake into the root systems of the native vegetation. The annual rainfall in Ward Valley is only 5 inches. Thus, the evapotranspiration losses are 25 times greater than annual rainfall! The water table at the site is at 650 feet depth. These conditions are substantially superior to existing LLW sites. At Barnwell, the annual rainfall is 42 inches with a evapotranspiration rate of 30 inches. The Hanford, Washington site has an annual rainfall of 6 inches and a evapotranspiration rate of 30 inches. See Sample Problem 2.

The actual disposal facility at Ward Valley would have occupied 100 acres, although the facility was to obtain 1,000 acres of land. Five trenches were proposed. Four would have been reserved for Class A segregated waste and one would accommodate Classes B and C. The Class A trenches, 60 feet below grade to the bottom, would have received a 20 foot cap to bring them back to grade level plus an additional 5 foot "surcharge." The Class B/C trench would have only allowed for 22 feet of waste plus the 20 foot cap and 5 foot surcharge.

Fig. 59 - View of the desert sited Ward Valley LLW facility

Sample Problem 2

GIVEN:

Use the data on evapotranspiration rates given in the text.

FIND:

Which existing LLW sites have the possibility of rainwater carrying radioactivity from the trenches into groundwater?

SOLUTION:

This situation is only possible if the annual rainfall exceeds the evapotranspiration rate at the site. From the data given, the evapotranspiration rate at the Barnwell site is 12 inches less than the annual rainfall. Thus, Barnwell has the possibility of rainwater reaching the water table. At the Hanford site, the evapotranspiration rate is 24 inches greater than the annual rainfall so there is no possibility of rainfall reaching the water table there.

Although California was "leading the pack" in the development of new disposal facilities, equally hard-fought battles occurred elsewhere. Progress was being made in Texas, a member of the compact that includes Vermont. In 1992, the Texas Low Level Radioactive Waste Disposal Authority purchased the 16,000 acre Faskin Ranch located about 90 miles from El Paso. A license application had been filed on the site two months earlier. The final design was completed in May, 1993. Unfortunately, the state legislature was apparently not pleased with the idea of a new nuclear dump so they shut down the Waste Disposal Authority and transferred its functions to the Texas Commission on Environmental Quality. In 1998 they permanently denied use of the Faskin Ranch for rad waste. Then, in 2003, the legislature voted to privatize the low level waste operation. A 30 day acceptance window was opened for license applications during the summer of 2004. They received one application, from Waste Control Specialists of Pasadena, Texas. This application was judged administratively complete in February 2005. The next step was a public hearing in Andrews County, the location of the proposed facility, followed by the Technical Review. All of these actions were completed successfully although the schedule was pushed forward a couple of years. As discussed earlier in this Chapter, the final license was issued in 2009 with construction expected to be completed in 2011.

With all the political problems surrounding this issue, it has occurred to a number of experts that perhaps the answer lies in this move toward the commercial sector. Already Energy Solutions is handling Class A radwaste in Utah. The West Texas is the second private facility. Something needs to happen soon. A 1999 study by South Carolina shows that the area available for disposal at Barnwell will be filled by 2009. There were only 17 acres left! In recognition of this, after July 2008, only the 6 states in the Southeast Compact will have access.

Another couple of ideas have surfaced recently. Some of the Department of Energy LLW sites have excess capacity. Perhaps regulations could be amended so that commercial licensees could also have access to these federal facilities. The second thought is to have the federal government develop one or two LLW sites on federal land and make them accessible to the 36 states which will soon have no disposal capability. This might get around some of the politics at the state level which have prevented new site openings.

Retirement of Past Disposal Facilities

Old radioactive disposal facilities never die – and they don't fade away either! In contrast to most nuclear facilities, shallow-land burial sites cannot be decontaminated and restored. The operators of the commercial and DOE LLW sites have long-term commitments to public safety and the environment. Under current U.S. law (10 CFR Part 61, Subpart E) licensees of new LLW facilities have to provide financial assurances to the regulatory authorities before the license can be granted. Among other things, the licensee must show the availability of funding (for example, using an automatically renewed surety mechanism) sufficient to decontaminate and/or dismantle above ground site structures, close and stabilize the site so that the site owner will need to provide "only minor custodial care, surveillance and monitoring." This process can work. As described earlier in this Chapter, the Beatty, Nevada site was

successfully closed and transferred to the state government for long-term custodial care.

The relatively new Part 61 regulations do not apply to LLW sites that have already closed. An example of the problems that can occur is presented by the Sheffield, IL site. In 1978, US Ecology threatened to abandon the closed site. Illinois then sued them which led to an $8 million settlement. US Ecology agreed to add a 5 foot thick cap of compacted clay and vegetated topsoil over all of the trenches, and to monitor groundwater, surface water, soil, fish and vegetation for a period of ten years. At the conclusion of that period, the state of Illinois then took over responsibility for the site. Finally, US Ecology provided financial guarantees, escrow deposits and commitments for $6.05 million.

The Department of Energy has had its share of problems, too. As a result of secrecy and the arms race, wastes from weapons production received rather low priority. For three decades, beginning in the 50s, Oak Ridge facilities disposed of some 2.7 million gallons in open ponds on the site. As of 1992, over $1.5 billion has been spent on cleaning up the area. The contract for cleanup of the Fernald, OH DOE uranium processing plant was awarded in 1992 for $4 billion. At the Hanford Site in Richland, WA both LLW and HLW from some 50 years of operation of nine different plutonium production reactors is awaiting final disposition. Much work still remains before we have a final solution to the “rad waste problem.”

Problem Set

1. What kinds of things would constitute the solid radioactive waste generated in a typical hospital? Would this be considered high level or low level waste?

2. What is the largest civilian source of radwaste generated in this country?

3. How does the production of yellow cake generate low level radwaste? In what forms is this waste?

4. List two reasons why the average volumes and radioisotopes produced as radwaste at a nuclear power reactor differ for PWRs compared to BWRs.

5. Estimate the total volume of high level waste, solidified into a final vitrified product, that was generated to fulfill your electrical needs for 2009, taking into account U.S. nuclear generating capacity.

6. What is the objective of fuel reprocessing? About how many curies of radioactivity are released per ton of fuel in the process?

7. How did the volume of LLW buried at commercial sites in 2000 compare with 1990?

8. State the basic principle for handling a) high level radwaste, b) intermediate level radwaste, and c) low level radwaste. Give one specific example of a technique used for each category.

9. How does Class A Segregated Waste differ from Classes B and C?

10. What radiation safety problems are associated with a waste compactor? How can they be dealt with?

11. Why are the high level liquid wastes from fuel reprocessing stored for 5 years before solidification? List some problems associated with this storage.

12. What advantages does calcine have over the liquid waste from which it was produced? Why is calcine not an acceptable solid end product for final storage?

13. Give an example of "mixed waste." Why is it so difficult to get rid of?

14. What are the desired properties in a material to be used as the final form for long-term storage in a geologic repository? How does glass measure up?

15. Describe a typical shallow-land burial site for low level waste.

16. How much longer integrity is designed into a Class C trench versus a Class A trench?

17. On the average, based on the Climax Mine experiment, how many spent fuel bundles could be handled at a repository such as Yucca Mountain per year before a worker exceeded 5 mSv annually (i.e., 10% of the allowed dose limit)?

18. Calculate the number of dry casks that would be needed to store the entire core inventory of the largest U.S. nuclear power plant. (Hint: See Chapter 6.) If these casks are then placed vertically, on square concrete pads with one meter of separation between casks, what area would be required to store all of the casks from this reactor?

19. What solution has been implemented at several nuclear power plants to deal with the problem of lack of storage space in their spent fuel pools?

20. Why is it necessary for the WIPP site to provide so much more storage integrity than a shallow-land burial site?

21. What steps have been taken to increase the number of shallow-land burial sites in the U.S.?

22. What is a "Regional Compact?" What are the advantages to the members of a compact?

23. Why is it important to separate radwaste by the radioisotopes contained?

S-1. What are the objectives of the UMTRAP project?

S-2. Estimate the dose equivalent rate at 10 meters from a single West Valley vitrified canister shielded by 3 feet of ordinary concrete. State your assumptions.

S-3. How does the Oklo reactor benefit the designers of long-term storage repositories for transuranic wastes?

S-4. Choose one long-term storage method from the "retrievable" list and one from the "non-retrievable" list. Tell why you think that method is either technically feasible or unfeasible at the present time.

S-5 For what purpose might a health physicist employ phytoremediation?

Other Resources

1. The Waste Isolation Pilot Project (WIPP) web site is http://www.wipp.ws/.

2. "Radionuclide Biological Remediation Resource Guide," U.S. EPA, Region 5 Superfund Division, August 2004, available at http://www.epa.gov/region5superfund.

3. The U.S. Nuclear Regulatory Commission issues annually the "Information Digest," NUREG 1350, which provides up to date information on the status of low level and high level disposal programs in the U.S. It is available online at http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1350.

14

Handling Nuclear Emergencies

Outline of this Chapter

Chapter Summary

In handling nuclear emergencies it is helpful to know what "type" of accident occurred. The chapter begins with several different classification schemes for looking at accidents. The sequential "phases" that an accident progresses through are examined next. Knowing the phase helps to focus on the proper response actions.

Emergency planning before an incident occurs is an important responsibility of every licensee. The various components that make up such a plan are covered. Then the early response to an accident in progress is discussed, followed by a listing of procedures for screening accident victims. Medical intervention will be required in serious radiological emergencies. Some actions that can be taken by licensed physicians are introduced, both for cases of internal exposure and for external exposure. Although the U.S. NRC does not recommend guidelines for emergency doses, the recommendations of other agencies are compared.

Nuclear terrorism involves achievement of criticality with fissionable material. Radiological terrorism involves the release of non-fissionable radioactive material. Both scenarios are discussed along with recommendations regarding the role of radiation protection technologists in such incidents.

Accidents from the past are invaluable for teaching lessons that can be applied to the future. Several "classic" accidents are examined. The events leading up to the occurrence are documented along with actions taken by emergency personnel. Lessons that were learned are stressed. The accident section begins with 4 reactor accidents starting with Windscale in 1957 and ending with Chernobyl in 1986. Next, a fatal accident involving loss of a teletherapy source in Brazil is discussed, followed by details from the 1999 Japanese criticality accident.

In March 2011, the Fukushima Daiichi reactor complex was severely damaged by an earthquake and tsunami. Events resulting from the early handling of the disaster are presented here. The accident was still in the emergency phase as this book went to press.

The historical section concludes with a discussion on radioactive sources that find their way into recycled scrap metal. Finally, some words of advice are given in handling the public relations aspects of a nuclear emergency. A brief review of some of the legal aspects around accident litigation are also presented.

Classification of Accidents & Incidents

Nuclear emergencies, in the United States, are managed as provided under law by the National Response Framework which was approved in January of 2008. The Framework designates a "Lead Federal Agency" responsible for the protection of the public and the environment at the accident site. The Lead Agency monitors the actions of the licensee, assesses the nature and extent of the emergency and advises the licensee on protective actions. The State has responsibility for protecting the public and the environment in off-site areas near the accident. For domestic licensees, the Nuclear Regulatory Commission is the Lead Federal Agency. For situations involving domestic unregulated radioactive materials, or for radioactivity originating in a foreign country (e.g., the Chernobyl accident), the Environmental Protection Agency assumes the Lead position. A number of other federal agencies also have an assisting role as provided in the law. They include Department of Energy, Department of Transportation, National Center for Devices and Radiological Health, Department of Defense, the Interstate Commerce Commission, National Aeronautics and Space Administration, Department of Commerce, Department of Agriculture, U.S. Postal Service, Department of Health and Human Services, and the Department of Labor. As provided in the Atomic Energy Act of 1954, some of the states have assumed control of radioactive byproduct material as will be discussed in Chapter 15. These "Agreement States" also have responsibility for handling nuclear emergencies within their boundaries. By granting of radioactive material licenses, the NRC or the agreement state passes responsibility to the administrative head of a facility (Chief Administrator, President, Director, etc.). This person ultimately has the final legal responsibility for the consequences of a radiation emergency at their facility. It is a matter of record that the initial actions taken immediately by technologists at the location of a radiation accident have often saved time, dose and dollars during the later cleanup phases. Many times, accidents will occur at remote locations or in the off hours when the radiation protection technologist may be the only available person to respond immediately. Thus, it is imperative that the technologist be prepared to take a key role in the early response to a nuclear emergency.

Classification by Damage and Dose

There are a number of different ways to classify radiation emergencies. Some of these ways will be covered here in an attempt to familiarize technologists with the variety of types of incidents and accidents. Title 10 of the Code of Federal Regulations contains a classification scheme involving damage. This system is published as Part 20.2202. Figure 1 summarizes the chief points. For purposes of this regulation, the term "immediate" is interpreted to mean within 30 minutes after occurrence of the incident. It should also be noted that where there is "reasonable evidence ('presumption') that an incident occurred," this situation is subject to the reporting requirements in Figure 1 even if you suspect that the evidence is incorrect. For example, a power failure in an electronic circuit may lead to an overexposure alarm trip. When in doubt, contact the regulatory agency first and then investigate. It is much easier to

IMMEDIATE NOTIFICATION:
TEDE 250 mSv or more
Eye dose of 750 mSv or more to the lens
Skin/extremities dose of 2.5 Gy or more
Release of material in a quantity of 5 ALI

TWENTY-FOUR HOUR NOTIFICATION:
TEDE 50 mSv or more
Eye dose of 150 mSv or more to the lens
Skin/extremities dose of 0.5 Sv or more
Release of material in a quantity of 1 ALI

THIRTY DAY WRITTEN NOTIFICATION:
All of the above events
Doses in excess of occupational limits
Doses in excess of public limits
Dose rates or concentrations > restricted area limits
Dose rates or concentrations > 10 X unrestricted area limits

Fig. 1 - Summary of emergency notification requirements per 10 CFR 20

change a tentative report than to explain why the report was not made at the first sign of a potential problem. See Sample Problem 1.

Each year in the United States, a number of nuclear incidents occur in which notification of the regulatory agency is required of licensees. Out of this number, radiation dosimetry procedures eventually demonstrate that some of the persons presumed to have been exposed in excess of some limit actually were exposed. A compilation of all of the overexposure accident data for the most recent 5 year period,

Sample Problem 1

GIVEN:
A fume hood explosion in a lab ruptures the filter and releases 160 MBq of P-32 to atmosphere.
FIND:
Classify this accident relative to 10 CFR 20 notification requirements.
SOLUTION:
160 MBq x 27 µCi/MBq = 4320 µCi of P-32 released. From Fig. 34, Chap. 9 or 10 CFR 20 Appendix B, inhalation ALI = 900 µCi. The release is thus 4320/900 or 4.8 ALI. Thus, this is a 24 hour notification situation under the 10 CFR 20.2202 release criteria.

(2004 - 2008) shows that only a total of 2 workers exceeded a dose limit for that period. Historically, a large fraction of the overexposures were to industrial gamma radiographers. Logically it might be supposed that a large fraction of radiation workers are industrial radiographers. In fact, they only make up 2% of all radiation workers based on U.S. EPA records. A number of efforts have been directed at reducing this disproportionate rate of overexposures.

Another statistic is enlightening. In the six years from 1975 to 1980, there were a total of 328 overexposures. In the 5 year period from 1995 - 1999 there were 44 cases. Compare these to only the 2 cases during the most recent five years. Presumably this means that you radiation protection technologists out there are really making a difference!

The NCRP introduced, in NCRP Report 111, a classification scheme that falls within this section. Their scheme was intended for use in academic, medical and industrial facilities. It proposes dividing radiological situations into three categories. An INCIDENT is a situation in which an unplanned release of radioactive material or unplanned personnel exposure occurs. The situation becomes a LEVEL ONE EMERGENCY if it is determined that a regulatory limit could possibly be exceeded following the incident. Finally, a LEVEL TWO EMERGENCY is declared if personnel doses could possibly produce non-stochastic biological effects, i.e., doses to skin higher than 3 Sv or doses to any other organs greater than 0.5 Sv are possible.

Classification by Location

For purposes of the regulations, nuclear emergencies are classified as taking place “on-site” or “off-site.” An on-site incident is one which occurs within the legal geographical boundary of a licensee and all the consequences are confined within that location. If radioactive contamination is carried “out the gate” or radioactivity is carried over the fence in a plume, then this classification no longer holds. The on-site incident is easier to handle since it does not require notification of outside public health authorities (if all consequences are confined to the site, no public health hazard exists). However, the incident is still subject to the same notification requirements of Figure 1.

An off-site incident is one which originates at a location outside the facility boundary (for example, a highway transportation accident) or occurs as a result of escape of radioactivity from within the facility. This type of incident is more difficult to handle as the outside public health authorities must be notified. This often results in immediate media coverage which can escalate a minor radiological problem into a major event. Public relations aspects will be covered near the end of this chapter.

Classification by Exposure Conditions

This system involves four different categories. In a NON-CONTAMINATING/ OBSERVABLE incident, persons are exposed to an external radiation field from a source or radiation machine for a relatively short, known period of time. This type of accident involves no release of radioactivity. The consequences are limited to a small

area, usually on-site. Examples of this type of incident would be failure of an accelerator interlock to shut down the beam upon entry of a person into the vault or failure of an industrial radiography source to fully retract into the shield assembly (radiographic camera) at the end of an exposure. The primary radiological problems in this category of incident are terminating the radiation field to prevent further exposures and obtaining a reasonable dose estimate for persons who were in the field.

A NON-CONTAMINATING/UNDISCOVERED incident is similar to the first category in that it involves only external radiation exposures of personnel. However it differs significantly in the fact that the radiation field was present for a long time before discovery, i.e., there was an undetected failure in the radiation protection features so that it is not immediately evident who might have received an exposure. The incident at the Vinca research reactor in Belgrade, Yugoslavia in 1958 provides an example of this category. Apparently a shield plug was accidentally left out of one of the pipes directed at the reactor core during maintenance operations. The pipe was used to extract a radiation beam for experimental purposes. When the reactor was restarted, a number of persons inadvertently walked through a field of neutrons and gamma radiation. By the time it was eventually discovered that the shield plug was missing, six persons had received high doses, five of them in the lethal range. (Doses were finally estimated to be 400 rem for the lowest and 1,100 rem for the highest). The five highest exposed individuals received bone marrow transplants about four weeks after exposure. Four responded with recovery of the blood cell counts to normal levels within two weeks. The fifth died five days after the transplant procedure, apparently as a result of kidney failure and obstruction of the intestinal tract. The accidental loss of industrial radiography sources also falls in this category when it is undetected. A large number of serious accidents of this type have occurred around the world, often with non-radiation workers as the victims. Figure 2 is a photo of a full size mockup gamma radiography “pigtail” source.

One of the more publicized industrial radiography cases involved a 5 curie (185 GBq) Co-60 radiography source which came into the possession of a family in Mexico in 1962. Family members handled the bare source and kept it in a drawer in the kitchen over a period of about four months before the radiographer returned and took possession of it. The ten-year-old son died a month after finding the source (whole body dose was estimated to be between 2940 and 5165 rem). His mother died seven weeks later (dose between 1995 and 2930 rem). The source was reclaimed three days later by the radiographer who wasn’t aware of the problems being caused. A month later, a physician treating the two-and-a-half-year-old daughter suspected that radiation might be the cause of the family’s problems. Five days later the daughter died (dose between 1373 and 1872 rem). The child’s grandmother died two months later of an estimated dose between 1518 and 2827 rem.

Two other cases have been well studied in which a person unknowingly placed a bare industrial radiography source in a pocket and carried it around. A serious case of this type occurred in California in 1979. A machine shop foreman picked up a loose source (28 curies or about 1,000 GBq of ^{192}Ir) at a job site and carried it in his back pocket for about 45 minutes. He then placed it on the desk of a secretary in the office where eight other persons handled it before the radiographer reclaimed it and left, after explaining that there was no problem since the object was a

Fig. 2 - A gamma radiography "Pigtail"

"radiation detector." The foreman received between 80,000 and 400,000 rem to the skin of his hip with an estimated dose at three inches depth of 1,000 rem. Extensive reconstructive surgery has been performed twice on the hip in which the open wound took about two years to heal. He was unable to return to work and is permanently disabled. Four other persons received skin doses from 11,000 to 60,000 rem to fingertips. The radiographer received a misdemeanor conviction and a 5 year probation sentence.

The other case involving carrying of a source in a pocket involved a man who found a 13 curie (480 GBq) Cs-137 radiography source in 1968 at a construction site in Argentina. He carried it in his front pockets for eighteen hours. Since the source was close to major arteries, subsequent tissue damage led to circulatory collapse and amputations of both legs. Isodose curves reconstructed after the accident are shown in Figure 3. A photo of the trunk of this person, ten years after the accident, is given in Figure 4. Part of the tragedy of these cases is that they all could have easily been prevented.

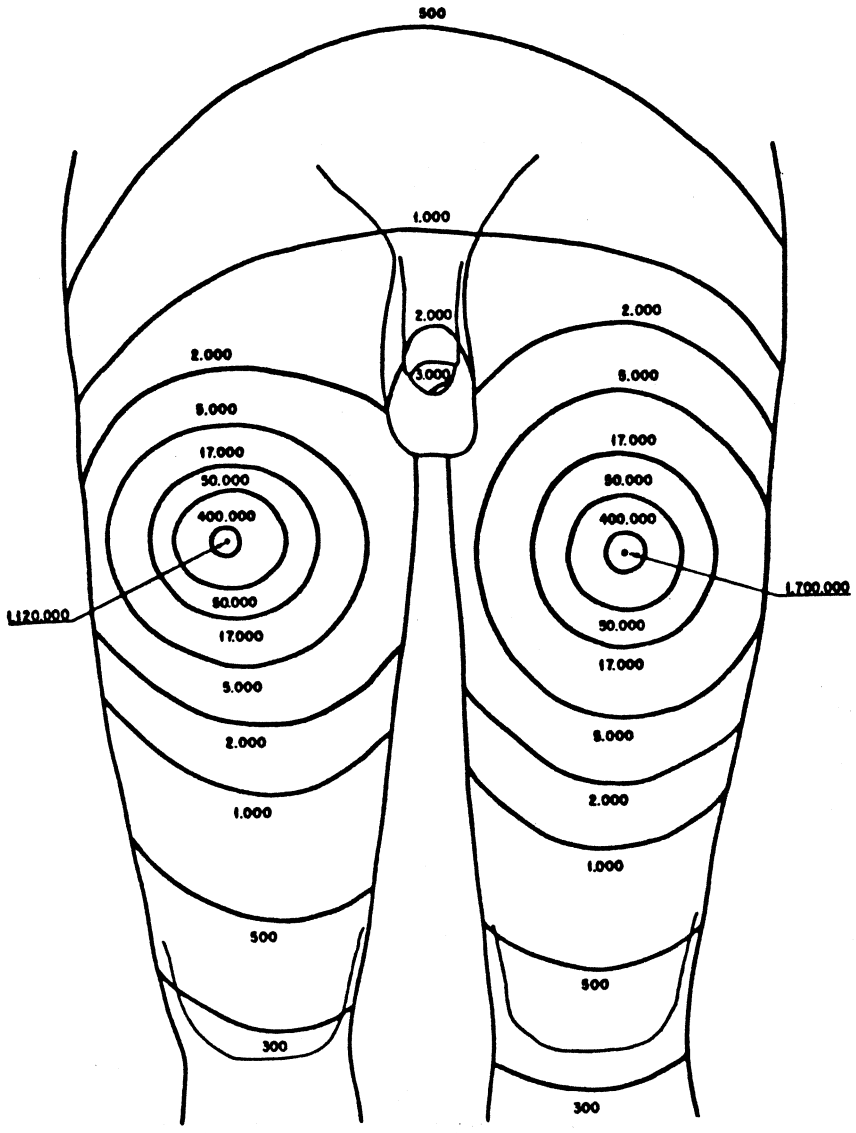


Fig. 3 - Isodose curves for the Argentine accident

The third type of exposure condition category is CONTAMINATING/OBSERVABLE. This involves obvious momentary loss of control of radioactive material in unsealed form. The loose contamination is subsequently inhaled or ingested by an individual. Priorities include limiting the further spread of contamination and prompt medical treatment of internally and externally contaminated persons. Some form of internal dose assessment program will have to be set up (bioassay or in vivo counting). Examples of this category range from a simple spill of slightly radioactive solution in a laboratory experiment to disasters such as the fire in the plutonium glove boxes at Rocky Flats or the 1959 chemical explosion in the plutonium processing

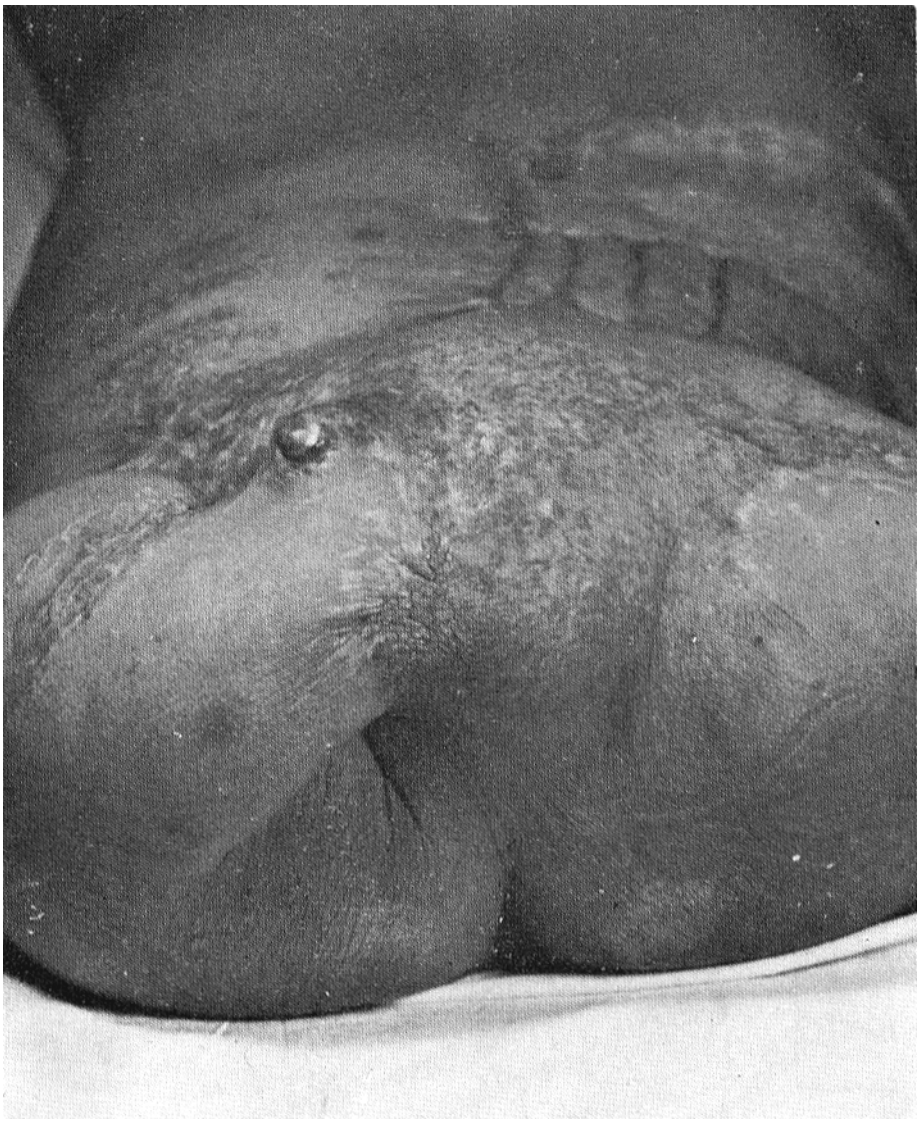


Fig. 4 - The Argentine victim of an industrial radiography accident

facility at Oak Ridge. Major decontamination efforts are time-consuming, costly and a source of "unnecessary" radiation exposure.

The last, and most difficult category to handle is an incident involving CONTAMINATING/UNDISCOVERED exposure conditions. This would result from an undetected release of radioactivity. The radiological problems are similar to the third category just discussed except for the major complication of locating persons who might have been exposed during the time between loss of control and discovery of the condition. This requires widespread radiation surveys to track down and find any possible contamination that was unknowingly carried from place to place. A recent case involving a leaking package transported interstate by air involved monitoring and decontamination in several different states.

An earlier incident points out the fact that even relatively small activities can cause big problems when sealed sources begin leaking. In 1970, some alpha contamination surveys were conducted at an Albany, New York state health department laboratory as a result of a rupture of a small check source. Extensive alpha contamination was found on chairs, desks, benches, file cabinets and other furnishings. It was soon concluded that the check source could not have produced this problem. The contamination pattern led to a jacket worn by an occasional worker in the laboratory. At the home of the worker, extensive americium-241 contamination was evident. A metallic object in a bureau drawer appeared to be the source. It was subsequently determined to have an activity of 49 microcuries. The worker reported that the primary source was likely a plated 10 millicurie Am-241 foil source which he had worked with at a former site three or four years before. Alpha contamination levels in the worker's home were measured up to 100,000 cpm with survey meters. High contamination levels were found in the master bedroom, bathroom, son's bedroom, den, family room and workshop. Air sampling measurements indicated three times the air concentration limit for uncontrolled areas in the workshop and five times the limit in the master bedroom. Whole body counts showed twice the occupational maximum permissible body burden of Am-241 in the worker, ten times the general public limit in one son, two times the limit in the wife and other son, 1.3 times the limit in a daughter and former maid, and 2.5 nCi in the family cat who died during the investigation. (The cat's uptake was legal since there are no published limits for cat burdens). The worker's former laboratory was, also, found to be extensively contaminated along with a moving van that had been used to transport family furnishings five years before. Two other private homes in Albany were contaminated. Decontamination efforts required about four months to complete. The worker is no longer employed by the New York state health department.

Accident Phases

Analysis of radiation incidents that have occurred in the past supports the idea that a given accident progresses through four stages or phases from start to finish. (See summary in Figure 5). Note that in some cases, the first and second of these phases are combined. For example, The U.S. Environmental Protection Agency recognizes three phases - Early Phase, Intermediate Phase and Late Phase for nuclear incident response.

Occurrence Phase

This is the shortest of the four accident phases. It includes the circumstances leading up to the accident and includes the accident itself. Generally this is the most difficult phase in which to get proper documentation. The persons involved have difficulty reconstructing "what happened." A common tendency on the part of accident victims is to assign longer exposure times than actually were the case. For example, in the 1979 California case of the worker carrying the gamma radiography source in a pocket, the estimated skin dose was widely reported as being

Occurrence Phase

The events immediately preceding the accident and the physical happening of the accident.

Emergency Phase

Immediate life and property-saving actions taken by personnel in the near vicinity.

Recovery Phase

Planned, specifically organized actions taken after the emergency phase to isolate accident consequences and physically secure the area.

Restoration Phase

Actions taken at a later date to decontaminate and repair the facility to restore it to pre-accident conditions or to decommission and safely dispose of the damaged sections of the facility.

Fig. 5 - The four phases for an accident

1.5 million rads based on a two-hour exposure time remembered by the worker. Later work based on the biological damage and on chromosome aberration dosimetry showed the most likely exposure time was forty-five minutes.

Emergency Phase

This second phase is also relatively short in time span. It includes personnel accountability to make sure that no one has been mistakenly left in the accident area. After removal of personnel to a safe distance, emergency first aid is performed at the scene. Due to the short time period to this point, only personnel in the immediate vicinity can be counted on to assist. This is a strong argument for requiring emergency training of all workers. The radiation protection technologist on the scene would be expected to play a key role during this phase as senior health physicists would not have had time to arrive yet. The key to success at this stage is a concise, flexible and well-learned emergency plan.

Recovery Phase

The third phase that an accident proceeds through is the recovery phase. Activities now involve the mobilization of a much larger work force to assist. There is time to call on the resources of outside agencies, emergency contractors and trained consultants. Notification of the responsible regulatory agency must be done in this phase. If the accident has off-site consequences, then public health officials must be notified. This means that public relations efforts will begin in this phase. The recovery phase

concludes with the establishment of a physically secure "hot line" around the perimeter of the accident scene. The hot line (a physical barrier) defines the limits of the contamination zone. A single control point is established on the hot line to regulate the movement of personnel and equipment into and out of the contaminated area.

Restoration Phase

This phase is the most time-consuming of the four. It involves all cleanup activities at the accident scene as well as medical treatment and long-term follow-up of exposed personnel. Any legal cases which result from the accident must be handled before this phase is complete. The major decision that must be made early in this stage is whether or not to try to "save" the facility. If the answer is affirmative, then decontamination efforts must be begun first. Next, any repairs needed as a result of the accident must be completed. Finally, the area is refurnished to permit resumption of activities. If the decision to save the facility is negative, then extensive work still remains. Again, the premises must be decontaminated to facilitate disassembly. When the contamination is under control, the facility can be taken apart, piece by piece, and removed to a burial ground. Finally, the site must be decontaminated to acceptable limits. Any removable contamination remaining must be fixed in place. Historically this has been done by applying an asphalt layer over the ground at the site of the SL-1 reactor in Idaho. In some cases of widespread alpha contamination at Oak Ridge, the material was finally fixed by applying paint over buildings AND the grass outside. Note that, although this phase requires a long time to complete, time is not usually a controlling factor in terms of preventing further damage or exposures.

Emergency Planning and Response

Emergency Plan Components

There are a number of factors to be covered in the planning stages as a licensee builds an acceptable emergency plan for a facility. In the case of a nuclear power reactor, the steps in this process are thoroughly covered in existing regulations. In licensed facilities possessing only small quantities of radioactivity, the emergency plan is almost trivial. Figure 6 shows the various factors which must be covered by an emergency plan.

Regarding Item 1 in Figure 6, a list of persons that have the authority to call in outside help, evacuate the site, initiate entry into a contamination zone and take other emergency related actions should be prepared and maintained up to date. It is probably a good idea to list titles of such persons so that personnel changes will not continually obsolete the plan. The specialties likely needed for emergency operations include radiation safety, site security, public information, fire protection, hazardous material control, physical plant services, industrial physician and legal counsel.

1. **On-site Authority and/or Responsibility**
2. **Off-site Agency Contacts**
3. **Credible Site Emergencies**
4. **Warning Monitor Specifications and Locations**
5. **Action Guideline Levels**
6. **Emergency Facilities and Equipment**
7. **Employee Training/Retraining Programs**
8. **Public Relations and Legal Assistance**
9. **Other Hazards Related to Radiological**
10. **Provision for Updating, Testing and Critiquing the Plan**

Fig. 6 - Components of a facility emergency plan

Contact ought to be made with outside agencies before you need their help due to an accident occurrence. Such agencies might include fire departments, police departments, civil defense organizations, hospitals and public health departments. Periodically, the outside agencies should be recontacted to obtain information on their changes of personnel and to inform them of new information or plan changes which might affect their operations in an emergency.

Another step in plan development is to consider all of the possible emergencies that might have some likelihood of happening at the facility. This is dependent on the types of radiation-producing equipment present as well as the radionuclides and activities of licensed sources on the site. Emergency plans should center around the most credible accidents rather than spend a lot of time and effort to plan contingencies for an event that would not have a reasonable chance of occurring.

Decisions should be made as to the types, numbers, ranges and placement of radiation monitoring equipment. Such decisions should be made on the basis of the credible accidents at the facility. Fixed, alarming types of monitors are useful for alerting personnel in the event of an accident, while portable instruments are necessary to effectively deal with the consequences of the accident. It is particularly important to consider the placement of air monitors relative to what you are trying to measure. Breathing zone samplers might give the best indication of routine occupational levels but they might not be the most sensitive or quickest to respond in the event of a major accidental release. One or two instruments should have much higher ranges than required by normal operating conditions. Valuable time can be wasted waiting for the arrival of suitable survey meters in the emergency phase of the accident. This was particularly the case in the SL-1 reactor accident in which lifesaving actions had to await the arrival of survey meters able to read over 500 R/hr.

A variety of action guidelines should be established before a nuclear emergency. Possible guidelines might specify alarming monitor set points, radiation levels to initiate evacuation, acceptable contamination levels, and acceptable radiation doses to emergency personnel engaged in lifesaving or property-saving actions. Current U.S. radiation protection standards do not address most of these issues. Title 10,

Part 20 of the Code of Federal Regulations does require that doses received during accidents and emergencies must be subtracted from the Planned Special Exposure limits for those workers. Although no specific numerical emergency limits are stated, certain licensees may be required to "develop a contingency plan dealing with foreseeable situations including provision of planned countermeasures." It should be noted that the ALARA principle still applies, even under emergency conditions.

Item 6 deals with a consideration of the amounts, types and placement of supplies, apparatus and survey equipment to be used for emergency operations. Multiple storage sites make for easier access in an emergency and also increase the chances of at least some of the materiel being available. With a single stockpile, there is always the chance that the nuclear emergency will directly involve that area, making access impossible. Decontamination supplies and protective clothing should be included. Consideration should also be given to communications systems. Portable radios might prove invaluable in coordinating response and cleanup operations. A dedicated telephone line is a consideration. Thought should be given to the possibility of power failures in conjunction with the accident. Backup systems may need to be arranged. Office and lab space will need to be available on short notice. A complete set of floor plans for the site should be handy. Finally, a computer and calculators will be useful to the emergency staff.

The next area of consideration in emergency planning involves training and education of site personnel. Federal and state radiation control regulations clearly place the responsibility for worker training on the facility management. 10 CFR 19.12 states, "All individuals working in or frequenting any portion of a restricted area . . . shall be instructed in the proper response to warnings or malfunction that may involve exposure to radiation or radioactive material." All workers must be required to have some familiarity with the emergency plan. Concise instructions should be printed and widely distributed around the facility, particularly on telephones. Periodic updating and review training sessions should be scheduled and documented to assure continued understanding of the basic response expected of workers.

Of course, members of the radiation protection staff should receive much more extensive emergency training and more frequent retraining. Their training should include both classroom lectures as well as hands-on exercises with the actual emergency equipment. Outside agency personnel need to be familiarized with the facility layout, use of radioactive materials and/or radiation machines, site emergency plans and procedures and access routes to the radiation areas.

From the earliest phases of an accident, the public relations coordinator has an important role in interfacing with the media and the public. This topic is covered in more detail near the end of this chapter. Legal assistance is almost always required as the consequences of an accident or incident begin to "sink in." Some preplanning in this area will pay dividends should the need arise for legal counsel during an accident.

It is common in institutional use of radioactive materials for the radioisotopes to be incorporated into other inherently hazardous substances. An example would be a radioactively tagged carcinogen used in biomedical research. In the event of an accident, it is thus necessary to worry about the associated other hazards as well as the radiological aspects. In the case of infectious agents, the disinfectant used must be evaluated to see if it has any negative influence on the decontamination methods to

be employed. Materials with chemical toxicity or fire or explosion potential need to be similarly evaluated. The Material Safety Data Sheets provided by the manufacturers are invaluable in this regard. Specific actions to be taken in an accident are spelled out on the Sheet.

Finally, it is necessary to step back and take an overall look at the emergency plan. Top management must give its approval. Outside agencies named in the plan should have a chance to comment. Once a draft plan is in place, it needs to be tested to point out weaknesses. This is best done by holding an emergency exercise or drill.

Planning an emergency exercise begins with a set of objectives which focus on credible emergencies at the facility. A scenario script is then produced to meet these objectives. It needs to be quite detailed and must meet regulatory requirements, if any. If emergency plan weaknesses have been identified, the scenario can focus on those aspects. Facets of the script that are to be simulated must be spelled out. Try to inject some surprises along the way.

Prior to the actual exercise, controllers (observers who produce input data at predetermined times) and evaluators (observers who critically evaluate the performance of emergency workers) must be selected from knowledgeable personnel and then trained in their tasks. At the conclusion of the scenario, the controllers and evaluators need to formally critique the exercise. A review of photos or videotapes can sometimes be useful. When this group has agreed on the strengths and weaknesses seen, the results are presented to the emergency workers and management. The emergency plan is then modified to reflect changes needed to overcome the observed weaknesses.

In addition to the general planning for emergencies just outlined, nuclear power plants in the U.S. have additional specific requirements in this area. Title 10 CFR Part 50.47 establishes two concentric emergency planning zones, EPZs, around each plant. The "plume exposure pathway EPZ" has a radius of about 10 miles. Most population dose in this zone would be due to direct exposure to a plume of released radioactivity and ground contamination. The "ingestion exposure pathway EPZ" has a radius of about 50 miles. Population dose would be due primarily to contaminated drinking water and contaminated food and vegetables. The law requires that the electric utility be able to:

- Classify an emergency into one of four standardized classes,
- Notify off-site authorities
- Recommend public protective actions.

The responsible off-site authorities must be able to:

- Determine the proper protective actions,
- Alert and notify the general public within 15 minutes.
- Assist the public in carrying out the recommended protective actions.

The NRC has standardized nuclear power plant emergencies into four classes according to severity. They are listed in Figure 7.

An "Unusual Event" is some abnormal condition in plant operation which does not pose any public health hazard. A fire in a storage shed might fall in this class as would loss of off-site power to the plant or failure of one or both backup emergency diesel electric generators. No off-site emergency response is necessary. An "Alert" is the next higher severity. The incident results in actual or potential reduction of plant safety, e.g., an accident involving handling of spent fuel. Off-site response is called for.

Unusual Event
Alert
Site Emergency
General Emergency

Fig. 7 - Nuclear plant emergency classes

Local agencies would activate their emergency operations centers to operational status. They would then await further developments. No public action would be necessary. A "Site Emergency" involves actual or likely failure of major plant systems with potential significant release of radioactivity. However, the event would not require protective actions beyond the plant boundary. A leak in the primary coolant system greater than the capacity to replace the water would be an example of this level of emergency. Finally, a "General Emergency" involves the actual or imminent release of high levels of radioactivity outside the plant boundary. This requires complete mobilization of the off-site emergency organizations and probable alert of the general public within the 10 mile EPZ. The loss of integrity of 2 out of three of the barriers to fission product release – fuel cladding, pressure vessel, and containment building – along with potential loss of the third would constitute a General Emergency.

Besides nuclear power plants, there are other NRC and agreement state licensees that are required to have detailed emergency plans. These are designated "major licensees" which means they have the potential to release material sufficient to produce off-site doses above the EPA established Protective Action Guides. Details on the PAGs are given later in this chapter.

To be designated a major licensee a facility has to have a radioactive materials possession limit for sources, in forms other than sealed or special form, which exceed specified limits. For example, a "major quantity" of C-14 is 1,000 Ci, Co-60 is 5,000 Ci, Tc-99m is 10,000 Ci, and Am-241 is 2 Ci. These licensees must then submit a radiological contingency plan that conforms with the "Standard Format and Content for Radiological Contingency Plans for Fuel Cycle and Materials Licensees."

Initial Accident Response

As indicated earlier, the radiation protection technologist would be expected to take a very active role in the handling of a radiation incident during the emergency and recovery phases. To provide some specific guidance, Figure 8 gives a step-by step plan of action for a radiation protection technologist directly involved in managing a nuclear emergency. A practicing radiation protection technologist should be prepared at any time to offer valuable assistance in the event of a nuclear emergency.

At the national level, following the 1979 TMI accident, the U.S. Congress realized that the country needed a better way to coordinate radiation emergency response assets from federal, state and local agencies. Ultimately, after several attempts, the National Response Framework (NRF) emerged (See Other Resources section for more info.) The NRF describes how the U.S. conducts a response to all hazards. Written

1. Lifesaving first aid is the FIRST PRIORITY.

2. Take a moment to familiarize yourself with the work area. Do not act too hastily.

3. Evacuate personnel to safe areas. Make certain all personnel are accounted for.

4. Turn off ventilation system ducts. Take other steps to minimize the immediate spread of contamination.

5. Call for help! Notify facility emergency personnel.

6. Separate injured, contaminated and/or exposed personnel for immediate treatment. See that medical assistance is provided for these accident victims.

7. Obtain survey instrumentation, protective clothing and respiratory protection suitable to the accident conditions.

8. Reenter the area cautiously to locate gross hazards. Take readings quickly to estimate radiation levels and/or set up air samplers. Leave rapidly and evaluate and document readings.

9. Interview witnesses. Determine the radiation sources involved. Relay this information to attending physicians.

10. Establish liaison with facility or outside authorities and turn over control to the facility radiation safety officer or designated person.

Fig. 8 - An emergency response plan for a radiation protection technologist

guidance is augmented with online supporting documents and resources for government executives, private leaders and emergency managers. Overall Federal coordination is under Homeland Security. Radiological incidents are organized under Emergency Support Function #10 - Oil and Hazardous Materials Response. The Environmental Protection Agency has the lead. More specific information is included in the online resources in the form of Incident Annexes, e.g., the Nuclear/Radiological Incident Annex. These are continually updated.

Once a radiation emergency has occurred, either the affected state or the Lead Federal Agency can activate the FRMAC, the Federal Radiological Monitoring and Assessment Center. This is an interagency asset and it responds in a phased approach after the Department of Energy issues the emergency declaration. The

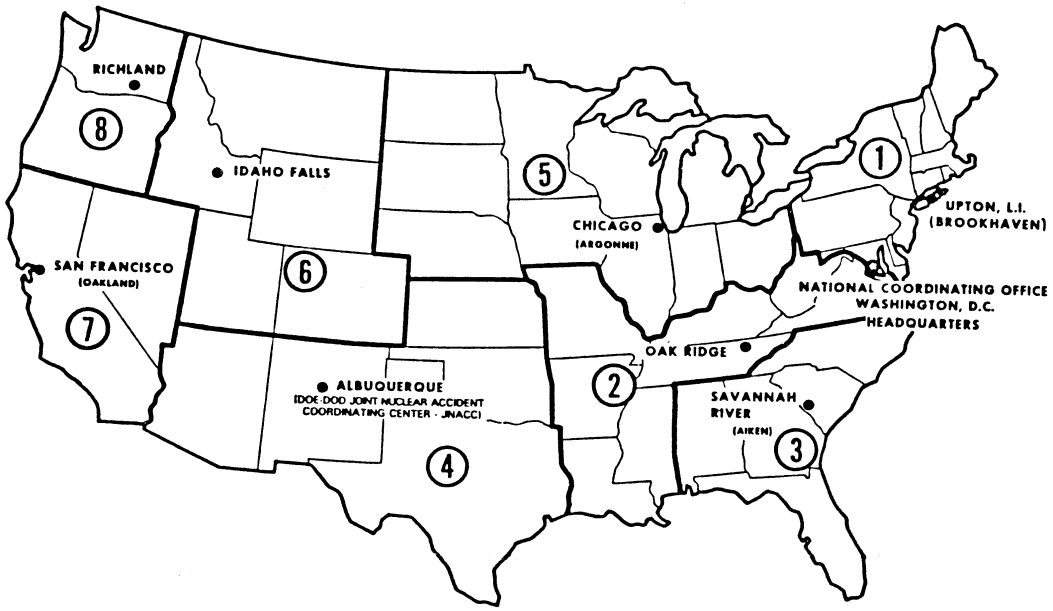


Fig. 9 - U.S. DOE radiological emergency team RAP regions

Department of Energy maintains Radiological Assistance Program (RAP) teams at a number of geographical sites around the U.S. (They can also be called upon by a licensee to provide expert help in handling a nuclear emergency.) Figure 9 is a map showing the geographical distribution of the 8 RAP team regions.

The RAP team acts as the Advance Party following FRMAC activation. Based on it's on-scene assessment, a FRMAC Phase I response can be in place on-scene in 8 hours if appropriate. A FRMAC Phase II response can follow in 11 hours and a Full-Field FRMAC, consisting of from 60 to 500 persons, can be set up in 24 to 36 hours.

The Nuclear Emergency Search Team, NEST, was established in 1974 by the Nevada Operations Office of DOE in Las Vegas. Its primary role is to maintain a capability to respond to potential nuclear terrorism. It consists of a group of volunteer scientists and technicians from the nuclear weapons development program who are on call in the event of an incident. The NEST response includes the search, detection, and identification of lost or stolen nuclear weapons or special nuclear materials or sources involved in radiation threats. They can be deployed within 2 hours. Their equipment includes briefcase gamma-neutron monitors, airplane mounted detectors, and roadblock monitors that scan traffic in and out of an incident area. They can provide on-site support in disarming or disabling nuclear devices. The team was used for three months in 1978 in the Canadian Northwest Territory to locate and recover hundreds of parts from the fallen Soviet Cosmos 954 satellite containing a nuclear reactor.

The Environmental Protection Agency staffs two Radiological Emergency Response Teams, RERTs, one of which is on continuous standby alert at all times. They can reach any point in the U.S. within 24 hours. They are supported by three EPA radiological labs and a fleet of mobile counting laboratories that can provide complete radioanalytical services and communications support. The EPA is the lead

agency in accidents involving unregulated rad material, e.g., lost sources, sources of unknown origin or NORM material.

The U.S. Nuclear Regulatory Commission and Department of Transportation both maintain emergency centers that can provide assistance in case of a radiation accident. The numbers that follow are available 24 hours a day. The second number is especially for spills of hazardous material, including radioactive material. Both numbers can refer a technologist to the closest available assistance.

NRC Emergency Operations Center - (301) 816-5100
DOT National Response Center - (800) 424-8802

If the nuclear accident involves human exposures, **competent medical assistance must be made available to the victims immediately**. One of the best sources is the Radiation Emergency Assistance Center Training Site or REAC/TS, operated by the Oak Ridge Institute for Science and Education in Oak Ridge, TN.

Contact Information for REAC/TS:
Telephone 24/7: (865) 576-3131
Web Site: www.orau.gov/reacts

REAC/TS has physicians available, around-the-clock, who are experienced in handling radiation accident victims. They can consult by phone with local doctors who are treating accident victims and provide them with a wealth of information. REAC/TS also maintains a registry of world-wide accidents involving radiation exposure or intake of radioactive materials. Details of treatments used in previous accidents can be very useful in planning for current victim treatment.

Technologists should be aware of one further resource. The U.S. Department of Health & Human Services maintains a website offering guidance managing the medical aspects of radiation accidents. It offers specific help in sorting victims (triage) and is an invaluable guide for physicians who find themselves swept up in a serious radiation emergency. It can be found at the following address.

Website for Radiation Emergency Medical Management:
http://www.remm.nlm.gov/index.html

Procedures used at an accident scene deserve a little more explanation. The purpose of the hot line is to physically isolate radioactive contamination. The "line" itself must be capable of preventing the spreading of loose contamination and it must be capable of preventing unauthorized entry by personnel. A single "control point" is established to provide for authorized entries, monitoring and decontamination services on exit and complete record-keeping. Particular procedures must be observed at the control point to prevent transport of loose contamination across the line when persons are exiting the restricted area. Figure 10 describes the steps

1. Place instruments and equipment on an area covered with paper or polyethylene, located on the HOT side of the hot line.

2. Remove tape seals from your protective clothing and place tape in a marked hot waste plastic bag.

3. Be frisked by hot line monitor who will pay special attention to hot spots on the back of the neck, hands and feet.

4. Remove hood and place in appropriate container.

5. Remove coveralls and place in appropriate container.

6. Remove one shoe cover. Place it in the container and step over hot line to the COLD side.

7. Remove remaining shoe cover, dispose of it and step over hot line.

8. Remove respirator and cap and place in appropriate container.

9. Slip off your gloves, without touching the outside surfaces, and place them in the container.

10. Be frisked by monitor.

11. Wash up. Be rechecked by monitor if skin contamination was found previously.

Fig. 10 - Hot line exit procedures at the control point

involved in passing through the control point to the uncontaminated side. These procedures are illustrated by Figure 11.

Before moving on, it is worthwhile pointing out the obvious that reducing the spread of contamination early on will pay dividends later. If less contamination is spread around, it will be easier to clean up. Stepoff pads, also known as contamination control mats or adhesive mats, are an accepted tool in reducing contamination problems. The pads are the size of an ordinary floor “Welcome” mat but they have a sticky surface on the top. The adhesive in the surface removes particulates from the bottoms of your shoes when you walk across it. Often they are supplied with 30 to 60 adhesive sheets all held together in a stack. When the top sheet is used up, the next sheet is exposed by peeling off the dirty sheet.

Fig. 11 - Illustration of hot line exit procedures

Medical/Radiological Priorities

Early responders in a serious radiation emergency must constantly have in mind the fact that medical responders and health physics responders operate with different priorities. Furthermore, the medical needs of the victims ALWAYS HAVE ABSOLUTE PRECEDENCE OVER THE RADIOLOGICAL NEEDS. Therefore, the first

medical priority is that exposed, contaminated and/or injured survivors need to receive life and limb saving treatment whether or not they are contaminated.

Once the medical patients are stabilized at the scene, the first radiological priority is determination of the extent and amount of the radioactive contamination. If initial measurements indicate that victims might have been internally contaminated, this information must be relayed to the medical responders. While gathering this information, try to limit the spread of further contamination.

The second medical priority is to deal with internally contaminated victims. This is a time critical situation as has been mentioned before, radionuclides rapidly seek out internal organs where they deposit. A number of techniques are available to physicians to reduce internal doses, but they are most useful if begun within hours of the accident.

The second radiological priority is to assist medical personnel in decontamination of contaminated persons. The order in which this should proceed is to clean up open wounds first. The next step is decon of body orifices (nose and mouth), followed by intact skin cleanup. The third radiological priority for the radiation protection technologist is full characterization of the contaminant(s). Which radionuclides are involved and in what quantities and physical forms? Are any of them airborne? Radiation spectrometers should be located and activated. Similarly, air samplers should be set up. If air sampler results prove negative, that information needs to be passed along as well. The following sections discuss these basic priorities in more detail.

Emergency Screening

As mentioned in the step-by-step response procedures for radiation protection technologists, the screening procedures should be begun at an early point during the emergency phase to separate accident victims into categories such as clean vs. contaminated, injured vs. non-injured and exposed vs. non-exposed. Persons with injuries should be directed to medical assistance while the contaminated, non-injured should be taken to a decontamination facility. The initial screening during recovery and restoration phases will take place at the hot line control point. Radiation monitors should collect dosimeters and check for external contamination. If appropriate to the hazards involved in particular entries, nasal swabs and urine samples should be collected.

Under emergency conditions, a rapid screening procedure is frequently used when exposure to a fast neutron field is suspected. Capture of the neutrons by sodium atoms in the blood results in the formation of Na-24 activity. This radionuclide decays by photon emission. A geiger counter placed near a large blood volume will thus detect this activity. The common practice is to place a geiger counter probe under the armpit of the exposed person. This surrounds the detector with a reasonably large blood pool. The rule of thumb is that 500 rem of fast neutrons delivered acutely to the whole body of Reference Man will produce a gamma exposure rate of 1 mR/hr in the armpit. Clearly, this measurement should be made in a low gamma background area. See Sample Problem 2.

Neutron exposure actually produces detectable levels of several activated radionuclides in a human body. For example, 20 millirads of fission neutrons will generate 10 nCi of ^{24}Na along with low levels of ^{49}Ca,

Sample Problem 2

GIVEN:

Following a criticality accident, a victim measures 0.3 mR/hr with a GM probe under his armpit. The background rate is measured to be 0.1 mR/hr.

FIND:

Estimate the fast neutron dose received.

SOLUTION:

From the text above, a 1 mR/hr net reading implies 500 rem of fast neutrons. The net reading is 0.3 mR/hr - 0.1 mR/hr = 0.2 mR/hr. Thus, the estimated dose is (0.2 mR/hr) / (1mR/hr per 500 rem) = 100 rem of fast neutron dose.

^{38}Cl, ^{28}Al and ^{13}N. Because of short half-lives, only the ^{24}Na is practically usable by a whole body counter facility under these conditions.

As indicated above, a decontamination station should be located near the hot line control point. Plastic sheeting is useful to have available. The person can stand on the sheeting while removing contaminated clothing. It can then be bundled up and labeled for future measurements and analyses. Washwater should similarly be saved to aid in determining the internal and external doses received by the person. A supply of clean clothing will need to be made available.

Early Dose Assessment

To properly manage the medical treatment of exposed victims, an initial "educated guess" as to the magnitude of the dose of each one is very helpful. Unfortunately, there are virtually no tools readily available to assist this process. The most commonly used method in the initial hours post-accident is "time to emesis." This refers to the elapsed interval between exposure of the victim to radiation and the time they start to vomit. REAC/TS has combed through its Radiation Accident Registry files and summarized the results in the form of a table of acute, gamma equivalent dose versus the emesis time interval. The data suggest that about half of a population of victims will vomit if exposed to about 2.7 Gy of photon radiation. Figure 12 graphs the time to emesis vs. dose. Note that all bets are off if the victim is exposed for an extended period of time, rather than acutely.

After the first day post-accident, the measured rate at which lymphocytes are decreasing in the circulating blood gives another dose estimate. However, this technique is not accurate if only partial body exposure occurred. Also, the result is affected by injuries to the patient. Finally, note that skin erythema is not particularly useful for early dose results. The appearance of the reddened tissue takes from days to a few weeks to develop.

There is some hope for improvement in early dose assessment. As covered in Chapter 4, researchers have developed ESR techniques to measure the dose by measuring teeth, and some chromosome aberrations can be detected by automated cell

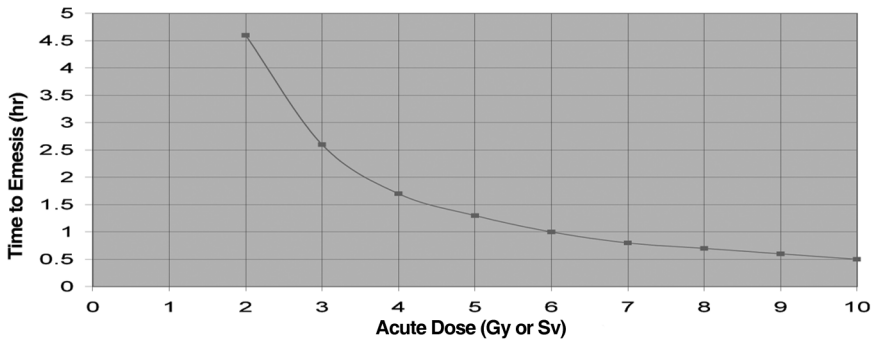


Fig. 12 - Approximate time to vomiting following an acute radiation dose

sorting equipment. At the present time, such equipment is not widely available and, because of the limited sales market, probably will not be for some time.

Medical Intervention

First we will consider intervention in those cases of internal contamination. Speed is essential. As a general rule, most of the activity that will be retained by the body is fixed in place within one to two hours following the uptake. In the case of actinides (e.g., Am or Cm), experiments have shown that bone deposition is 76% complete in only one hour. Inhaled radioiodines reach equilibrium with the fluids in the body in just thirty minutes. Passage of soluble contaminants through the wall of the intestinal tract takes only thirty to sixty minutes. Through proper medical intervention, the dose due to an uptake of radionuclides can often be greatly reduced.

Three basic techniques have proven effective. These are isotopic dilution, metabolism stimulation, and chelation therapy. Before considering the details of these forms of treatment, it must be emphasized that **any medical intervention must be done only by a licensed medical practitioner. If the techniques to be described are recommended by a radiation protection technologist, you could be convicted of practicing medicine without a license**, a serious offense.

In isotopic dilution, the idea is to "block" the uptake of a particular radionuclide by flooding the body with the stable form of that same chemical element. The most widely known example involves radioiodine uptakes. The thyroid gland is the critical organ in these cases. If 100 milligrams of ordinary, stable iodine are administered within two hours or less of uptake, the final uptake fraction by the thyroid is reduced by about 90%. Treatment begun after a 4 hour delay reduces the blocking effectiveness to less than 50%. In 1982, the U.S. Food and Drug Administration approved the use of potassium iodide in radiation emergencies under the direction of state and local public health officials. In 2001, they placed the action levels at 5 rem to the thyroid of children (birth - 18 years) and to pregnant or lactating women. Adults 18 to 40 years were set at 10 rem while persons over 40 were set at 500 rem. A recommended dose is 130 mg/day of KI for persons over 18. Children 3 through 18 should receive 65 mg/day. Ages 1 month to 3 years should receive 32 mg/day while infants birth to 1 month were set at only 16 mg/day. Treatment should continue for 48 hours past the last exposure to radioiodine. Numerous sources are available for KI pills produced by FDA approved pharmaceutical companies. (A Google search for "KI pills for radiation" produced over 98,000 hits!)

In 1985, FEMA, the Federal Emergency Management Agency reaffirmed the use of KI by public health officials and offered additional guidance on its use under emergency conditions. Their recommendations conclude that "the use of KI to prevent radioiodine from accumulating in the thyroid gland can be an effective ancillary protective action during a nuclear power plant accident. However, many factors make stockpiling and/or pre-distribution to the general public questionable. Whether KI should be stockpiled and distributed to the general public around a particular site depends on local conditions."

In 1989, a research group from Penn. State College of Medicine published an interesting human study. When 8 ml of 2% tincture of iodine solution was painted over a 200 cm^2 area of abdominal skin on volunteers, the thyroid uptake of I-131 at 24 hours was statistically identical with a group of volunteers that took 130 mg of KI prior to the orally administered I-131! In both cases, the blocking was around 90%.

The NRC's NUREG-1633 (see Other Resources) points out that the ICRP thyroid model is not representative of Americans because the high levels of stable iodine in fast-foods already provide partial blocking against radioiodine uptake. (Fast-foods have over 30 times the minimum daily requirement of iodine!)

Potassium iodide is not for everyone. Following the Chernobyl reactor accident, 18 million KI doses were administered in Poland. 36,000 significant medical reactions and 2 serious medical reactions were reported. The list of adverse reactions includes swelling of the arms, face, legs, lips and tongue, joint pain, and hives. KI should not be taken by pregnant women, nursing mothers, and persons with hyperthyroidism, enlarged thyroid glands or known sensitivity to iodine.

Metabolism stimulation refers to speeding up of normal body physiology with the intent of causing the internally deposited radionuclides to clear more rapidly. (Remember that the committed dose equivalent is directly proportional to the effective half-life in the body). Tritium uptake provides a familiar example of this technique. By increasing the turnover of body water, the biological (and hence the effective) half-life is reduced. Diuretic drugs and increased fluid intake can reduce doses by about 50% or so. Again, remember that such treatment must be prescribed by a physician.

Chelation therapy involves administering a chelating agent which is a chemical compound that will readily bind metal ions into a soluble complex. These, in turn, will be readily excreted in urine. This decreases the effective half-life and prevents uptake by other tissues. One of the most common chelating agents in use today is diethylenetriaminepentaacetic acid (affectionately known as DTPA). In experimental animals, DTPA administration within one hour of uptake has caused the urinary excretion of 83% of injected plutonium-239 citrate. In general, DTPA is effective with inhaled or injected (e.g., via a wound or skin break) soluble actinides if administered within fifteen to forty-five minutes after intake. Experience with human uptake of plutonium (Pu) shows that DTPA can remove 65% of bone deposited Pu over the course of six months of treatment. DTPA is considered safe for non-pregnant adults unless there is a history of kidney disease or bone marrow depression. The table in Figure 13 shows some of the medical treatments recommended for uptake of various radionuclides.

One caution should be sounded with regard to chelation therapy. The long-term use tends to deplete the body of necessary trace elements. Early research with Ca-DTPA readily demonstrated toxic side effects after

Nuclide	Medication	Comments
Am-241	DTPA by IV or aerosol	Works on bone even after long deposition
Cf-252	DTPA by IV	
Cs-137	Prussian Blue	Consider ion exchange resin
Co-60	Stomach pump	
H-3	Forced H_2O	Isotopic dilution
I-131	KI or KIO_3	Give in first 2 hours
P-32	Phosphates	Isotopic dilution
Pu-239	Ca DTPA	EDTA less effective
Sr-89/90	Sr or Ca by IV	Consider alginates

Fig. 13 - Some medical intervention procedures for internally deposited radionuclides

prolonged use. Zn-DTPA seems to be much less hazardous in the long-term, and is the current chelating agent of choice for removing plutonium from liver tissue and americium from both liver and skeletal tissue. Regular monitoring is recommended for essential trace elements in the blood. These levels can be increased by ingestion of vitamin and mineral supplements during the treatment course. Also, the patient should be advised to avoid dehydration as this reduces blood flow in the kidney which, in turn, increases kidney damage from the DTPA.

The U.S. Food and Drug Administration (FDA) has approved DTPA for treatment of internally deposited Pu, Am and Cm. Some additional radioactive elements can be treated with "special permission." The drug is administered using a slow drip intravenous bag or alternatively, over a 3 to 5 minute period by injection into the vein. In some cases it is injected into muscle but this causes severe pain near the injection site. For cases of lung contamination by transuranic elements, it has been found that DTPA is still quite effective when inhaled in a mist. Current practice (as of 2010) suggests that Ca-DTPA should be given as the initial dose and then subsequent doses should be Zn-DTPA.

For gross contamination cases, DTPA has been given up to 5 days per week over periods up to several years. The Strategic National Stockpile maintained by the U.S. Centers for Disease Control and Prevention, CDC, has both Ca-DTPA and Zn-DTPA available for emergencies.

While not chemically classified as a chelate, Prussian Blue (PB) performs a similar task when administered to persons with a radioactive cesium intake. First used as a blue dye for Prussian military uniforms in 1704, ferric ferrocyanide, PB, is considered safe for adults, children and infants. When it encounters cesium in the intestines, it traps the cesium which is then excreted through bowel movements. The

side effects of use, such as constipation and stomach upsets can be readily treated by other medications. PB is approved by FDA for cesium contamination treatment.

The Goiânia accident in Brazil, to be discussed near the end of this Chapter, was probably the first widespread use of Prussian Blue to reduce radiation doses to a human population exposed to Cs-137. There were 46 victims treated successfully. The administration of PB reduces the biological half-life from the normal 115 days down to about 40 days. The Strategic National Stockpile carries 500 milligram capsules which are prescribed 3 to 4 times daily for up to 5 months. The patients must be monitored for loss of electrolytes and potassium.

Turning now to medical intervention for victims of external radiation exposure, the treating physician's "bag of tricks" is rather small. The classical treatment of choice was bone marrow transplantation. The procedure is commonly used in leukemia patients. Patients are treated with 10 to 12 grays of total body irradiation acutely. and then receive new, non-cancerous marrow from a tissue matched donor. Historically, the experience in the use of bone marrow transplants with radiation accident victims is unimpressive. Of the 13 Chernobyl victims treated, only 2 survived. In the 5 cases treated in the Vinca reactor accident, it was not felt that the procedure helped the victims. In a 1968 accelerator overexposure case, the victim received marrow from a twin brother and his blood counts recovered completely, well before expectations. Thus, current thinking is that bone marrow transplants should be one consideration for treatment in cases of whole body exposure in the 8 to 12 gray range. However, expectations should not be high.

In just the last few years, recombinant gene technology has developed a promising new technique, cytokine therapy. Cytokines are proteins released by cells when they encounter specific antigens. Human growth factors are now commonly used with radiation accident victims. The historic first use on human victims was in the Goiânia, Brazil accident mentioned earlier. It was recognized in the 1960s that blood cells are incapable of cell division without the presence of very low concentrations of specific molecules named colony-stimulating factors (CSFs). Modern recombinant DNA techniques allowed for the cloning of CSFs in sufficient amounts to be used for therapy. Eight Brazilian victims were treated with granulocyte macrophage CSFs (GM-CSF). Four subsequently died of hemorrhage and infection. The other four showed increased granulocyte counts following treatment. In one case, the drug was stopped and the count fell. Resumption of the G-CSF resulted in the granulocyte count rising immediately.

The trade name for the CSF drug originally developed was Neupogen®, marketed by Amgen, Inc. It is currently available as the drug Leukine. This drug was FDA approved in 1991 for cancer patients with bone marrow damage. It has not been approved for radiation accident patients although it is expected to work as well for them. This application would be considered an "off-label" use. The pharmaceutical is considered safe for most adults but not for persons with high sensitivity to E. coli-derived proteins.

As of this writing (2005) the CDC has submitted paperwork to the FDA to make Neupogen® available in the National Strategic Stockpile specifically for treating acute radiation syndrome in persons exposed to high external doses of radiation.

This human granulocyte colony stimulating factor, G-CSF, is a hormone-like protein that acts as a communication link between cells. Neupogen® stimulates the body to produce more white blood cells. It is injected daily under the skin at a dosage of 5 micrograms per kg body weight for up to two weeks post-irradiation. The possible side effects include fever, diarrhea, skin rash and weakness.

A very close cousin, drug-wise, is Neulasta®, also marketed by Amgen, Inc. It differs from Neupogen® in that a polyethylene glycol group has been added. This causes a large increase in the biological half-life of the drug. In effect, Neulasta® is a long-lasting version of Neupogen® so that daily injections are no longer necessary.

In the case of external whole body exposure, experience has shown that early medical treatment improves the chance for survival. Thus, it is important, in the early accident stages, to be able to identify victims with high doses. In the case of inadequate dosimetry, the responding medical personnel need alternate methods to perform this task. In a study of lymphocyte depletion in 43 accidental exposures reported by the Radiation Emergency Assistance Center/Training Site (REAC/TS) in Oak Ridge, TN, they suggest a formula for calculating an approximate whole body dose from the first 8 hours of lymphocyte counts in a victim (see "Other Resources" Item 9 in this Chapter for more information). The method is suggested for an early dose estimate only, so that treatment can begin if warranted.

As of 2011, there have been enough advances in the treatments for acute radiation syndrome that the chances of surviving high radiation doses have been significantly improved. Currently, exposures in the 800 to 1200 rem whole body may be survivable with treatment. In the 1980s, survivability was not expected above 600 rem.

Special Cases

Plutonium is hazardous from three standpoints. It presents an internal hazard from the alpha particle emissions. It is a pyrophoric material which means that pieces of the metal can spontaneously ignite when exposed to air. Finally, there is a potential for criticality accidents. As shown by Figure 57 of Chapter 8, as little as 900 grams of Pu-239 in solution can achieve criticality.

The atomic number of Pu is 94. The most common isotope has a mass number of 239. This isotope decays by emitting alpha particles with an average energy of 5.1 MeV. The dose rate to skin in contact with a metal surface is about 400 mrad/hr. The physical half-life is 24,065 years while the biological (and effective) half-life in bone is 200 years. Of the plutonium that becomes dissolved in the blood, 90% normally will attach to the skeletal system. Inhalation of Pu particles of about 1 micron in diameter results in 25% reaching the lung. 40% of that activity clears rapidly with a one day half-life while 60% clears with a 500 day half-life via lymphatic drainage.

Clearly, respiratory protection should be used when working with this material. What is not always realized is that, in an emergency, MANY COMMON HOUSEHOLD ITEMS CAN BE USED TO GIVE SIGNIFICANT RESPIRATORY PROTECTION. An Army Chemical Corps study measured the

DESCRIPTION	COLLECTION EFFICIECNCY
Man's cotton handkerchief, 16 thicknesses	94%
Man's cotton handkerchief, 8 thicknesses	88%
Toilet paper, 3 thicknesses	91%
Bath towel, 2 thicknesses	85%
Cotton shirt, 2 thicknesses	65%

Fig. 14 - Filtration efficiencies of several common items

efficiency of several common items for 1 to 5 micron diameter particles. The results are displayed in Figure 14. It was further determined that all of these items performed better when DRY RATHER THAN WET, usually because the breathing resistance was too high when they were wet.

Criticality accidents are another special case. These were covered in Chapter 8 under the section on criticality badges. It might be helpful to review that material at this time.

A fire emergency involving nuclear material is particularly hazardous. Firstly, fires increase the possibility of the spread of airborne contamination. In addition, the heat can melt lead source shields so that emergency personnel may be exposed to high radiation fields. Finally, if fissionable materials are present in significant quantities, fire suppression techniques may increase the chances of criticality accidents. Water acts as a moderator and, thus, increases the effective multiplication constant for the assembly.

The radiation protection technologist has several roles when fire is a part of an accident. Personnel should be kept upwind before the firefighters arrive. Monitoring of the fire personnel and their equipment is a second task. The technologist should see that all emergency personnel are provided with respiratory protection. Finally, the on-site technologist has the responsibility of informing firefighters of the potential radiation hazards. If criticality is possible, the amounts of water (including the RUN-OFF WATER FROM DISTANT LOCATIONS) should be limited as well as the numbers of personnel in the vicinity of the fissionable mass (firemen and radiation protection technologists are good moderators, too!).

Fire prevention is important in the design of nuclear facilities. In glove boxes, noncombustible construction materials should be used. If pyrophoric materials are used, provide thick metal glove box floors so these materials will not burn through if dropped. The thick floor also acts as a heat sink to help extinguish the fire. Windows should be of safety glass or self-extinguishing plastic. Air filters can be obtained which are fire resistant.

The final special case involves accidents with loose contamination as a complication. Decontamination techniques then play an important role in the overall management of the incident. As a rule of thumb, 90% of personal contamination will be removed with the clothing and another 7% will be taken care of with a shower.

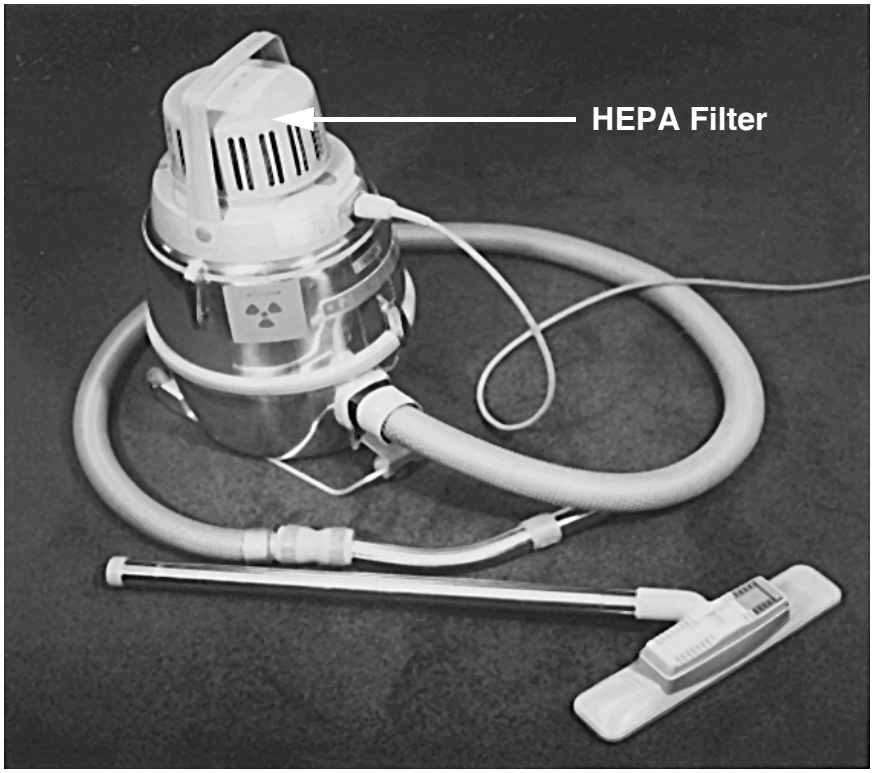


Fig. 15 - A HEPA filtered vacuum cleaner

Remember that intact skin is an excellent barrier. Decontamination steps should be avoided which would reduce the skin's effectiveness. For this reason, wash carefully with mild soap and water, at least in the beginning. Work from clean areas toward hot spots of localized contamination. Take care to adequately check body folds, finger nails and hair for contamination. Also watch out that the contaminated washwater doesn't spread contamination more widely.

In the case of contamination on surfaces, try removing some of it by applying masking tape or "strippable" paint. Protect newly cleaned surface areas with taped plastic sheeting to prevent re-contamination. The wet type of vacuum cleaners are very useful in decontamination work. High efficiency particulate filters (HEPA) can also be attached to some models of both wet and dry vacuum cleaners (Figure 15). This prevents resuspension of the radioactive particles as an airborne hazard.

When more aggressive techniques are called for, one solution is the use of abrasive blasting equipment. Conventional sandblasting is effective at removing contamination. However, the downside is that it generates a large volume of contaminated blast media which is expensive to dispose of. Recyclable systems reduce the waste volume. An example of a unit that is commercially available is the PlasBlast™ decontamination system offered by Bartlett Services Inc. of Plymouth, MA. By enclosing the work area, the possibility of creating an airborne hazard is greatly reduced. The unit uses a high velocity stream of plastic particles to clean virtually any surface. The decontamination is accomplished without removing surface metal. A HEPA filter

follows a prefilter to prevent removed contamination from being suspended in room air. The plastic particles are recycled which greatly reduces the volume of radwaste that is generated. The particles are also incinerable. A more recent development is the availability of dry ice pellet blast equipment. The frozen carbon dioxide is quite effective at removing surface scale and then it sublimes (changes to gaseous form) and dissipates. Thus, there is no additional secondary waste volume added to the waste stream. In the future, look for lasers to play a role. Figure 16 summarizes some recommended decontamination techniques for a variety of situations.

A number of standard contractor tools have found homes in commercial decontamination service companies. Concrete planers have rotary diamond disks that can remove thin sections of surfaces. Concrete scabblers include needle scalers with multiple tungsten carbide needles that are driven against the surface to break out small chunks as the tool is moved over it. Concrete scarifiers are another contractor tool. They employ a diamond embedded drum that rotates at high speed over the surface. Both electric and gasoline motor versions are readily available, typically in widths such as 8", 10" and 14". The depth of cut is adjustable.

Nuclear and Radiological Terrorism Response

Nuclear and Radiological Terrorism Scenarios

The news media seems to raise the threat of a terrorist incident involving nuclear materials or weapons almost weekly. In 2001, the Russian government reported on 601 attempted sales of nuclear material since 1998. Then in 2003, 19 persons were arrested and charged with conspiring to destroy a nuclear power station on Lake Ontario. Revelations in 2004 of the sale of uranium enrichment technology and nuclear weapon designs to Iran, Libya and North Korea by Dr. Kahn, Pakistani head of their nuclear weapons program, has increased the risk of an attack.

There are four scenarios described by experts that cover probable terrorist events involving radiation and/or radioactive material. To be considered nuclear terrorism, a fissionable yield must be produced. Thus, theft and use of a complete nuclear weapon, and construction and explosion of a crude nuclear weapon using fissionable material (an "improvised nuclear device" or IND) both qualify. In contrast, a physical attack on a nuclear facility that releases significant radioactivity and the explosion of non-fissile radioactive materials (a "radiological dispersal device" or RDD) are classified as radiological terrorism. Each scenario will be treated here briefly.

It has been estimated that the cost of a nuclear weapon exploded in an urban U.S. area would reach trillions of dollars. This includes property damage from the blast, cleanup of radioactive contamination, economic losses due to long-term shutdown of the area and cost of long-term health effects. For some photos of the Japanese bombing, (see Figure 17). Figure 18 lists the four major effects of a nuclear explosion and lists 50% survival distances for each. A rough estimate of the explosive yield in kilotons of TNT can be made by the duration of the bright flash accompanying the detonation. Figure 19 provides the necessary information.

METHOD	SURFACE	TECHNIQUE	ADVAN-TAGE	DISADVANTAGE
VACUUM CLEANING	Dry	Conventional vacuum.	Avoids water reactions	Dust must be HEPA filtered on exhaust
WATER	All nonporous	High pressure hose. Work from top to bottom.	Can work at a distance. About 50% reduction. Can add decon agents to water.	Won't work on oiled surfaces. Must catch drain water. Absorbed by porous surfaces.
STEAM	Painted or oiled nonporous.	Work top to bottom. Can add detergents.	About 90% reduction on paint.	Wear waterproof outfit. Spray is contaminated.
DETER-GENTS	Nonporous	Rub then wipe with dry rag. Powered brush is more efficient. Only need a moist application.	Dissolves contam. industrial film.	Relatively mild.
COMPLEX-ING AGENTS	Nonporous. No rust.	Use about a 3% solution.Spray application. Stand for 30 min. & rinse.	Holds contamination in solution.	Little penetrating power. Won't work on weathered surface.
ORGANIC SOLVENTS	Nonporous	Immerse entire unit. Can also be wiped on.	Quick dissolving action.	Needs ventilation & fire precautions.
INORGANIC ACIDS	Metal surfaces. Circulating pipes.	Dip-bath. Keep acid at a concentration of 1 to 2 Normal. 1 hr. on weathered surfaces. Pipe systems, 2 to 4 hours. Neutralize & rinse.	Corrosive action on the metal	Needs ventilation. Possible excessive corrosion.
ACID MIX-TURE	Nonporous	Mix 0.1 gal. HCl and 0.2 lb. sodium acetate to 1 gal H_2O	Can reduce contam. by 90% in 1 hour on unweathered surfaces.	May require prolonged treatment if weathered.
CAUSTICS (Lye, TSP)	Painted surfaces.	Apply & leave until paint softens. Wash & scrape.	Minimum contact with surfaces.	Can cause skin burns. Slow acting. Don't use on aluminum or magnesium.
ABRASION	All surfaces.	Wet sand blasting, scabbling and scarifying are practical. Collect used, contaminated abrasive and surface layers.	Can decontaminate to any desired low level.	Penetrates porous surfaces. Contamination spread unless inside closed system. Abrasion equipment contaminated.

Fig. 16 - Surface decontamination methods in order of increasing strength

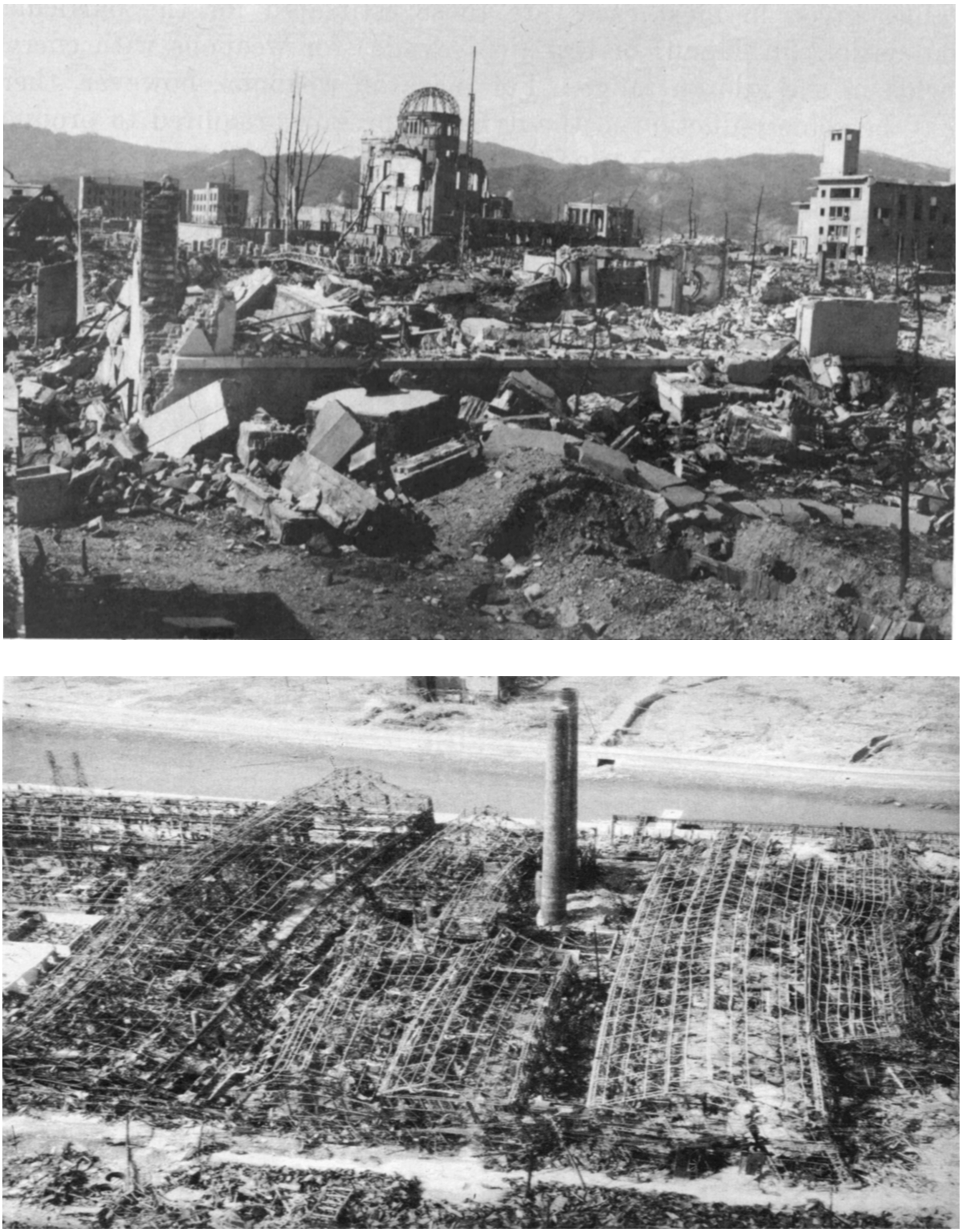


Fig. 17 - Effects of the Japanese atomic bombings

A nuclear **Blast Wave** carries high speed debris:

A one kT weapon	50% survival at 425 feet
A ten kT weapon	50% survival at 2000 feet

The fireball produces fatal **Thermal Burns** at large distances for a few seconds:

A one kT weapon	50% survival at 2000 feet
A ten kT weapon	50% survival at 1 mile

The nuclear fission releases initial **Ionizing Radiation** over a one minute burst:

A one kT weapon	50% survival at ½ mile
A ten kT weapon	50% survival at ¾ mile

For about 24 hours post-explosion, **Fallout** of visible particulate debris:

A one kT weapon	50% survival at 3½ miles
A ten kT weapon	50% survival at 6 miles

Fig. 18 - Atom bomb survival chances for low yield weapons

In a 1997 interview, former Russian National Security Advisor General Alexandr Lebed discussed the 132 suitcase nuclear weapons built for the KGB in the 1970s. He stated "We do not know what the status of the other devices is, we just could not locate them..." when referring to the 48 that had turned up in an inventory. They lost 84 of them! Each "suitcase nuke" measured 24" by 16" by 8" and could be

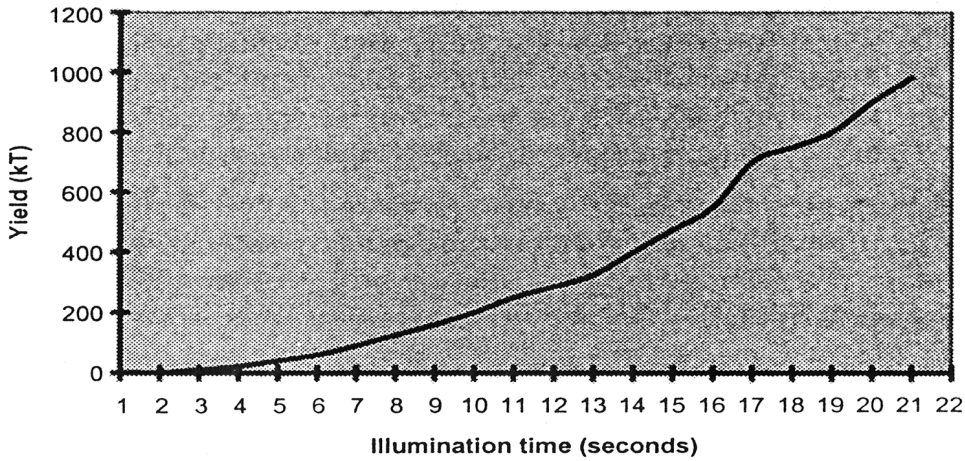


Fig. 19 - Nuclear weapon fireball flash duration vs. weapon yield

activated by a single individual in about 30 minutes. The design yields were from 1 to 10 kilotons.

Regarding the second scenario (building an IND), as discussed in Chapter 8, the operating principle is quite simple (recall the "Little Boy" device used on Japan). Terrorists must assemble two subcritical chunks of fissionable material and then cause them to rapidly collide to produce a supercritical mass. Even if the device is quite inefficient, i.e., it "fizzles," it could have a yield in the range of 1 kiloton of TNT. Probably the most difficult task for a terrorist is acquiring sufficient fissionable material for an IND. As recorded above, there have been hundreds of cases in which smugglers have been apprehended with nuclear material, many of them involving fissionable material. A question to ponder - how many were NOT apprehended?

The next two situations fall under the radiological terrorism label. The third scenario, breaching the security of radioactive material at a licensed facility such as a nuclear power station, with subsequent atmospheric release, is "easier said than done." Particularly since September 11th, security at U.S. plants follows the defense-in-depth philosophy. Many layers would have to be compromised before release of significant quantities of reactor core material would take place. On the other hand, sabotage of the plant equipment, even without radionuclide emissions, would cause disruption of electrical generating capacity and could cost billions of dollars in repair/recertification charges and loss of revenue.

The fourth and final scenario is the RDD. This is probably the easiest to accomplish of the four, and, thus, the most likely to occur. Common radioactive materials and sources are available on the black market. Thousands of sites in the U.S. have rad waste stored temporarily on-site. Terrorists could likely break-in and steal rad waste or actual sources relatively easily, particularly from institutional licensees e.g., hospitals and schools. However, depending on the physical form (liquid vs. solid) it might take a lot of effort to disperse the material. Placing a stick of dynamite next to a sealed radioactive source in a doubly welded stainless steel capsule merely results in a high speed projectile! Even if the dispersal following detonation of an RDD is not great, the results can still be devastating in the eyes of the public. As the NCRP Report 138 on Nuclear Terrorism points out, the psychosocial effects may overwhelm the physical destruction. Radiation is perceived as toxic, it is invisible and an RDD is a deliberate act by people against other people. "Taken together, these features and perceptions make radiation a powerful stressor." Many persons near the event site will experience PTSD - Post Traumatic Stress Disorder. This diagnosis was found at much higher levels than expected in the aftermath of the Oklahoma City bombing. Natural disasters are easier to relate to than deliberate human acts of terrorism.

First Responder Operational Considerations

Early responders to nuclear terrorism incidents would probably include firefighters, EMTs, law enforcement, and HazMat teams. The appropriate actions will depend on the type of incident - detonation of a nuclear weapon (Scenarios 1 and 2 above) or dispersal of radioactivity (Scenarios 3 and 4 above). In this latter case, it would not likely be immediately evident that radioactive materials were involved. This discussion will begin with the case of a nuclear detonation.

Presumably, the characteristics of a nuclear explosion would be readily recognized by first responders. The U.S. Dept. of Health & Human Services offers recommendations for the Emergency or Early Phase response to a nuclear terrorism event:

1) Keep out of the ground zero area
2) Enter adjacent areas only for lifesaving actions
3) Be aware of the radiation field strength in your operations area
4) Use respiratory protection and disposable coveralls or a suit
5) On exiting, remove outer clothing and shower (e.g., in a fire hose)
6) For injured survivors, lifesaving first aid ALWAYS precedes decon
7) Wash vehicles down before permitting them to leave the scene
8) Don't eat, drink or smoke if radioactive dusts are in the area
9) Stay alert to symptoms of heat stress in yourself and fellow responders
10) If possible, preserve evidence for law enforcement personnel - you are working a crime scene!

For survivors of the initial blast wave, thermal pulse and ionizing radiation pulse, fallout becomes the next big problem. A characteristic of nuclear weapon detonations is the rapid reduction with time of the dose rate due to fallout. From 1 hour post explosion out to 6 months, for each factor of 7 in elapsed time, the dose rate is reduced by a factor of 10. This rule of thumb is called the "7:10 rule." See Sample Problem 3.

For the more mathematically inclined technologist, the "7:10 Rule" is algebraically equivalent to $t^{-1.2}$ where t is the number of hours since 1 hour post detonation.

Sample Problem 3

GIVEN:

One hour following detonation of an improvised nuclear device, IND, fallout levels at a downwind location produce a measured dose rate of 30 R/hr.

FIND:

What is the estimated dose rate 2 days later at this location?

SOLUTION:

The "7:10 Rule" is used. At 7 hours post detonation the dose rate would be 1/10 of the 1 hour reading. So 7 times longer than this is 49 hours when the rate is 1/10 X 1/10 or 0.01 times the initial rate. But two days = 48 hours which is close to the 49 just calculated. Therefore, the estimated dose rate would be about 0.01 X 30 = 0.3 R/hr or about 300 milliroentgens per hour.

Moving along to the case of an RDD or dirty bomb (previously discussed radiological terrorism Scenario 4), one of the early technical issues for responders is identification of the radioactive material that's been dispersed. This should then clarify whether the scene presents chiefly an external threat (high radiation fields) or mostly an internal threat (lots of contaminated material likely to become airborne or ingested).

These days, most first responder units in urban areas have at least some radiation detection equipment readily available. A RADIATION DETECTOR SHOULD BE ACTIVATED WHEN RESPONDING TO ALL EXPLOSION INCIDENTS. An alarming dosimeter meeting the Homeland Security criteria (ANSI N42-32) would give an early indication that GAMMA EMITTING radioactive material was involved. If pure alpha and/or beta emitters were dispersed, the alarming dosimeter would be useless. A contamination monitor would be needed instead, e.g., a thin window GM tube or pancake probe. If neither type of detector measures above background, then radioactive material is probably not involved. See the section on monitoring instruments below for more information on the special considerations of first responder survey meters.

However, if substantial activities of radioactives are in fact dispersed, the ANSI qualified alarming dosimeter or pancake GM will be too sensitive to quantify the hazard. The alarming dosimeters are tested with a 0.05 to 0.1 mR/hour field (0.00005 to 0.0001 R/hr). Pancake or thin window GM probes will saturate at less than 1 million cpm which could easily be produced by a few microcuries of contamination. To determine if there is a significant external gamma radiation hazard at the scene, a high range, e.g., ionization chamber, instrument must be used.

Of the thousands of known radionuclides, only a small number are readily available in significant activity levels. The likely list of radionuclides for an RDD includes C-14, P-32, Co-60, Sr-90, Tc-99m, I-125, I-131, Cs-137, Ra-226, U-238, Pu-239, and Am-241. If huge activities are obtained and dispersed, lives are at risk. An analysis of the Goiânia Accident (discussed later in this Chapter) showed that the human fatality rate was about 0.003 deaths per Curie of the Cs-137 contamination that was widespread. If small activities are released by an RDD, the chance of fatalities is minimal. However, in either case, the psychosocial effects will be horrendous.

Once again, the U.S. Dept. of Health & Human Services offers recommendations for the Emergency or Early Phase response to an incident involving the deliberate release of radioactive material, i.e., a radiological terrorism event:

1) Approach from upwind of the scene

2) Be aware of other hazards such as fires, electrical lines, etc.

3) Be aware of the radiation field strength in your operations area

4) Use respiratory protection and disposable coveralls or suit

5) On exiting, remove outer clothing and shower (e.g., in a fire hose)

6) For injured survivors, lifesaving first aid ALWAYS precedes decon

7) Wash vehicles down before permitting them to leave the scene

8) Don't eat, drink or smoke if radioactive dusts are in the area

9) Stay alert to symptoms of heat stress in yourself and fellow responders

10) Cover radiation meters with clear plastic bags to reduce contamination risk to the meter

11) If possible, preserve evidence for law enforcement personnel - you are working a crime scene!

Once the incident scene is secured and search & rescue operations are complete, radiation protection technologists have a large role in the Recovery Phase. In the case of an RDD, this means large scale decontamination. Many industry-wide standard techniques from the D and D field discussed earlier in this Chapter are directly applicable. Since the contamination was only recently deposited, much of it can be washed and vacuumed up without needing vigorous abrasive processes such as scabbling or scarifying.

Dose Guidelines & PAGs for Emergency Operations

Doses received by commercial sector U.S. workers during emergency operations are subject to the same 10 CFR 20 standards as doses received during routine operations. There are no special dose limits that apply in emergencies. Other organizations have been less reluctant to make recommendations for occupational workers during accident operations. The Department of Energy Rad Con Manual does contain such guidelines which are to be used "in extremely rare cases." Doses over 25 rem are conditional upon "a voluntary basis to personnel fully aware of the risks involved." A 1993 report of the NCRP suggests that only lifesaving activities justify doses well over the annual limits. They further qualify their recommendation by saying that volunteer older workers with low lifetime doses should be used whenever possible. NCRP Publication 138 offers guidance on dose rates for first responders. They recommend a dose rate of 10 mR/hr as an initial alarm level. This level is several hundred times the natural background so there is no chance for a false positive indication. Also, at 10 mR/hr, it is unlikely that emergency responders would exceed the 5,000 millirem radiation worker U.S. standard during the course of the emergency phase of the incident. The NCRP further recommends 10 R/hr as the "turn around level" for emergency responders. Personnel should vacate the area at this level and regroup in a location with a lower ambient radiation field. Entry into fields over 10 R/hr should be restricted to lifesaving actions such as removal of injured victims. Fatalities should be left in place at this stage of operations.

ICRP Publication 60 finds that "Emergencies involving significant exposures of emergency teams are rare, so some relaxation of the controls for normal situations can be permitted in serious accidents...." ICRP Publication 63 on intervention in emergencies changed the numbers slightly. The numerical guidelines from these various sources are compared in Figure 20.

Doses received by the population during radiological emergencies are limited by Environmental Protection Agency developed Protective Action Guides (PAGs) in the United States. The PAGs are defined as "the projected absorbed dose to individuals in the general population which warrants protective action following a contaminating event." They are used by public health officials to make decisions during radiological emergencies. They are in the form of numerical guides which can be used to trigger

Organization	Property Protection	Lifesaving
U.S. NRC	No guidance	No guidance
U.S. DOE	10 rem	> 25 rem (TEDE), >250 rem (skin)
NCRP	5 rem	50 rem (TEDE), 500 rem (skin)
ICRP 60	50 rem (TEDE), 500 rem (skin)	>50 rem (TEDE), >500 rem (skin)
ICRP 63	100 rem (TEDE), 500 rem (skin)	>100 rem (TEDE), >500 rem (skin)

Fig. 20 - Comparison of emergency dose limit guidelines

actions if exceeded. These actions include 1) altering production or distribution practices of contaminated foodstuffs such as storing food and animal feed to allow for radioactive decay, 2) diverting contaminated products to nonhuman consumption uses and 3) condemning contaminated foods. One example of a PAG is the 10 rad value for iodine-131. It is based on a population average dose which in this case is composed of one-year-old infants. The 10 rad thyroid dose will be reached if the I-131 concentration in milk reaches 60 to 70 nanocuries per liter maximum. See Sample Problem 4.

Sample Problem 4

GIVEN:

Following a release from a "dirty bomb" the I-131 levels in milk rise to a maximum concentration of 1.4 Bq/ml and then fall slowly back to background levels.

FIND:

What is the estimated dose to the thyroids of infants in the affected area?

SOLUTION:

The conversion factor given in the text is 60 to 70 nCi/l milk = 10 rad thyroid dose. The measured concentration is 1.4 Bq/ml x 1000 ml/ l x 1 nCi/37 Bq = 38 nCi/l. Setting up a proportion, and using 65 nCi/l for the conversion factor gives 38 nCi/65 nCi = X rads/10 rads. Thus, X = 38 x 10/65 = 5.8 rads.

Monitoring Instruments for Nuclear Terrorist Incidents

This section on handling nuclear terrorism incidents will conclude with some comments on portable field radiation detectors with respect to the four Scenarios introduced at the beginning of the section. A number of the portable meters discussed at length in Chapter 7 on detectors and Chapter 12 on monitoring would be suitable for use during terrorist incidents. However, there are several differences between the

normal workplace of a radiation protection technologist and the nuclear terrorist incident scene in the Emergency Phase. The incident scene is likely to be outdoors and have fatalities and/or injured personnel lying about. It will be chaotic (not that you don't have occasional chaos at work!), and may be life threatening. Most of the survey meters used regularly by radiation protection technologists will be too sensitive - they will only read off-scale. Levels of ground contamination could be orders of magnitude above anything measured at your regular worksite. The meters need to be particularly rugged and particularly simple to read. Remember that while meter reading is second nature to radiation protection technologists, first responders will be encountering a real radiation hazard for the first time in their careers!

So, one of the requirements for first responder instruments is that they have really high ranges available. For high gamma radiation levels, ionization chamber instruments, e.g., the Eberline RO-20 discussed in Chapter 7, would be appropriate. Many first responders are being equipped with surplus Civil Defense meters. (Many of these are being furnished at no charge through the Homeland Defense Equipment Reuse program, HDER, a cooperative arrangement between the Health Physics Society and the U.S. Departments of Justice and Energy. See "Other Resources" section at the end of this Chapter.) The CD V-715 is very popular. The exposure rate ranges available are from 500 mR/hr up to 500 R/hr. Refurbished and recalibrated CD V-715s are available on the internet for around $250. This compares to about $1,300 for an RO-20. An even wider exposure rate range is covered by the Ludlum Model 2242. It has an effective range of 0.1 mR/hr up to 1,000 R/hr. It has passed the ruggedness requirements of ANSI N42.33 as recommended by the U.S. Department of Homeland Security. The unit actually uses two different geiger tubes to cover the wide exposure rate range and results in a unit which is a whole lot less fragile than an ion chamber type of instrument.

Alarming dosimeters meeting the Department of Homeland Security recommended ANSI N42.32 Standard are way too sensitive for significant nuclear terrorist incidents. As discussed earlier in this Chapter, they are designed and tested to indicate changes in ambient background of less than 1 millirem per hour. This is useful to determine whether an incident scene involves gamma emitting radioactivity but if significant levels are present, a high range meter is going to be needed promptly before rescue actions can be carried out.

There is one high range alarming device that is proving popular. This is the NukAlert™ Compact Key Chain 24/7 Radiation Monitor & Alarm. See Figure 21. The distributor claims ten continuous years of battery life. (It has no on/off switch!) It is calibrated to Cs-137. It indicates exposure rates from 0.1 R/hr to 50+ R/hr by the number of chirps (from 1 up to 10) in each group. At the bottom end of its range, it takes several minutes to begin chirping - at the high end, it responds in seconds. It is extremely rugged and will not be damaged by temperatures from -40° F to +185° F. The detector is a small piece of scintillating material attached to a CdS photocell. (There is an operator learning curve - it occasionally produces isolated chirps when moving from a colder location to a warmer one, e.g., exiting an air conditioned automobile, or if static electricity builds up on the case, e.g., rubbing against synthetic fabrics.) These handy little devices can be ordered on the internet at www.NukAlert.com.

Fig. 21 - NukAlert™ Radiation Monitor & Alarm

Finally, we'll conclude this section with a discussion of contamination meters. The first problem is that there is no ANSI Standard for contamination monitors for Homeland Security applications. It is understandable that pancake probes might not survive the gamma meter test criterion of sequential one meter drops on each of the six sides. Still, some test criteria for alpha/beta contamination monitors would be very useful to enable emergency responders to make informed purchasing decisions.

A second major problem with contamination meters is, to reiterate, the high sensitivity of commonly available commercial instruments. The ever popular pancake GM probe would quickly read off-scale in a significant RDD incident. Pacific Radiation technicians use a simple technique to gain a factor of 100 in the range of a 2" commercial pancake GM probe. A circular steel plate with a diameter equal to the probe, about 2½", is cut out of a 3/32" sheet. A hole with an area of 1% of the active area of the pancake tube is drilled through the center. (The correct drill size is 13/64" or 5 mm diameter.) This plate is then taped over the probe face while surveying. It filters out 99% of the alphas and betas that normally would be detected. **The cpm scale of the meter must now be multiplied by 100 to account for 1% transmission**.

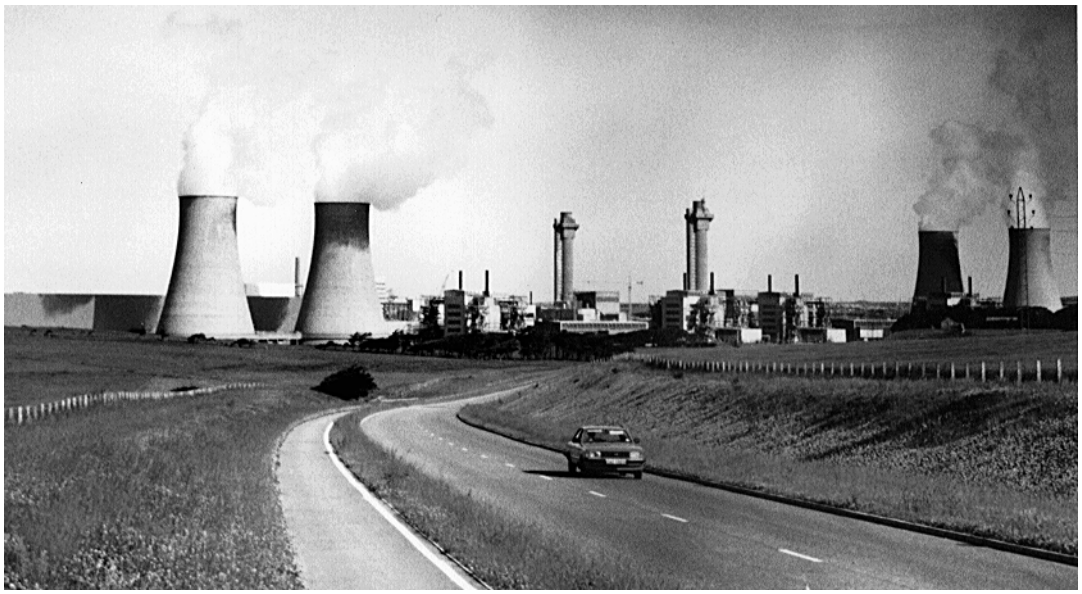
Review of Past Accidents

In addition to some accidents already covered earlier in this chapter, there are some "classic" accidents that ought to be familiar to a practicing radiation protection technologist. Several such accidents will be briefly discussed here.

Windscale Reactor Core Fire

DATE: October 1957 **LOCATION: England**
TYPE: Power Reactor **FATALITIES: 0**

This accident occurred at the Sellafield site (Figure 22) on the northwest English coastline near the town of Windscale. The graphite-moderated, air-cooled, natural uranium core reactor was being used for plutonium production. In the process of annealing the graphite moderator, the core temperature rose to a point where the uranium ignited. The fire



Gollnick

Fig. 22 - The Sellafield site on the English coastline

burned undetected for four days. The problem was discovered when an air sample taken a half-mile downwind from the reactor showed a high reading. The accident released 20,000 Ci of I-131, 600 Ci of Cs-137, 80 Ci of Sr-89 and 9 Ci of Sr-90 into the countryside.

I-131 levels in milk exceeded present action guidelines for distances up to 200 miles downwind of the site. The accident involved national level coordination to handle. The information relating iodine levels in milk to thyroid dose is still used to set protection standards for radioiodine releases.

Following the accident, the reactor core was disassembled and removed from the building. After decontamination, the building was converted to office space. The reactor stack is currently in use as a meteorological tower (see Figure 23). The radioactivity in the countryside was ploughed under and it subsequently decayed to an acceptable level. The photograph in Figure 24 shows the land just immediately downwind of the Windscale reactor in a 1985 photo. The area shows no signs of having been the site of a major nuclear reactor accident and dispels the newspaper prediction at the time that the area would be uninhabitable for 100 years.

Gollnick

Fig. 23 - The Windscale reactor building at present

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Fig. 24 - The current use of land adjacent to the Windscale reactor

The SL-1 Reactor Explosion

DATE: January 3, 1961 **LOCATION: Idaho**
TYPE: Power Reactor **FATALITIES: 3**

This incident occurred during the early morning hours. The reactor (designated Stationary Lowpower-1) was designed to provide electric power at remote locations. (A similar model was used for many years at a U.S. base in Antarctica). This particular reactor was located at a national laboratory in Idaho for training purposes. Three military technicians were performing routine maintenance. A technician manually pulled out the main control rod, causing a criticality burst and steam pulse which tore the reactor vessel loose from its foundation and lifted it several feet into the air.

The first outside indication of a problem was a heat alarm which sounded at a nearby firehouse. The Chief measured 25 R/hr near the building and retreated. No sign of fire was seen externally. About 15 minutes later, a health physicist entered the building, measured over 500 R/hr and exited. Some 65 minutes after the accident, the 3 technicians were unaccounted for so 2 persons entered the control room and found two technicians, one of whom was alive. With additional help, the living technician was removed but he died shortly afterward. The third technician

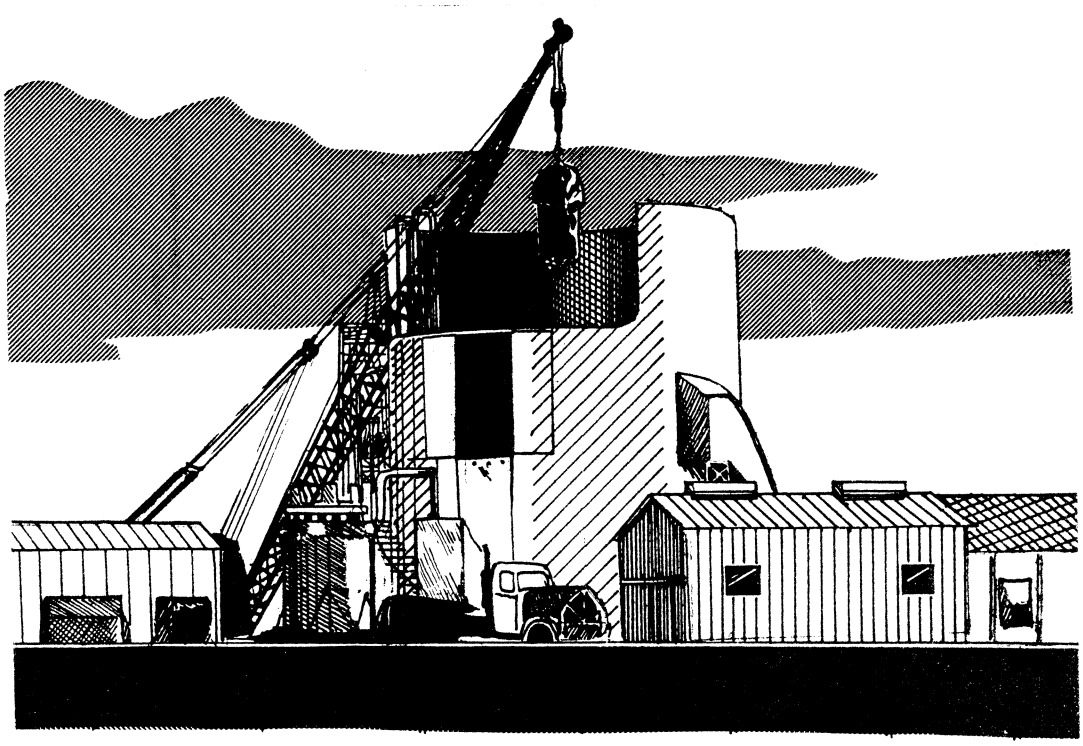

Fig. 25 - Core removal at the SL-1 accident site

was found several days later, pinned to the ceiling by a control rod cover that had ejected from the reactor.

Building levels reached 1,000 R/hr. It took 200 persons to handle the accident at a cost of $4.5 million, in 1961 dollars. Twenty-two persons received doses above 5 rem (maximum whole body dose = 27 rem). Although the reactor building was not designed as a containment vessel (it was primarily constructed of corrugated sheet metal) it was found to retain virtually all the activity released. Survey meters were replaced with higher range models on the fire trucks. A number of remote operations were devised "on the spot" to deal with the disassembly of the entire facility. See Figure 25.

Three Mile Island Accident

DATE: March 28, 1979 **LOCATION: Pennsylvania**
TYPE: Power Reactor **FATALITIES: 0**

Unit 2 of this facility was a 906 MW electric PWR nuclear generating station. Three months after the reactor began commercial operations, failure of a feedwater pump eventually caused a pressurizer relief valve to open, venting coolant into a relief tank. Instead of re-closing, as designed, this valve became stuck in the open position which reduced the water level in the pressure vessel, leading to automatic coolant injection. Due to malfunctioning pressurizer instruments, coolant injection was terminated at eleven minutes post-accident. The overflowing relief tank dumped activated coolant water on the floor of the containment building where it was eventually automatically pumped into the auxiliary building, causing massive contamination of both buildings. Finally, the water level in the pressure vessel dropped below the top of the fuel elements leading to fuel rupture. Shortly thereafter, high radiation levels measured in the coolant led to declaration of a Site Emergency. A General Emergency was declared thirty minutes later. The radiological consequences of the TMI accident are summarized in Figure 26. Within the 10 mile EPZ, the average dose to

1. Maximum dose to an individual outside plant = 0.080 rem
2. Total collective population dose equivalent
   = 2000 person-rem to 1.87 x 10^6 persons
   = 3 days of natural background radiation!
3. No detectable drinking water radioactivity
4. Milk samples had 14-40 pCi/l of I-131 (compared to 400 in this area following a 1976 Chinese atmospheric nuclear test)
5. Maximum off-site air sample activity = 24% of public MPC
6. Maximum thyroid dose to an individual = 0.050 rem

Fig. 26 - Radiological consequences of the TMI accident

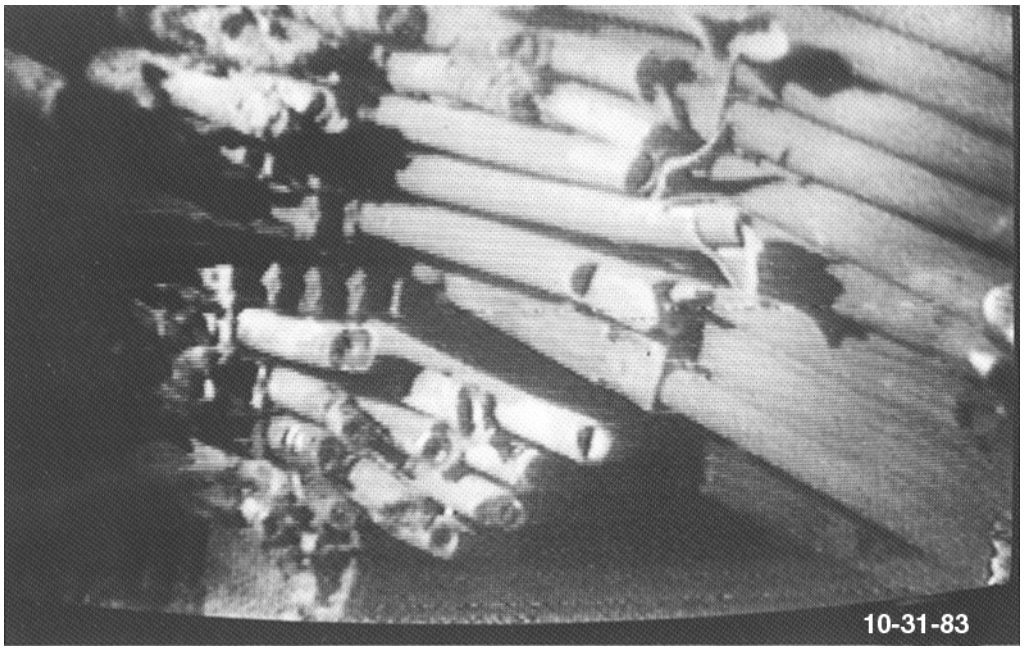

Fig. 27 - Underwater photo of damaged core of TMI Unit 2

Courtesy, GPU Nuclear

the population was 8 mrem – the equivalent of 29 days of background radiation. Although very little radioactivity was released during the emergency phase, a later, planned release of about 57,000 Ci of krypton-85 was carried out to reduce radiation levels inside the containment building to a point where human access was possible.

During most of 1986 and 1987, workers slowly disassembled the core of Unit 2. Figure 27 shows the damaged core internals as seen by an underwater camera. Larger sections were cut up and then lifted out through an 18 inch wide slot in the top of the vessel. Based on detailed analysis of the samples removed, approximately 70% of the core was damaged and 35-40% of the fuel melted. About 10 tons of melted fuel was fused together at the bottom of the pressure vessel. The basement of the reactor building was filled with about 1 million gallons of contaminated water. About 90% of the radioactivity attached itself to the concrete walls. The walls were slowly cleaned by two robots using high pressure hoses and chipping equipment.

The cleanup was completed in 1993. The process involved an average work force of 1,000 persons per year. It required 3.6 million person-hours to complete. Over 98% of the reactor core, some 150 tons of fuel and debris, were removed. About 235,000 cubic feet of low level waste was shipped to radioactive disposal sites. The collective dose for these operations totaled less than 6,500 person rem – significantly less than the official projections of 13,000 to 46,000 person-rem.

The next phase involved preparing Unit 2 for "post defueling monitored storage." About 2.3 million gallons of slightly contaminated water have been evaporated, unneeded plant systems have been drained and radioactive waste prepared for shipment. Unit 2 entered the monitored storage phase at the end of 1993. A plant staff continues to monitor the facility. Access is available to most of the plant without the use of protective clothing. The exceptions are the reactor building basement and a few locations in the auxiliary building. TMI-2 will remain in monitored storage until sometime later in the 21st century. Then, it will be decommissioned along with Unit 1. By that time, radioactive decay will have reduced ambient dose rates in the remaining contaminated areas to about half of their present level.

Chernobyl

DATE: April 26, 1986 LOCATION: Soviet Union
TYPE: Power Reactor FATALITIES: 31

Details of plant design and safety systems for the 4 Soviet RBMK-1000 power reactors installed at the Chernobyl site (along with 10 other reactors of this type at other locations in the former U.S.S.R.) are included in Chapter S-1. The station consists of four identical units. Figure 28 is a Ukrainian-made model of the site showing the four reactor units (center, right) and the cooling lake (lower left). Unit 4, the reactor involved in the accident, went operational in December 1983. It was a 1000 MW_e rated light water-cooled, graphite-moderated boiling water design. The plant was scheduled for shutdown on April 25, 1986 to enable significant repairs to be completed. About 3/4 of the fuel assemblies were from the original 1983 loading. The

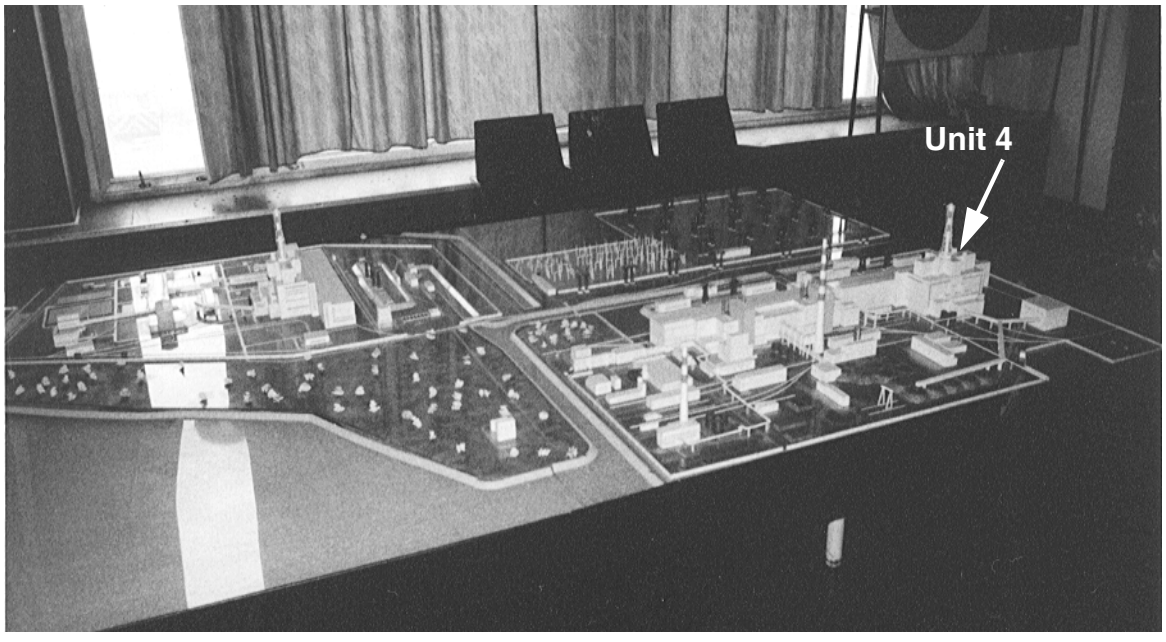

Fig. 28 - Chernobyl plant site model in the administration building

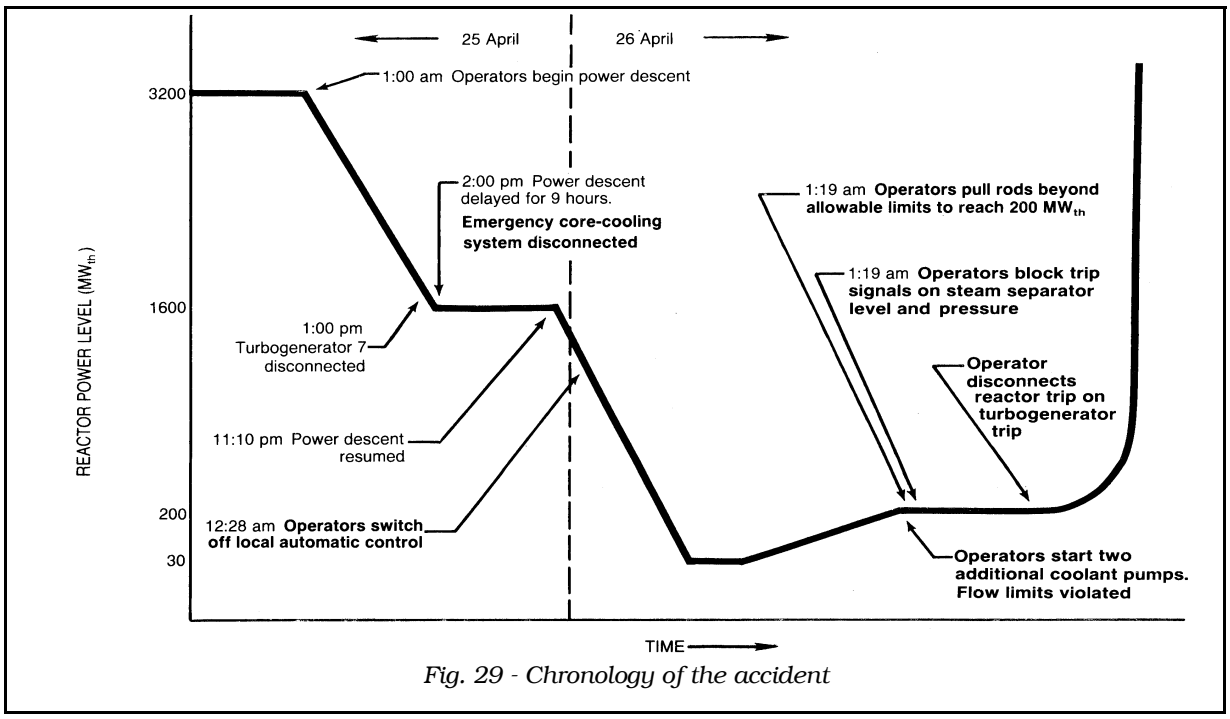

Fig. 29 - Chronology of the accident

remainder had been replaced during power operations, a special feature of the RBMK-1000. Figure 29 shows the reactor power level during the events to be described.

The operators began shut down at 1 A.M. on the 25th. By 1 P.M. later that day, the power level had dropped to 50% (1600 MW_t, down from an initial 3200 MW_t) and one of the two turbine generators was disconnected. The other turbine generator was to be tested to determine if, under a reactor trip (SCRAM), the rotational energy (inertia) in the generator was sufficient to power certain safety systems during the short time needed to turn on the emergency diesel electric generators and transfer the electrical load to them.

At 2 P.M., the emergency core-cooling system, ECCS, was disconnected to prevent it from activating during the generator test (a violation of operating procedures). At that point, the test was delayed nine hours to enable the plant to continue to supply the electric grid. At 11:10 P.M. the shutdown was resumed. One of the two control rod systems (see Chapter S-1) was then turned off. Unfortunately, the other control rod set had been mispositioned and this caused the power to drop to 30 MW_t – well below the level needed for the test. By 1 A.M. the next morning, the operators had managed to stabilize the power at 200 MW_t. But because of normal "xenon poisoning" the control rods had to be manually driven out beyond allowed safety limits. Although the 200 MW_t was below the 700 MW_t power level needed for the test, the operators decided to conduct the test anyway. For some reason, the operators then started up the 2 remaining cooling pumps, thus having all 8 pumps on line. But because the power level was so low, too much coolant was now being circulated, running the risk of pump breakdown due to cavitation and excess vibration. At 1:19 A.M., the automatic SCRAM systems for pressure and water level in the steam separators were disabled. Three minutes later (1:22 A.M.) the operators blocked the turbine trip circuits

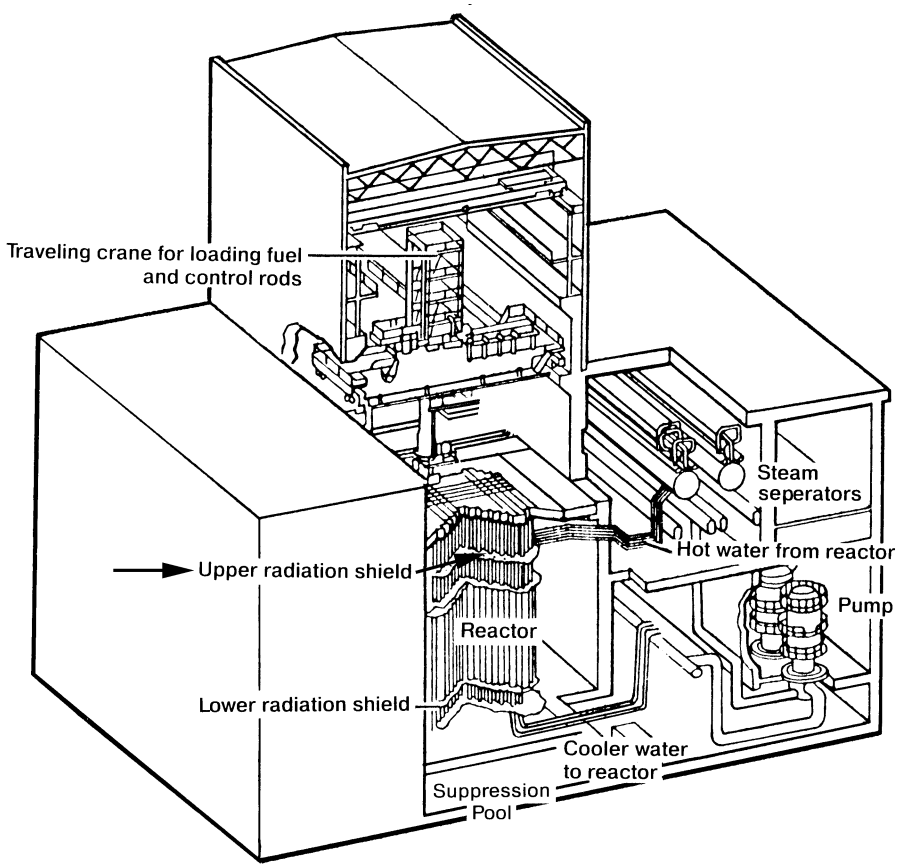

Fig. 30 - Location of the upper shield in the RBMK-1000

on both turbines (another violation). Half a minute later, the computer notified the operators that the excess reactivity remaining in the control rods was insufficient and required immediate shutdown of the reactor. The command was ignored.

The test was then conducted, beginning at 1:23:04 AM. Steam was removed from the turbine generator. The generator, along with the four cooling pumps it powered, began to coast down. As the coolant flow rate dropped, the water temperature rose producing more steam. Because of the unique configuration of the graphite moderator in the RBMK-1000, the core has a "positive void coefficient." This means that as more of the water volume is replaced by steam, the reactivity, and hence the power level, INCREASE! At 1:23:40 the shift manager ordered reactor SCRAM. Unfortunately, two other RBMK-1000 design features prevented a successful SCRAM. First is the slow rod insertion time of 20 seconds under SCRAM conditions. Second was the fact that the 1st meter of the 7 meter control rod length is an empty tip section. The simultaneous insertion of 200 empty sections caused a spike in reactivity. At 1:23:43 the high power surge melted the fuel and burst the cladding. Hot fuel fragments injected into the cooling water caused high pressure failure. The power surged to about 300,000 MW_t, 100 times the rated full power. This lifted the 1100 ton upper radiation shield up and turned it on its side (Figure 30) and then blew the reactor internals through the roof (Figure 31) ejecting burning graphite and fuel which

Courtesy, USSR Council of Ministers, Foreign Relations Dept., Moscow

Fig. 31 - Roof hole following explosion in Unit 4

started about 30 fires on the neighboring buildings.

The first outside firefighters arrived from Pripyat (a nearby city of 45,000) at 2:54 A.M. (Figure 32). Fire teams were highly trained military hazardous duty units. They concentrated on saving Unit 3 (attached to Unit 4 through the turbine building). This was accomplished within 40 minutes. Another hour and a half allowed the fires in all remaining buildings to be put out. Firefighters were working next to ejected fuel and graphite pieces radiating 20,000 R/hr.

The Unit 4 graphite core continued to burn out of control. Injected water failed to quench the fire. Between April 27 and May 2, helicopters dropped neutron absorbent and shielding through the torn open roof – Figure 33. They deposited 40 tons of boron carbide, 2600 tons of clay and sand and 2400 tons of lead. The dose rate at 110 meters altitude in the helicopters was 1,800 R/hr. The core fire was finally extinguished 12 days after the explosion by injection of liquid nitrogen into coolant passages remaining below the core. Two fatalities occurred on the first day. One worker was killed in the building collapse following the explosion and a second died of severe burns. Within 36 hours, 203 persons were hospitalized with acute radiation syndrome. They were categorized by severity. The highest dose range (400-1600 rem) had 21 fatalities out of 22 persons, all with severe skin burns over 60%-100% of their bodies. The next group received 400-600 rem and showed 7 deaths out of 23 patients, 6 of them with severe skin burns. The remaining fatality was in the 200-400 rem exposure group. All deaths were among plant personnel or outside firefighters.

The highest doses to the public were to people living on farms in the 3 to 15 km zone from the plant. Their 50 year dose commitment averaged 43 rem. Within one

Fig. 32 - Arriving Soviet fire fighters

Courtesy, USSR Council of Ministers, Foreign Relations Dept., Moscow

Fig. 33 - Shielding being air-dropped onto the reactor core

Courtesy, USSR Council of Ministers, Foreign Relations Dept., Moscow

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Fig. 34 - Former downtown of the village of Kopaci

day, 5,500 medical personnel were brought in by airlift to assist in the calamity. The citizens of Pripyat were evacuated, in only three hours, by 1100 school buses a day and a half after the explosion. The town is 2 km from the Unit 4 reactor, but the initial plume bypassed the community. At the time of evacuation, the dose rate had reached 1000 mR/hr. This group received an average external dose of 1.3 to 1.4 rem whole body and 10-20 rem to the skin. Internal doses were 10% to 15% of the external doses. The 90,000 children in the vicinity received thyroid doses from the dispersed I-131, I-132, I-133 and I-135. 10% of them had doses above 2 Gy. A total of 135,000 persons from 176 separate communities were evacuated out to 30 km from the plant. No radiation sickness was observed in this population. Farm animals within 18 km of Unit 4 were killed. Also, within this area, all soil in the top 1 meter layer was removed and buried. Figure 34 shows the remains of the village of Kopaci after it was bulldozed into a trench and covered with soil due to the high levels of contamination.

Estimates of the radionuclide releases to the environment have been made by many parties. The values given here are taken from a June 1987 U.S. Department of Energy report. 100% of the noble gases (Kr & Xe) were released, about 50 MCi. The volatile fission products of concern were chiefly ^{137}Cs and ^{131}I. The accident released about 2.4 MCi of Cs and 46 MCi of iodine. Another 3 MCi of remaining fission products were also dispersed. (For comparison, the total amount of Cs-137 released into the atmosphere by all nuclear bomb testing in the past was 26 MCi. Windscale released a total activity of 0.02 MCi.) The average dose commitment to persons in Europe and Asia as a result of these releases was 200 mrem, about two years of natural background radiation.

Courtesy, USSR Council of Ministers, Foreign Relations Dept., Moscow

Fig. 35 - Decontamination of streets by water truck

Investigations following the accident gave a rather complete picture of the events and their causes. Severe penalties have been assessed against the perpetrators by the courts. Persons interested in the "human side" of the tragedy are directed to the book **The Truth about Chernobyl** by the former deputy chief engineer for Chernobyl operations Grigori Medvedev. He points out the fact that top management and operations positions at the plant were filled under a patronage system rather than merit. Most top positions went to people with little or no nuclear plant experience. These persons increased the consequences in the early stages by continuing to insist that the reactor was intact in communications to Moscow. Medvedev also states that the turbine inertia test had been proposed to several other plants but they all rejected it as too risky.

Since the accident, the Soviets have applied vigorous efforts to reclamation of the site and surrounding area (See Figure 35.) By October of 1986, Unit 4 had been successfully encased in a concrete tomb or "sarcophagus." A massive tunnel was excavated from underneath the building foundation as shown in Figure 36.

Heat exchangers were installed and then the tunnels were back-filled with concrete. The roof was covered with a massive I-beam structure (Figure 37) which was then filled with concrete. The undamaged Units 1 and 2 were restarted and brought to full power in October and November 1986 (an amazing accomplishment in view of the SIX year delay in restarting Three Mile Island Unit 1). Unit 3 was placed in operation again during the summer of 1987. Surface decontamination of the area resulted in low radiation levels a year after the accident – 0.05 mR/hr in the town of Chernobyl, 0.4 mR/hr in the plant parking lot, and 0.06 mR/hr in offices and the control room

Courtesy, USSR Council of Ministers, Foreign Relations Dept., Moscow

Fig. 36 - Excavating the tunnel beneath the reactor building

on-site. Villages outside a 35 km radius of the plant were resettled by March 1987. The cleanup workers, termed "liquidators" by the government, worked from 1986 through 1991. Their average external dose was 12.5 rad. In 1986, the annual average was 17 rad per liquidator. The 1991 average dose to Chernobyl plant workers was 1.14 rem with 6 workers exceeding a 5 rem annual limit. The Dept. of Radiation Safety at the 4 unit site employed 240 persons on three shifts, comprised of 65 scientists and engineers and 175 technicians.

At the time of the author's site visit in 1992, no further decontamination was planned. Hot spots of 10 R/hr were still present within the 30 km zone but access was prohibited to members of the public. A control point on a main road at the 30 km boundary is shown in Figure 38. About 1500 elderly farmers had moved back into the zone and were tolerated by the government as long as they didn't attempt to take foodstuffs outside the zone. Radiation levels were at background (10 μR/hr) in Kiev, 80-100 μR/hr in Chernobyl town, 1,000 μR/hr in Pripyat, 1,000 to 2,000 μR/hr on the road in front of Unit 4, several R/hr adjacent to the sarcophagus (Figures 39 and 41) and 30 μR/hr inside the administration building on site (Figure 40). The town of Chernobyl (Figure 42) was used to house some workers on 15 day per month rotating shifts. The majority of the 5,000 plant workers were accommodated in the newly constructed town of Slutovich, 50 km from the plant, where they commuted daily on a special rail line.

A number of design changes had been implemented in all the RBMK-1000 plants as of February 1987. The number of control rods were increased, fuel enrichment has been increased to 2.4%, and a 10 times faster control rod shutdown system has been installed. Additional control rod position instrumentation and mechanical

Courtesy, USSR Council of Ministers, Foreign Relations Dept., Moscow

Fig. 37 - Support structure for concrete roof placed over damaged reactor

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Fig. 38 - Control point at 30 km zone boundary

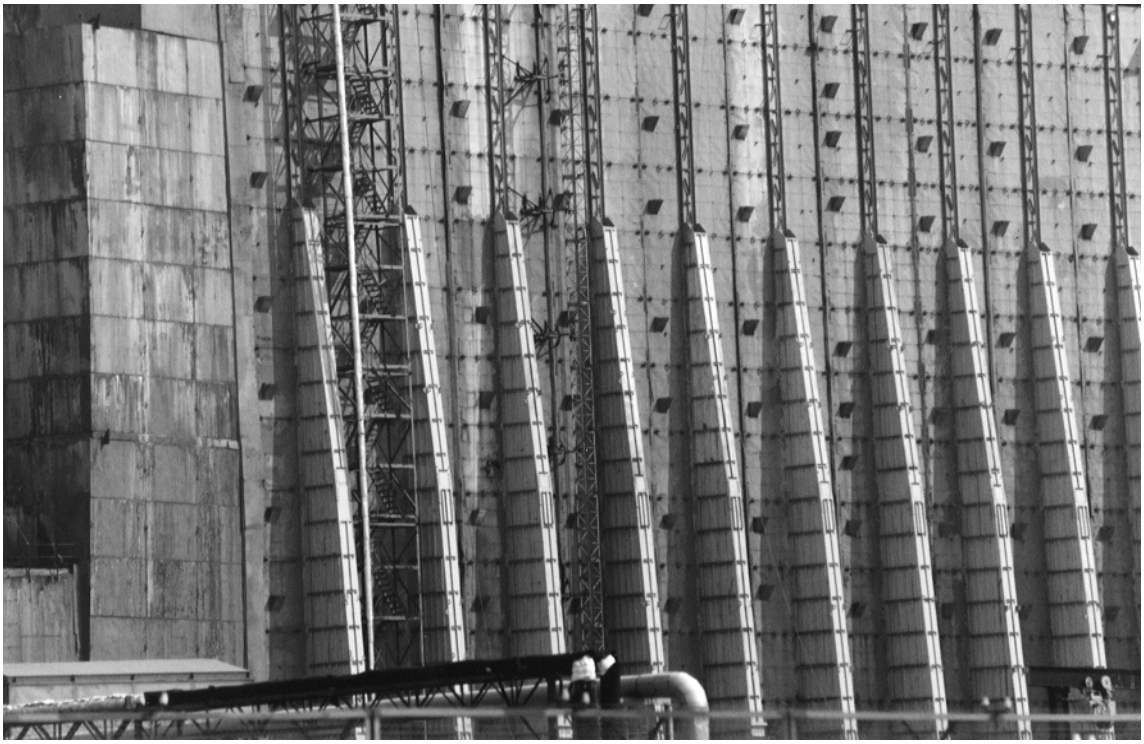

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Fig. 39 - Close-up view of the sarcophagus structure

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Fig. 40 - The Chernobyl plant administration building

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Fig. 41 - The sarcophagus covering Unit 4

Fig. 42 - Abandoned buildings in the town of Chernobyl

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stops to limit rod withdrawal have improved safety at the plants. All new reactors constructed will be of the VVER-1000 design, pressurized water reactors which are inherently easier to control. They will all have containment buildings as well.

During 1990 and 1991, the International Atomic Energy Agency, at the request of the Soviets, coordinated an international team of 200 experts from 25 countries to assess the radiological consequences and the health situation in the 25,000 km^2 contaminated area populated by some 825,000 residents. Since the short-lived isotopes had already decayed, the study, called **The International Chernobyl Project** focused on Cs-137, Sr-90 and Pu-239. Project scientists re-measured contamination levels and compared results to the Soviet data. The results for cesium and plutonium agreed while the Project values for strontium were lower than Soviet results. No appreciable contamination was found in drinking water and food from contaminated areas. The population dose estimates from the study were about only one-third to one-half the Soviet estimates. The Project found "no health disorders that could be attributed directly to radiation exposure." In terms of long-term effects, the Project had difficulty in predicting future cancers and genetic effects due to the relatively low doses received and due to the general lack of statistically valid data on these conditions in the population involved before the accident. The Soviets estimate an additional 300 cases of thyroid cancer in exposed children and 100 cases in adults over the next 70 years. The Project scientists conclude that these small increases in rate will be undetectable in the normal fluctuations of population cancer rates.

One attempt by the former Soviets to get a handle on the health effects is the Chernobyl Registry. Health and radiation dose data on 600,000 exposed persons is

being examined for long-term effects. The registry includes the liquidators, many of whom came from long distances to volunteer for cleanup activities (induced by high pay and pension benefits), as well as residents from the contaminated areas. As of 1992, the dosimetry had been completed on 25% of registrants. The doses to the rest were being computed using dose reconstruction models. Only 1.3% of the registrants had doses falling in the highest 50 rad category. This long-term study should be valuable in answering future questions about this population and in predicting consequences in the event of similar accidents.

Another study, by the U.S. National Cancer Institute, was published in 2003. It involved 25,000 subjects from Belarus and from Ukraine and was aimed at estimating thyroid doses. The conclusion was that the median thyroid dose was about 30 rem for both Belarus and Ukraine. About 20% of the Ukrainian subjects and 30% of the Belarusian subjects had estimated doses over 100 rem.

The future of the Chernobyl power station is no longer in doubt. In October 1991, Unit 2 had a serious fire in its turbine building. This resulted in permanent shutdown of Unit 2. Shortly thereafter, the Ukraine Parliament voted to shut down all the Chernobyl units at the end of 1993. Although the plant had provided 20% of all electric power in the country, economic conditions were such that demand for electricity had dropped 30% as of August 1992 so Chernobyl's power was not being missed. Officials at the plant were hopeful that as conditions stabilized, demand would increase and two or three units could be recommissioned. Then, in a surprising move, the Ukraine Parliament voted (by a large majority) in October 1993 to overturn the shutdown order and also to remove the moratorium on future nuclear plant construction in the country. In 1995, Ukraine reached an agreement with western countries to close down the site by 2000. But as the new millennium began, only three of the four reactors were permanently shut down. Unit 3 was brought back online in December 1999 to provide power during the winter. As of 2005, all Chernobyl units have been permanently removed from service.

Regarding Unit 4, the temporary sarcophagus is showing signs of failing. The structure is shifting with time and numerous large cracks are visible. Site engineers are worried that a minor earthquake or severe storm could cause it's collapse. In 2000, the Group of Seven industrialized nations set up a Shelter Implementation Plan managed by the European Bank for Reconstruction and Development. In 2001 they approved the design of what is now called The New Safe Confinement. It is basically a hugh arch, a few feet taller than the Statue of Liberty, with a span of 853 feet! It will be movable. Construction will occur past the end of Unit 4 where radiation levels are relatively low. Upon completion, the arch will be pushed horizontally until it covers over Unit 4. Then the ends will be closed in. A request for construction bids went out in 2004. A French consortium was the winning bidder. The construction contract was finally signed in 2007. The hoped for completion date of the project is 2013.

Goiânia Cs-137 Dispersal

DATE: September 18, 1987 **LOCATION: Brazil**
TYPE: Ruptured Therapy Source **FATALITIES: 4**

On September 13, two persons located and removed a stainless steel cylinder from a cancer therapy machine in an abandoned medical clinic in the town of Goiânia, 1,000 km northwest of São Paulo. They sold it 5 days later to a junk dealer who, 3 days later, had the lead shielding removed for resale. In the process, the inner platinum capsule was broken open, releasing 1,400 Ci of ^{137}Cs in luminescent powder form. The glowing powder attracted a crowd which took some of it to their homes. The six-year-old daughter of the junk dealer applied the powder to her body and also ate some on a sandwich. The dealer's wife slept in clothes covered with the powder. Both the daughter and wife, along with the junkyard worker who opened the source capsule have died. A total of 244 persons were found to be contaminated, 54 serious enough for hospitalization. The daughter received around 2000 rem. The 20 highest exposed survivors received 100 to 800 rem. All were internally contaminated and 19 had radiation-caused skin burns. The victims were treated with Prussian Blue, an iron compound that binds with cesium to allow excretion. Eight victims also received treatment with granulocyte macrophage colony stimulating factor (GM-CSF). Indictments for criminal negligence have been issued against the medical clinic owners who had moved to new quarters and left the therapy machine unattended for two years.

After nearly two decades without a criticality accident anywhere in the world, experts felt that this condition would prevail indefinitely. Unfortunately, not one, but two such accidents occurred, in May and June of 1997, in Russia. The Tokaimura accident followed about two years later.

In June of 1997, a Russian scientist working in their Arzamas-16 facility received a lethal radiation dose when his hands slipped while he was manually manipulating metal pieces of HEU. He showed signs of acute radiation syndrome within a half-hour and died about 64 hours later. His whole body dose estimate was 5,000 rem while his hands received an estimated 15,000 rem.

Tokaimura Criticality Accident

DATE: September 30, 1999 LOCATION: Japan
TYPE: Uranium Processing Criticality FATALITIES: 1

A 1999 study of the 21 criticality accidents between 1953 and 1997 showed that human error was the main factor in each. Also, 20 out of 21 occurred in the liquid state and in all cases the radiological consequences were contained within the accident building. The Japanese accident followed this pattern.

Japan's first criticality accident took place in the village of Tokai at the JCO Company, Ltd. nuclear fuel processing facility. Uranium hexafluoride gas, with the high enrichment of 18.8% U-235 was being converted to solid uranium dioxide for manufacture of fuel rods for a Japanese experimental fast breeder reactor. (Recall that commercial nuclear power plants use fuel with a typical enrichment of 2.5%.) Three workers were supposed to pour the liquid uranium solution into a tall, 20 cm diameter storage tank where it would be slowly pumped into the precipitation tank. In violation of written procedures, and in the interest of finishing the batch before the next shift arrived, they poured seven buckets of the uranium solution directly into a

45 cm diameter precipitation tank. This tank had a mechanical stirrer which would speed up the mixing. Unknown to the workers, (although this should have been stressed in their training) the larger diameter precipitation tank was unsuitable for two reasons - the larger diameter tank had a lower surface to volume ratio, allowing less neutron leakage than the tall narrow storage tank. Also, the precipitation tank had a cooling water jacket that reflected neutrons back into the solution. The criticality mass safety limit on enriched uranium in the precipitation tank, unbeknownst to the workers, was the equivalent of one bucket. The seventh bucket caused the tank to go critical.

All three workers reported seeing a blue-white flash. As a result of the cooling and moderation provided by the tank water jacket, the tank remained critical for 20 hours! Calculations indicate a total of 2.5×10^{18} fissions occurring. The estimated thermal power level was between 5 and 30 kilowatts, equivalent to a small research reactor. To terminate the reaction, a worker smashed a tank supply pipe with a hammer and forced the solution out onto the ground with high pressure argon gas. They also added boric acid to the tank with a fire hose.

The initial dose estimates on the three workers were 17, 10 and 3 sieverts. (Recall that the maximum survivable dose is 10 Sv.) The highest exposed individual, "Mr. A," was hugging the tank at the time of initial criticality. The second highest was "Mr. B" who was on a ladder over the tank.

The medical progression of the acute radiation syndrome followed the classic scenario closely. Mr. A's white cell count fell to "0" in 3 days. Two bone marrow transplants were performed in the first week. Hair loss was seen at 2 weeks. He began undergoing daily blood transfusions. After 1 month, liver and kidney damage were apparent. He was losing massive amounts of fluid through open skin radiation burns. Intestinal damage showed up at about 7 weeks post-accident. Attempts to transplant artificial skin were somewhat successful. At 2 months, he suffered cardiac arrest, but his heart was restarted. He was then placed on continuous hemodialysis. At 9 weeks, the transfusions became continuous and he experienced massive intestinal bleeding. Mr. A succumbed to acute radiation syndrome at 12 weeks.

Mr. B appears to have survived the accident as of April 2000. After 2 weeks he required platelet transfusions and he experienced severe throat, mouth, finger and foot pain. At 3 weeks he received transplanted umbilical cord blood cells. This apparently helped his bone marrow to recover more rapidly. At 4 weeks he suffered facial swelling, numb fingers and soles, and blisters spread down his hands and wrists. At 8 weeks, he showed severe intestinal damage. His intestines eventually recovered. Around 11 weeks, skin grafts were successfully accomplished. Four months post-accident, Mr. B's skin grafts had taken, he was in good spirits with stable bone marrow function and was beginning therapeutic sitting and standing.

There were a total of 119 persons who received radiation doses over 1 mSv from the Tokaimura accident. A grand total of 439 people received significant doses including 207 members of the public and 148 workers on-site at the time of the criticality. The Japanese government evacuated 161 persons from a 350 meter radius and advised about 300,000 nearby residents to take shelter in their homes for up to 18 hours.

Fukushima Daiichi Reactors Accident

DATE: March 11, 2011 LOCATION: Okuma, Japan
TYPE: BWR Power Reactors FATALITIES: 0

The Fukushima Daiichi nuclear power station is located on the northeast coast of Japan. The site currently contains six reactor units, all General Electric BWRs operated by Tokyo Electric Power Co, TEPCO. Electrical power ratings of the units ranged from 460 MW to 1100 MW with a total site capacity of 4.7 GW_e. Two Advanced BWRs are planned to be added in the future. Unit 3 had been operating with MOX fuel since September 2010. Existing Units 1 - 5 had the Mark I containment and Unit 6 used the Mark II containment. A schematic of the Mark I design is shown in Figure 43. In the case of a loss of coolant accident, LOCA, hot steam should enter the drywell volume. It would then vent through the large pipes at the bottom of the drywell into the water contained in a circular torus where it would be quenched. This is designed to reduce pressure inside the containment. The design has had a number of critics over the years. It is more cost-effective than the large concrete containment buildings

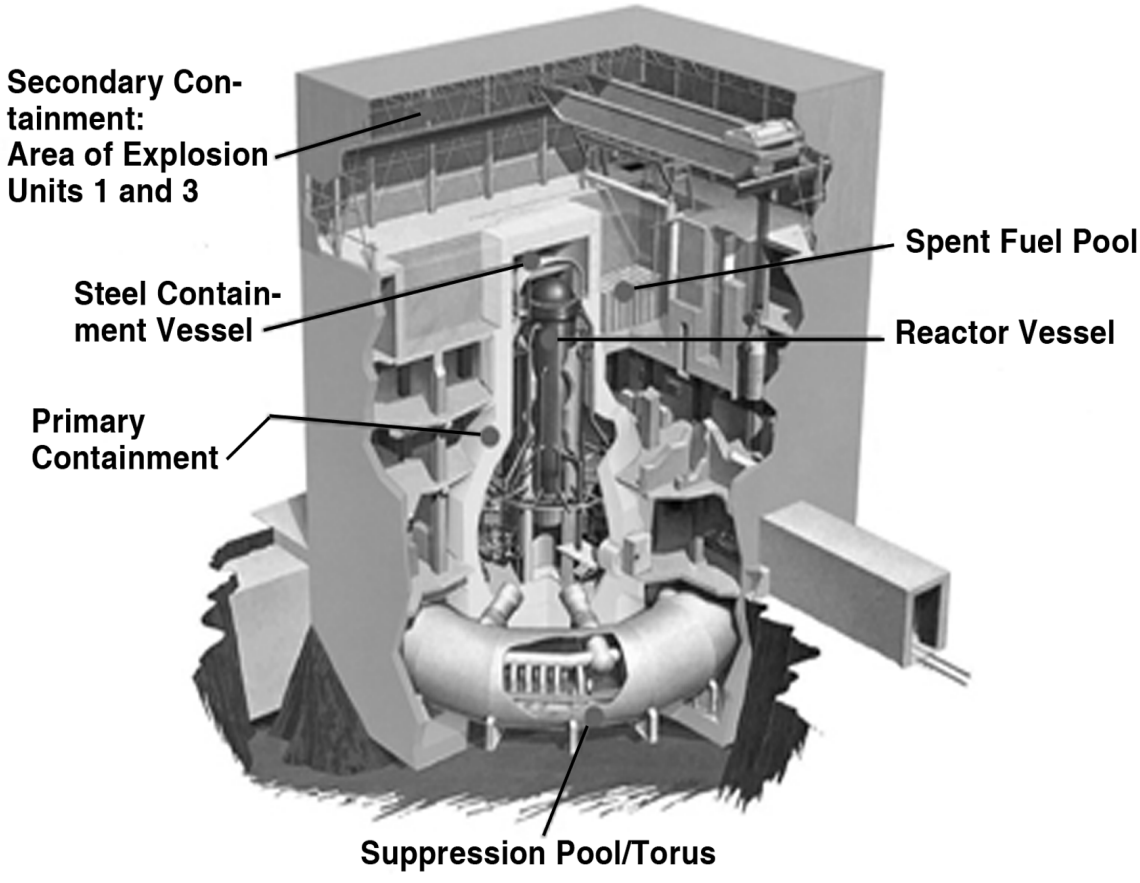


Fig. 43 - The General Electric Mark I containment structure

placed over PWRs but it cannot withstand the high pressure designed into conventional PWR containments. Thus, atmospheric releases of radioactivity are more likely with the Mark I design following a LOCA accident.

On March 11, 2011, an earthquake registering 9.0 on the Richter Scale occurred just off the Japanese coast. About 15 minutes later, a tsunami damaged the emergency backup generators at the Fukushima Daiichi plant. These generators powered the cooling water pumps for both the reactor vessels and the spent fuel pools. The earthquake and a subsequent tsunami also eliminated off-site power sources (the Japanese electrical grid) which normally backed up the on-site backup generators. At the time of the earthquake, Units 4, 5 and 6 were shutdown for scheduled maintenance. The entire core inventories of all three of these had been transferred to the respective spent fuel pools located inside the reactor buildings on an upper level. A chronology of events follows.

About 6 hours after the earthquake, the Japanese government issued the first of a series of evacuation orders to the population near the plant site. Persons living within 3 km of the plant were ordered to leave the area.

Around 4 AM the next morning (Day 2), the emergency batteries that had been powering the Emergency Core Cooling System were depleted. The water level in the Unit 3 vessel fell enough so that the tops of some fuel elements were above the waterline. Water levels in Units 1 and 2 soon followed the same scenario. Thus, high temperature steam was produced inside the reactor vessel. It reacted with the zirconium cladding on the fuel elements to release hydrogen gas. The hydrogen gradually built up in the tops of the reactor buildings. It also caused an unwanted pressure buildup inside the three reactor vessels.

At 5:30 A.M on Day 2, steam was released from the Unit 1 vessel to reduce the internal pressure. It contained some radioactive material, chiefly tritium and nitrogen-16. At 11:00 AM steam was also released from the Unit 2 reactor vessel. The evacuation order was amended to include residents within 10 km.

Mid afternoon, Day 2, a huge hydrogen explosion blew off the top third of reactor building 1. Four employees were injured by the blast. However, containment was maintained by the reactor vessel. The evacuation zone was extended to 20 km.

Early in the morning of Day 3, Unit 1 was declared a level 4 accident on the International Nuclear and Radiological Event Scale, (INES). This logarithmic classification system for accidents was established by the International Atomic Energy Agency. There are 7 levels where each level represents an estimated 10-fold increase in consequences from the previous level. Levels 1 through 3 are for “incidents” and 4 through 7 are used for the more severe “accident” category. INES Level 4 is considered an “Accident With Local Consequences.”

Late morning on Day 4, Unit 3 suffered a hydrogen explosion that destroyed the outer building, but, as was the case in Unit 1, the reactor vessel remained unbreached. TEPCO began pumping seawater and boric acid into the Unit 2 reactor vessel in an attempt to keep fuel rods under water and decrease the possibility of criticality.

Day 5 began with a fire in the spent fuel storage pool in Unit 4. About 15 minutes later, the Unit 2 building exploded. Most of the on-site workers were temporarily evacuated. About 3 hours later, the Unit 4 building exploded along with an accompanying fire. The site boundary radiation level spiked again (see Figure 44). A few hours

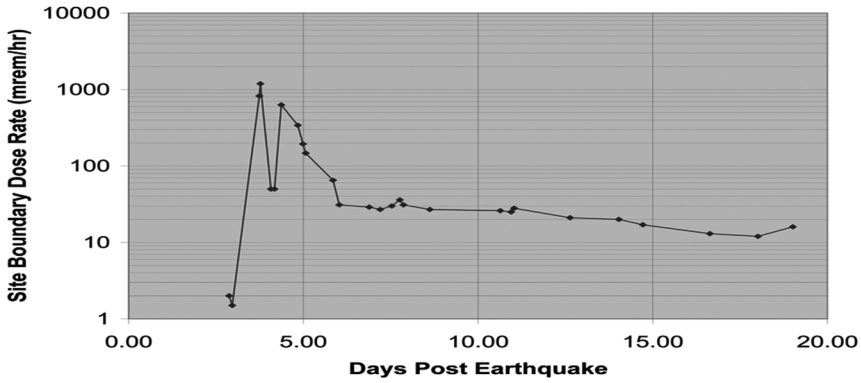


Fig. 44 - Fukushima Daiichi site boundary radiation levels

later, a "no fly zone" was established within 30 km of the plant site. The fire in Unit 4 was finally extinguished around 5 PM.

A photo of the site taken early on Day 6 is shown in Figure 45. Smoke and steam can be seen coming from Units 2 and 3 (located behind the two long white turbine generator buildings). Day 6 started with a report of fire, again, in Unit 4. The location was about the same as the fire the day before. It took around three hours to put out the new fire. Later that morning, rising radiation levels forced the evacuation of the remaining 50 nuclear workers at the plant. One hour later, they were able to

Fig. 45 - The Fukushima Daiichi site on day 6 after four hydrogen explosions

return and resumed emergency operations. Near midnight, an additional 130 nuclear workers were able to return to the site as radiation levels continued to fall. On Day 7, the Japanese Self Defense Force used Chinook twin engine medium lift helicopters to drop water on top of Unit 3. The aircraft were modified with lead sheeting on the floor. The operation was only partly successful due to wind dispersion of the water. By late afternoon, backup power from the local grid was brought to Unit 2. Water spraying from ground based fire trucks began at Unit 3.

On Day 8, water drops and spraying at Unit 3 continued from both the air and from the ground. Late in the day, the IAEA raised the INES classification to Level 5, "Accident With Wider Consequences," for Units 1, 2 and 3. Unit 4 was assigned to Level 3, "Serious Incident." Radioactivity was found in food products on Day 9. Milk and spinach obtained near the plant had levels above legal limits so food sales from the area were halted. To reduce the risk of more hydrogen explosions, holes were purposely cut in the roofs of Units 5 and 6 to ventilate the trapped gases. By Day 11, the availability of site emergency power enabled cooling systems for the spent fuel pools in Units 5 and 6 and for the Unit 5 containment vessel to be restored.

On Day 12, electrical power was restored to Units 1 through 4 so that the lights finally came back on in those control rooms! Seawater continued to be pumped into the containment vessels of Units 1 through 3 and fire trucks continued to spray water through the top of Unit 4. Three plant workers were overexposed when they stood in contaminated water on Day 14. They were working under Unit 3 to restore electrical power. Early dose estimates were in the range of 200 to 600 rem to the skin of their ankles. The next day, a voluntary evacuation zone was established within 30 km of the plant. On Day 19, radioiodine from the accident was detected in the United Kingdom.

Highly contaminated water used to cool Unit 2 was found to be leaking into the ocean on Day 22. Following several attempts, the leak was finally sealed after 4 days. On the 23rd day of the accident, the deaths of the first two workers at the site were confirmed when their bodies were found in the basement of Unit 4. They had died of injuries caused by the tsunami.

Water handling continued to be a major frustration. The pumping, spraying and water dropping activities all led to highly contaminated cooling water seeking the lowest levels - the reactor building basements and various below ground tunnels used for maintenance and to carry pipes and wiring between the units. By Day 24, TEPCO had filled up all available storage tanks on-site and so they began releasing the least contaminated water into the ocean. The idea was to make space available for more highly contaminated water being pumped out of the lower levels. By Day 39, some water cleanup facilities were in place. Contaminated water from the Unit 2 basement was the first to be processed. Having had some success, the water problem was attacked more vigorously. By Day 96, a large water treatment facility was in place. It was expected the facility would be able to remove radioactive contamination for a month before needing to shut down for filter replacements. The treatment facility lasted 5 hours! Apparently the inlet water was carrying much more activity than realized. Finally, by the end of June, water treatment was working well enough that 80% of the cooling water being supplied to the three reactors was recycled water. After another week, a key milestone was reached when the treated water accounted for 100% of the continuing cooling water needs.

As more and more information became available, it gradually became evident that core meltdown had occurred much sooner than expected. On Day 86, the Japanese agency responsible for industrial safety concluded that the pressure vessel in Unit 1 possibly released melted fuel down into the primary containment as early as 5 hours into the accident. The estimates for Units 2 and 3 were around 80 hours post earthquake. Finally, the estimated atmospheric releases of radioactivity were put at about 20 MCi in the first 6 days. (For comparison, the total activity released from Chernobyl was about 100 MCi.).

Four months into the accident, the work force at the Fukushima site numbered around 3,000 persons. Units 1, 2, and 3 were cooled to a reasonably stable level and all three containment vessels had been filled with nitrogen gas to prevent any further devastating hydrogen explosions. TEPCO stated that they were on track to achieve cold shutdown conditions in all reactors on the site by the end of the 2011 calendar year.

New reactor cooling systems had been installed and became operational in all four damaged units at the 5 month anniversary of the accident. As of 6 months post accident, the total release of radioactivity into the ocean up to that time was 405 kCi as calculated by the Japanese Atomic Energy Agency. This brings us up to presstime for the second printing of this edition (October 2011). Further updates will not be covered in the Sixth edition of this Text.

In conclusion, the Fukushima Daiichi Accident is now considered the worst reactor accident in history. Four separate reactors at the same site were involved almost simultaneously in facing a large earthquake, a massive tsunami and uncounted unique roadblocks. In spite of all the hurdles, the cooperative efforts of the international community, Japanese officials and workers offer hope that future activities will ultimately result in a safe nuclear site and allow the surrounding area to be re-occupied by the Japanese evacuees.

Contaminated Scrap Metal

Between 1983 and 1994, there were thirty-five incidents reported to the regulatory authorities in which radioactive material had found its way into scrap metal smelters worldwide. Twenty-two of these occurred in the USA. A number of these cases resulted in widespread contamination at the smelting facilities. Some of the radioactive material was subsequently discovered in commercial products made from contaminated scrap metal and distributed internationally. In addition to the actual smelting cases, U.S. records indicate another 293 cases over the same time frame in which radioactive sources or contamination was detected in scrap metal before it reached the smelter furnaces. The rate of reported instances appears to be on the rise, as shown by Figure 46. This is probably due to increased use of radiation detectors at scrapyards.

Studies indicate that about one-third of the radioisotopes found are NORM. Various byproduct materials (chiefly Co-60 and Cs-137) and Ra-226 account for about one-half. Radioactive cobalt is usually in metallic form while cesium is usually as a salt. Thus, when smelted at high temperatures, the cobalt ends up in the metal product while the cesium usually vaporizes and ends up in the furnace dust.The NRC has gathered data on the decontamination costs associated with several of these inci-

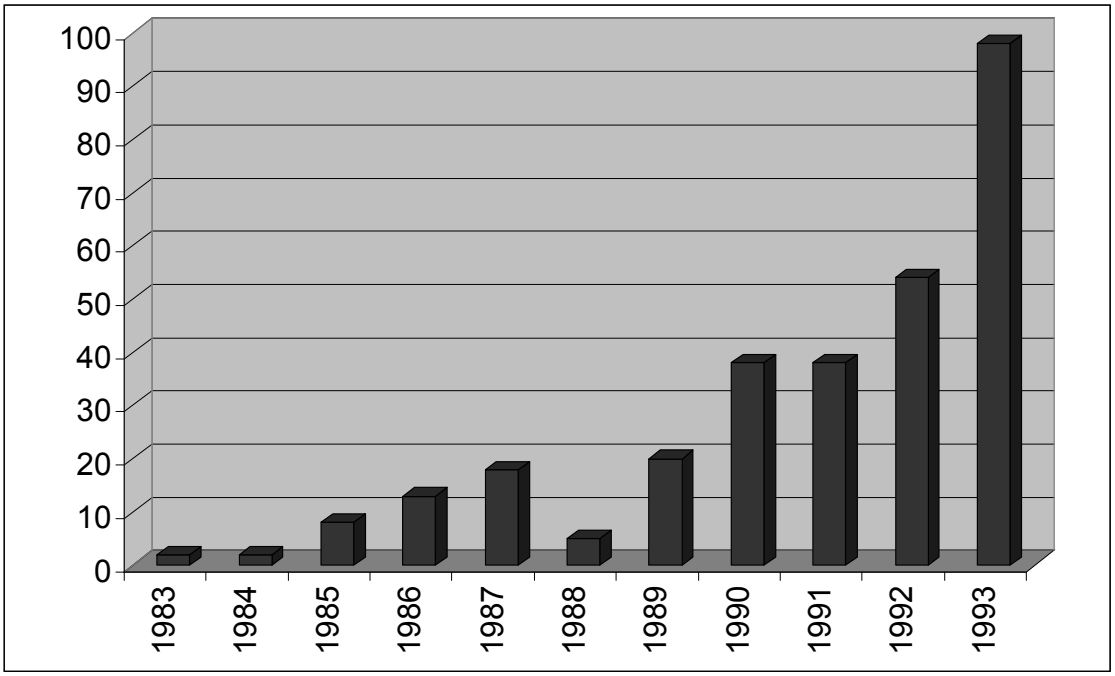


Fig. 46 - Reported cases of radioactivity in scrap metal at U.S. recyclers.

dents. They varied from a low of $450,000 to a high of several million dollars. The average furnace cleanup costs about $1,000,000 to $2,000,000.

Often, the mill must be shut down during decontamination efforts so down-time losses can easily double the decontamination and disposal costs. Thus, it is very cost-effective to monitor the scrap metal stream to intercept the radioactivity before it makes it to the smelter. A scrap monitoring system can greatly reduce the potential for a contaminating incident at a smelter. However, it is actually more effective at the recycling yard than the smelter. This is because the recyclers usually compact and bale the scrap metal before shipment. This increases the density of the bale making it more difficult to obtain an exterior gamma level reading above background. Figures 47 and 48 show some of the popular Bicron equipment used in the industry.

The Bicron ASM-6000-D is designed for monitoring both scrap-filled railroad cars and trucks moving past the detectors (four plastic scintillators with a surface area of 2880 square inches, capable of detecting Co-60, Cs-137, Ir-192, Ra-226, neutrons and Am-241). This system can reliably catch a 50 mCi cesium or cobalt source buried in #1 or #2 sheared steel scrap in railcars or trucks moving past at 5 m.p.h. The false alarm rate is about one per three months.

One of the earliest and most notorious instances of radioactive consumer products involved the Jonke Felix junkyard in Juarez, Mexico. A Co-60 teletherapy head with an activity of 1,000 Ci was shipped to Juarez in 1977 and placed in storage. In 1983 it was unknowingly sold as scrap metal to the Jonke Felix junkyard which cut it apart and sent the

Fig. 47 - Bicron scrap monitor hardware

Courtesy, Bicron Corporation

metal to a foundry. Then, through a fortuitous error, a radiation alarm was triggered at Los Alamos National Laboratory by a load of construction

Fig. 48 - A scrap metal monitor in use

Courtesy, Bicron Corporation

Fig. 49 - Contaminated table base and Falcon de Juarez foundry logo

rebar. It eventually turned out that 4,500 tons of rebar had been contaminated. 500 metal table bases (Figure 49) were also contaminated. The table bases had contact exposure rate readings up to 600 mR/hr and the rebar read up to 700 mR/hr. Four residents of Juarez were estimated to have received over 500 rem whole body doses based on chromosome aberration dosimetry done at Oak Ridge. Aerial surveys by the EG&G search team turned up loose cobalt-60 pellets over a four-state area. The junkyard sold the teletherapy head for $10. Cleanup costs were several million dollars.

A more recent case occurred in 1991. A smelter in India became contaminated with Co-60 during a run used to make chainlink fencing products (tension bars, gate bars and truss rods). The concentration of radioactivity was lower than in the Juarez accident but the total amount of affected steel was much higher. Figure 50 shows some of the tension bars collected from throughout the southwest U.S. prior to their return to India. Contact dose equivalent rates ranged from background up to 50 µSv/hr (5 mrem/hr). Over 21 tons were collected from west coast states.

Fig. 48 - Contaminated fencing products awaiting return to the country of origin, India

Gollnick

There are three ways that radioactivity has entered the scrap stream. The most obvious is the discarding of a steel radionuclide source capsule into scrap metal. In the U.S., the majority of nuclear gauge sources are distributed under a State or NRC "General License." Due to the inherent safety built into the device, they are basically unregulated after arrival at the final user. Approximately 150,000 of these generally licensed gauges are in use at present in the U.S. alone, with no requirements in place for periodic inspection and accountability. Another common route to the scrap metal stream is for pipes contaminated with NORM from the oil production industry to be chopped up and added to the scrap pile. Finally, it has been occasionally observed that metals with induced radioactivity from nuclear accelerator facilities have ended up as scrap. Hopefully, through the use of scrap monitoring equipment and through better accountability for radioactive sources, fewer incidents of this nature will occur in the future.

Public Relations and Legal Aspects of Nuclear Incidents

Public relations is often the biggest problem in handling a nuclear emergency. To the ordinary man in the street, radiation is "an invisible, silent, intangible force with magical capacity to produce harmful effects at great distances." After an incident, lack of information creates a psychological void which is filled by rumor and speculation. This can lead to growing anxiety. The antidote is information, even if it's

bad news. The licensee has the primary responsibility to release information to the media. Proper public relations actions can avert panic on the part of the public. Pre-education before an incident is an obvious principle, yet it often is not carried out by the "pro-nuke" side. Police, fire and public health officials need to be informed ahead of time concerning nuclear matters as the media often seek them out during an accident.

A public information plan should be worked out by each licensee in advance of an accident or incident. An Emergency Public Information Manager should be designated in the facility emergency plan. This person is responsible for releasing information in a timely manner to the public. THEY SHOULD BE THE SOLE POINT OF CONTACT WITH MEDIA PERSONNEL. They should attempt to describe the possible impact of the emergency on public health in general and on the licensee in particular. It is a good idea to have a backup person also named in the emergency plan to cover this important function in case the chief designee is unavailable.

When interacting with reporters, don't speculate about hypothetical situations. These hypothetical situations often make the evening news. If at all possible, coordinate all news releases through a single spokesperson. It is important to inform the media when some measure being taken is strictly precautionary. Use of experienced persons as media contacts can go a long way toward keeping a proper perspective.

Most accidents require the services of an attorney at some point in the overall management. Accidents can easily lead to claims of negligence by affected workers or members of the public. To legally demonstrate negligence, a worker acting as plaintiff must show that "a legal duty was owed to the worker" and that "the legal duty was breached." To receive compensation, the plaintiff must also have a demonstrable injury and that injury must be causally related.

At present, about half of the radiation injury claims pending in the judicial system involve radiation doses less than 500 mrem. In cases filed before the revised 10 CFR 20 went into effect, the "legal duty" owed to a worker was keeping the doses within the maximum limits specified. Under the new regulations, ALARA is mandated so the "legal duty" of the employer becomes ALARA.

In preparing for litigation, dose reconstruction is often of primary importance. In this context, it is best to make the most realistic assumptions rather than the most conservative. Radiation surveys are useful legal records but are of value only if the survey includes the name of the surveyor and instrument serial number used. Calibration logs must also be available or the survey is not legally admissible. In 1990 it cost an average of $300,000 to defend against a claim of radiation injury.

Problem Set

1. Which federal agency has the overall responsibility in managing a large off-site nuclear radiation accident in the U.S.?

2. What are the notification requirements for an accident in which a radiation worker receives a dose estimated to be 55 rem to the skin of one arm?

3. Which class of licensees accounted for the largest number of overexposure accidents in the U.S. in recent years?

4. Name some reasons why loss of an industrial gamma radiography source might have such serious consequences. How can the probability of such losses be reduced?

5. Why is the contaminating/undiscovered category of accident usually so much more difficult to manage than the other types?

6. How can a licensee's emergency plan be tested?

7. How big is the plume exposure pathway emergency planning zone? What is the name given to the larger EPZ?

8. Discuss the appropriate response for a radiation protection technologist who arrives at the scene of an external/short-term radiation accident just after it occurs.

9. Name some emergency screening procedures. What is each designed to detect?

10. Why is speed so essential in the management of persons suspected of having a significant internal uptake of radioactivity as a result of a radiation accident?

11. Briefly describe some methods of reducing the effects of an internal uptake of radioactivity.

12. What criteria do public health officials use in deciding whether to recommend the issuing of KI tablets to the general population?

13. How does the U.S. NRC's guidance on emergency lifesaving whole body doses compare to the U.S. DOE's? Are these consistent with international guidance?

14. Name some reasons why plutonium is so hazardous.

15. Describe some of the recommended procedures for decontaminating the skin of workers cleaning up a spill of radioactive material.

16. Using the information presented on the Goiânia accident on page 680 and the ingestion ALI values from Chapter 9, Figure 34, estimate the number of human fatalities that might be expected with widespread contamination following terrorist detonation of an RDD containing 1500 curies of cobalt-60 chloride. State your assumptions.

17. Choose one of the radiation accidents discussed in detail in the text. What was the cause of this accident? What steps can be taken to prevent this type of

accident in the future? What radiation protection lessons were learned as a result of handling this accident?

18. Calculate the number of grams of U-235 that fissioned in the Tokaimura Criticality Accident.

19. Calculate the maximum contact exposure rate at the side surface of a 10 foot wide railcar loaded with steel scrap with a 1 curie Cs-137 source at the geometric center of the car. The TVL for scrap metal (typical density of 65 lbs/ cubic foot) is 16".

20. Why does the smelting of a Co-60 source potentially have more public health risk than a Cs-137 source at a steel mill?

21. List three techniques that can be used to reduce the public relations impact of a nuclear emergency.

22. Do you personally meet the guidelines of the DOE Rad Con Manual for performing a lifesaving action in a radiation emergency?

S-1. What is the purpose of a "hot line?" Why should it have only a single "control point?"

S-2. List two reasons why ^{24}Na instead of ^{28}Al is the body activation radioisotope of choice for whole body counting following an acute neutron exposure.

S-3. List three specific causes of the Chernobyl accident.

S-4. Name some actions that might be taken during the restoration phase in an accident to assure that exposures would be ALARA.

S-5. What is the mission of the DOE NEST?

S-6. What is CSF therapy as applied to radiation accident victims?

S-7. Calculate the "exact" solution to Sample Problem 3 using the $t^{-1.2}$ equivalency of the "7:10 Rule."

Other Resources

1. "Report on the Accident at the Chernobyl Nuclear Power Station," NUREG-1250, U.S. NRC, Washington, DC, 1987.

2. "Coping with Radiation Accidents - Hospital and Community Planning," Michael Vince, Editor, Envirotox Management, Inc., Ravenna, Ohio, 1990.

3. “Developing Radiation Emergency Plans for Academic, Medical or Industrial Facilities,” NCRP Report 111, NCRP Publications, Bethesda, MD, 1991.

4. “Management of Persons Contaminated with Radionuclides,” NCRP Report No. 161, Bethesda, MD., 2008.

5. “The Goiânia Radiation Accident,” Special Issue, Health Physics, Volume 60, Number 1, Pergamon Press, New York, January 1991.

6. “Principles for Intervention for Protection of the Public in a Radiological Emergency,” ICRP Publication 63, Pergamon Press, New York, 1992.

7. “The International Chernobyl Project - An Overview,” International Advisory Committee, International Atomic Energy Agency, Vienna, 1992.

8. “Early Dose Assessment Following Severe Radiation Accidents,” R. Goans, et. al., Health Physics 72, pp. 513-518, 1997.

9. “Case Histories of Radiography Events,” NUREG/ BR-0001,Vol. 1, U.S. NRC, Washington, D.C., 1980.

10. “Management of Terrorist Events Involving Radioactive Material,” NCRP Report 138, NCRP Publications, Bethesda, MD, 2001.

11. “Responding to a Radiological or Nuclear Terrorism Incident: A Guide for Decision Makers,” NCRP Report 165, NCRP Publications, Bethesda, MD, 2010.

12. “The Four Faces of Nuclear Terrorism,” C. Ferguson & W. Potter, Monterey Institute of International Studies, 2004, ISBN 1-885350-09-0.

13. “The Effects of Nuclear Weapons,” Samuel Glasstone, United States Atomic Energy Commission, 1962. This classic work is still available through used book sources.

14. “The Medical NBDC Battlebook,” Tech Guide 244, U.S. Army Center of Health Promotion and Preventive Medicine, Aberdeen Proving Ground, MD, 2002.

15. More information on the Homeland Defense Equipment Reuse program, HDER, can be found at www.ojp.usdoj.gov/odp/, then click on “Equipment” and finally HDER or telephone the Office of Domestic Preparedness at (800) 368-6498.

16. The complete text of the 2008 National Response Framework, NRF, and all current Annexes can be found at www.fema.gov/NRF.

15

Radiation Protection Standards and Regulations

Outline of this Chapter

Chapter Summary

This is the final chapter of Unit 3, Radiation Protection Operations. It is also the final chapter of the main text. It begins with some background information relative to standards development. The main international and U.S. organizations that are active in the field are identified. Their organizational structure and role are each examined.

There are a number of different types of radiation protection standards. The nature and interrelationships between four of these are examined. The standards covered here are regulations, regulatory guides, recommendations and license conditions.

Practical, useful standards must be based on a firm foundation of scientifically valid risk data and sound protection philosophy. The ALARA philosophy is the driving force behind all protection programs. The ALARA approach is necessary due to the lack of sufficient fundamental knowledge of the biological effects and potential risks of human exposures. Another basic philosophy is the idea that radiation work should offer a potential risk of harm to the worker which is comparable to that found in other "safe" occupations. Finally, the added risk of radiation work is offset by some practical benefits to the worker.

Each of the world's developed countries has a set of radiation control regulations in place to protect workers and the general populace. The details of current U.S. regulations as found in 10 CFR 20 and 10 CFR 835 are presented. The Part 20 standards apply to the vast majority of U.S. workers. Workers and subcontractors under the direction of the Department of Energy are subject to the Part 835 standards. The Rad Con Manual is the practical interpretation of these DOE standards.

Many countries depend heavily on recommendations of the ICRP for their protection programs. The current version of these is ICRP Publication 103 published in 2007. The philosophy and numerical standards of this approach are covered. The chapter concludes with a brief look at the NORM problem. Numerical standards from the few published or proposed standards are given and compared.

Standards-Setting Organizations

ICRP

Historically, the International Commission on Radiological Protection, the ICRP, was established by the Second International Congress of Radiology in 1928. (It was called, at the time, the International X-ray and Radium Protection Committee). This body is charged with preparing recommendations at the international level on basic principles of radiation protection. The present organizational form was assumed in 1950. The Commission has a chairman and up to twelve other active members. Each person is chosen to serve a four-year term through nominations by national delegations to the meetings of the International Congress of Radiology or by present members of the ICRP. The Commission usually meets annually to conduct business. Nonvoting experts are sometimes invited to assist in the deliberations.

Developments in radiation protection are overseen by four standing committees. These are responsible for Radiation Effects, Internal Exposure, External Exposure and Application of the Commission's Recommendations, respectively. Most of the actual day-to-day work is carried out by temporary "task groups" of international experts who volunteer their time to consider particular topics of interest in their fields. These studies ultimately become incorporated into Committee Reports or are issued as Recommendations of the ICRP.

These recommendations of the Commission are frequently used as a basis for national regulations and codes. The basic ideas and the numerical values are adapted to the style and needs of each country. Herein lies the value to the practicing radiation protection professional. The latest publications of the ICRP are like a crystal ball – they enable the forecasting of things to come in the continuing evolution of radiation protection standards.

ICRU

The International Commission on Radiation Units and Measurements, the ICRU, was established in 1925 by the First International Congress of Radiology. It has as its basic objective the development of internationally acceptable recommendations in three areas:

1) Quantities and units of radiation and radioactivity,

2) Procedures that are suitable for the measurement and application of these quantities in clinical radiology and radiobiology,

3) Physical data needed in the application of these procedures, the use of which tends to assure uniformity in reporting.

At present, the Commission consists of a chairman and twelve members. One or more of these members are assigned to the twelve "technical areas" which constitute the ICRU's field of interest. The technical areas are responsible for evaluating the latest data to identify potential topics for ICRU activity. They cover various aspects of the uses of radiation and radioactivity in medicine, physics, chemistry and biology.

The actual preparation of a new document is assigned to a specific "report committee" set up for that purpose. When the final report is approved, it is assigned a serial number and published by the ICRU for distribution.

IAEA

The International Atomic Energy Agency, the IAEA, was chartered by the United Nations on October 23, 1956. The headquarters is situated in Vienna, Austria. The stated objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world." It is particularly active in "developing" countries where funds are committed upon invitation of the host country. This work is usually carried out through international experts who are assigned to projects for some specified time period. Over 100 countries are "member states" of the IAEA.

In addition to projects around the world, the Agency also sponsors numerous international conferences on the latest developments in the atomic energy field. It has a vast publishing section which issues comprehensive reports in four major series – the Proceedings Series, the Panel Proceedings Series, the Safety Series and the Technical Report Series. Many of these Technical Reports are of direct use to radiation protection technologists. Current availability can be obtained from the website www.iaea.org/Publications/.

NCRP

Turning now to national radiation protection standards organizations in the United States, the National Council on Radiation Protection and Measurements, NCRP, is one of the oldest. The Advisory Committee on X-ray and Radium Protection was formed in 1929 to adapt the recommendations of the ICRP to the needs in the USA. In 1946 the Committee felt that the problems in radiation protection had become so much broader that it should change its name and scope. The Committee enlarged and became the National Committee on Radiation Protection. Eight subcommittees were formed to prepare recommendations in their area of expertise.

In 1964, The U.S. Congress chartered the successor as a nonprofit organization, the National Council on Radiation Protection and Measurements. Its task was to act as an information clearing-house for radiation protection and radiation measurement developments, to provide liaison among scientific organizations working in the field, develop new basic concepts and to cooperate with ICRP, ICRU and other organizations concerned with radiation measurements and protection. As of 2000, it was composed of one hundred twenty-six regular and honorary members, and five officers. The workload is distributed among ninety-three Scientific Committees composed of experts which draft recommendations that are acted upon by the full Council. Approximately fourteen of these Scientific Committees were actively engaged in formulating recommendations during 1999. The NCRP Reports represent the state of the art in radiation protection and many are of direct applicability to work by technologists. A listing of current reports and prices is available at www.ncrponline.com/.

A group of NCRP Collaborating Organizations facilitates and stimulates cooperation among a wide variety of national professional organizations. Similarly, the NCRP Special Liaison Program involves international and U.S. Government organizations which have particular expertise in radiation protection and measurement.

NRC

The United States Nuclear Regulatory Commission, the NRC, was established by the Energy Reorganization Act of 1974 which abolished the Atomic Energy Commission (the AEC was formed in 1954) and transferred its licensing and related regulatory functions to the NRC. The Act became effective January 19, 1975. The NRC consists of five commissioners, one of whom is named Chairman by the U.S. President, with a large supporting staff and four Regional Offices. Commissioners must be confirmed by the U.S. Senate. A map of the regions is given in Figure 1.

The basic regulations established by the NRC are published in the Federal Register and become incorporated into the Code of Federal Regulations, CFR. In addition, the NRC publishes Regulatory Guides, Branch Position Papers and NUREG reports which provide supporting information. Some of these different publications will be described more fully in the next section of this chapter.

The stated mission of the NRC is "to ensure adequate protection of the public health and safety, the common defense and security, and the

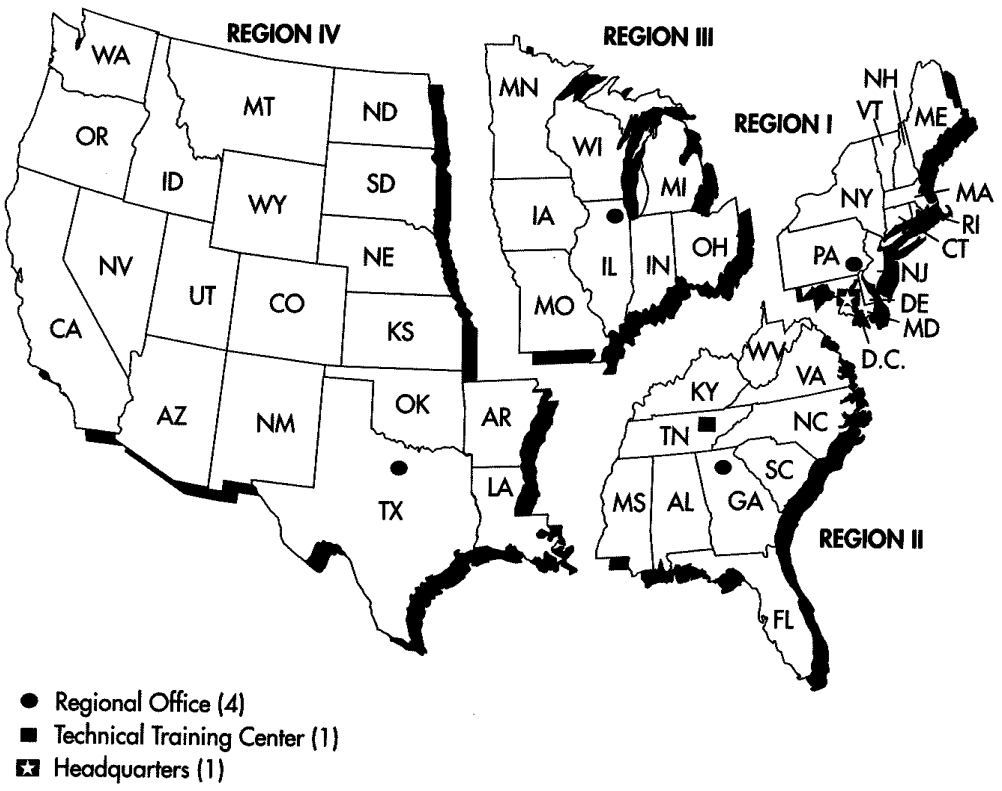


Fig. 1 - A map of the four NRC regions

environment in the use of nuclear materials in the United States." Major activities include licensing of reactors, nuclear materials, reactor operators, and low level waste sites, inspection of the licensed facilities, research in reactor safety, investigation of incidents and enforcement of licensing responsibilities. They also certify private uranium enrichment facilities, conduct public hearings on matters of nuclear and radiological safety, maintain the NRC Incident Response Program (which includes the Operations Center) and collect information about the operational safety of commercial nuclear power plants. During 2010, 75% of the 3,960 staff members were assigned to reactors, 23% to nuclear materials and waste safety and the remaining 2% to the Inspector General. The operating budget was $1,067,000,000 in 2010.

ANSI

The American National Standards Institute, Incorporated (known as the American Standards Association prior to 1966) is in the business of developing standards, many with direct application in the nuclear energy field. ANSI is a federation of organizations from the government and from the private sector including representatives from trade, professional, consumer, and technical groups. It acts as a coordinating body to reduce costly overlap of efforts. Experts serve on ANSI committees which formalize procedures for radiation protection practices as well as specifications for a wide variety of components and equipment used in radiation protection. The actual writing is done by a working group which is established specifically for that purpose. The opinions of consultants and, in some cases, public review contribute to the final document. The ANSI standards are frequently referenced in NRC Regulatory Guides. To obtain a copy of a standard, visit their website http://www.webstore.ansi.org/.

Agreement States

The Atomic Energy Act of 1954 (which founded the AEC) also provided for the establishment of "agreement states." Such states enter into a legal contract with the NRC in which they assert their willingness and competence to regulate the safe uses of radiation and byproduct radioactivity within the boundaries of their respective states. If the NRC concurs, they become added to the list of agreement states. Figure 2 shows the 37 states which, as of 2010, have assumed responsibility for issuing radioactive materials licenses and inspecting and enforcing compliance with stated conditions and provisions. During 2010, the agreement states administered some 19,600 different radioactive materials licenses. Note that the agreement states are NOT given authority to regulate federal facilities (e.g., DOE weapons development facilities or military installations) or nuclear reactors located within their boundaries. The reactor limitation applies to both power reactors and to research/training reactors. The NRC retains total regulatory control in all of these cases. Also, the radiation control regulations in force in each agreement state must provide at least as much protection for the health and safety of the population as the NRC regulations do in the non-agreement states.

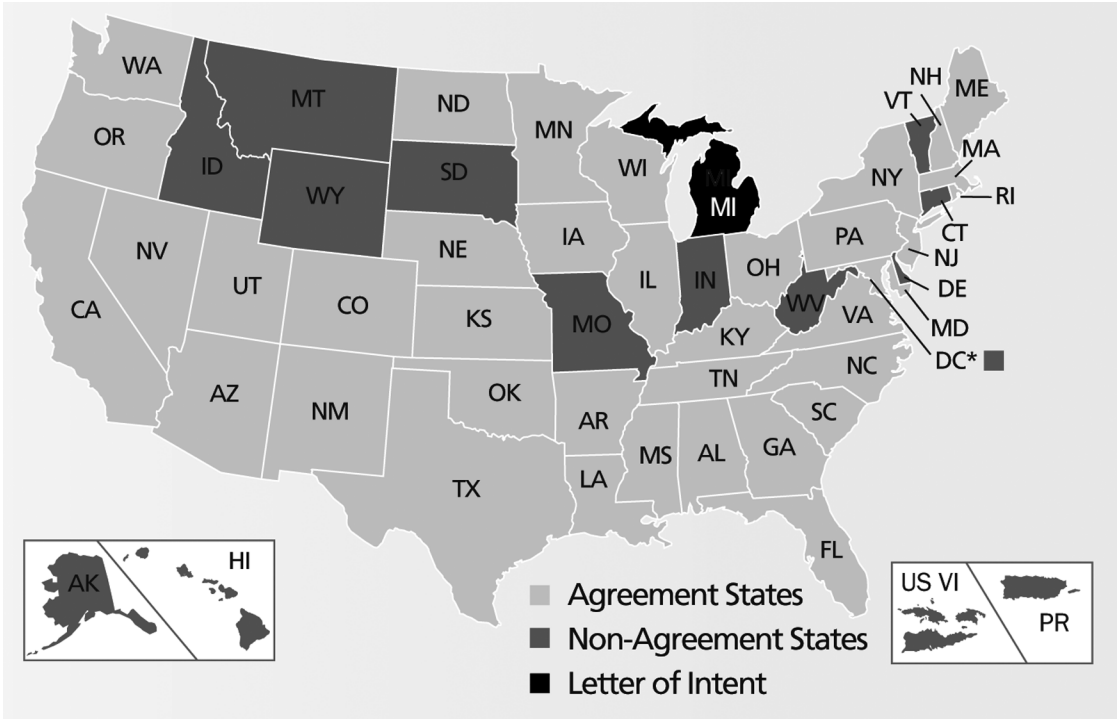


Fig. 2 - The 37 NRC agreement states as of 2010

Types Of Standards

Regulations

A regulation is a legal statement actually written into a governmental code. It carries the force of law which means that penalties can and will be imposed for non-compliance. For example, in 1992, the NRC imposed fines against 91 licensees for a total of $4,384,050.00! On appeals, only $22,500 of this total was withdrawn. Individual fines ranged in size from $125.00 to $300,000 (against a nuclear power licensee). In fiscal year 1998, the NRC collected fines and civil penalties of $5,200,000 from licensees. Technicians beware – enforcement is on the increase. In 1987, only thirty-one violators received civil penalties. An analysis by the NRC identifies three major causes for action – failure of management to understand license conditions, failure to train workers properly and failure to assure that employees follow approved procedures.

Regulations generally are written in the form of basic principles and philosophy. This avoids the complications brought on if the legislators attempt to write codes too specifically, which means that they then must attempt to cover all possible loopholes in advance. By keeping the language general, future changes in practice and

loopholes can be dealt with through interpretations by the regulatory agency. This allows for a much quicker response and requires much less paperwork than amending the national code. Part 20 of Title 10 of the Code of Federal Regulations (abbreviated 10 CFR 20) is an example of a regulation that a practicing radiation protection technologist should be familiar with. As an illustration of the concept of general vs. specific regulatory language, 10 CFR 20.1203 specifies, "The determination of the deep-dose equivalent... should be based upon measurements using instruments or individual monitoring devices." At no place in the regulations are the specific types of equipment that are acceptable for these measurements spelled out. As the state of the art in personnel dosimetry changes, the interpretation of acceptable instruments also changes by the NRC or the regulatory authorities in the agreement states.

Regulatory Guide

A "Reg Guide" is published by one of the regulatory agencies such as the NRC or an agreement state division. It is a technical aid in the interpretation of some regulation. In a sense, it tells the licensee what the compliance inspector expects to see in the way of a program, protective equipment and radiation safety procedures. Since the regulatory guide is not incorporated into the governmental codes, IT DOES NOT CARRY THE FORCE OF LAW. If a licensee receives a citation for a deficiency at a facility, the citation will always reference the original code section rather than a reg guide. A licensee may choose to ignore the suggestion of a particular reg guide, but at the time of inspection it must be demonstrated that the alternate practice provides at least the same degree of protection as the practice recommended by the guide. Thus, it often becomes simpler to follow the guide in the first place. To illustrate the variety of NRC Regulatory Guides, a sampling of some titles is presented in Figure 3.

With staff cuts and reduced budgets, in at least one instance, the NRC is phasing out a Draft Regulatory Guide without issuing a final guide. The draft guide for license termination has been replaced by a Consolidated NMSS Decommissioning Guidance NUREG report. Formerly, these documents were intended for internal NRC use by persons reviewing applications. They now appear to be directed at a wider audience.

"Loose Parts Detection Program for the Primary System of Light-Water-Cooled Reactors"

"Review of Experiments for Research Reactors"

"Nuclear Criticality Safety in the Storage of Fissile Materials"

"Leak Testing Radioactive Brachytherapy Sources"

"Procedure for Picking Up and Receiving Packages of Radioactive Materials"

"Film Badge Performance Criteria"

"Radiation Dose to the Embryo/Fetus"

"Demonstrating Compliance with the Radiological Criteria for License Termination"

Fig. 3 - A sample of NRC regulatory guide titles

NUREG

The NUREG series of publications is published by the U.S. Nuclear Regulatory Commission. These documents DO NOT CARRY THE FORCE OF LAW. Instead, they are reports or brochures on regulatory decisions, results of research, summaries of incident investigations and other technical or administrative information. They are authored by NRC staff members, or NRC contractors. A few NUREGS are conference proceedings and some result from international agreements. A number of the technical reports are a ready source of information for technologists. All of the documents are available free for download from http://www.nrc.gov/reading-rm/doc-collections/nuregs/. A few titles are listed here to illustrate NUREG content.

- Mixed Oxide Fuel News
- Characterization of Radioactive Slags
- Nuclear Power Plant Fire Protection Research Program
- NRC Inspector Field Observation Best Practices
- Circuit Bridging of Components by Smoke

Recommendation/Consensus

A recommendation standard (sometimes called a consensus standard) DOES NOT CARRY THE FORCE OF LAW. It is a statement promoted by some body of experts such as the ANSI or the NCRP. Since these organizations are not legislative, their guidance is strictly of an advisory nature. In the case of recognized international or national groups such as the ICRP or the NCRP, these statements clearly constitute the state of the art in good practice. However, the language and the large amount of detail and practical information of the reports of such organizations does not usually lend itself to the direct incorporation into legal codes. Following such recommendations does carry some assurance of satisfying actual regulations due to the reputation of the issuing group. Note that, in some cases, recommendation standards are incorporated into legal codes by reference. For example, the California Administrative Code section dealing with shielding design for medical x-ray units specifies that “This requirement shall be deemed to be met if the thicknesses of such barriers are equivalent to those computed in accordance with Appendix C of the National Council on Radiation Protection and Measurements Report No. 49.”

License Condition

This last category of “standard” refers to the “fine print” provisions that are part of the radioactive materials license issued to a licensee. These requirements are usually very site-specific. That is, conditions beyond those specified in the license application are imposed as a result of the analysis of the license application by the regulatory agency. As examples, the frequency of certain tests or calibrations may be increased or certain areas designated as restricted that were not so designated originally in the application. By its nature, A LICENSE CONDITION CARRIES THE FORCE OF LAW relative to the license into which it is incorporated. Citations can be issued if the licensee does not live up to the specified conditions.

Bases For Protection Standards

ALARA

The foundational basis of all protection standards is that all exposures should be As Low As Reasonably Achievable, economic and social factors being taken into account. Thus, the permissible levels represent upper limits rather than a goal to be achieved. These conservative upper limits represent a dose to the individual which "in the light of current knowledge, carries negligible probability of severe somatic or genetic injury." As an underlying basis of the standards, it demands that continual reassessments be made regarding all phases of the radiological operations being undertaken by a licensee. It forces us to conclude that a given degree of radiological protection is never "good enough."

Biological

Ultimately, all of the numerical standards for radiation protection are tied back to radiobiology data. In the earliest standards, this data was chiefly from animal experiments. As experience has grown, more and more human data has been obtained through the use of radiation in a variety of medical treatments, through data from occupational workers, from radiation accidents and from the survivors of the Japanese atomic bombings. Unfortunately, much of the data has been obtained at doses and dose rates well in excess of those encountered in the workplace. For example, the average dose rate received by workers (who actually received a recorded dose above background radiation levels) in 1991 at U.S. commercial nuclear power stations was 0.00029 rem per hour. The typical dose rates for medical uses might be 100,000 or 1 million times higher. Thus, traditional practice has always been to assume the LINEAR HYPOTHESIS, i.e., that we can extrapolate (guess) the effects of a radiation dose at high levels by drawing a straight line to zero dose. This means that small amounts of radiation are assumed to produce a small amount of deleterious effects. Another way to state this is that biological repair is assumed to not exist! Actually, at low rates, such repair has been demonstrated in virtually all living tissues. Thus, the standards have a "safety factor" built in. We say that the standards are conservative, i.e., they overestimate the risk of harmful effects. The recent work by Professor Luckey and others demonstrating that radiation appears to be beneficial to health at low levels has yet to be taken into account by standards-setting organizations. Therefore, the permissible levels of radiation exposure in the standards for occupational workers and members of the general public represent conservative guesses as to the deleterious effects on health that would be seen in the average member of the exposed population.

Comparable Risk

Another basis of the radiation standards is to make the occupational risk in the nuclear industry no greater than that in other "safe occupations." In 1977 the

ICRP defined the risk of safe occupations as having an annual mortality not in excess of 10^{-4}. This means the risk of dying as a result of working in a safe occupation is 1 chance in 10,000 per year. Although the idea of comparable risk is relatively simple, it is difficult to apply in radiation protection. This is because radiation risk is difficult to quantify since, as mentioned above, we do not have definitive biological data at low doses and rates. Thus, the standards are again applied conservatively. Radiation work is probably safer than most "safe occupations."

Checks and Balances

The setting of protection standards for workers involves a system of checks and balances. In general, the technical competence of the radiation worker in assessing the risks is higher than that of the general public due to required training programs. The workplace is subject to inspection by a regulatory agency and the licensee can be fined or lose the license in cases of noncompliance. The worker has the benefit of workmen's compensation laws which make financial awards if it can be shown that the conditions of employment caused or aggravated some disease or injury. In contrast to the general public, the actual doses received are individually measured and recorded. The number of radiation workers is small compared to the total population. Persons under eighteen years of age are excluded from the radiation workplace. Finally, most radiation workers are subjected to some form of pre-employment medical examination which is designed to exclude persons who might be adversely affected by such employment.

Dose Limiting Regulations

Occupational Workers

In the United States, the permissible occupational doses of ionizing radiation are detailed in two different standards, depending on employer. Persons employed in the private sector (about 95% of the nuclear work force) are subject to the Code of Federal Regulations, specifically Part 20 of Title 10, 10 CFR 20, or the equivalent agreement state standards. Persons employed at U.S. Department of Energy facilities fall under the provisions of Part 835 of Title 10, 10 CFR 835. They are required to comply with policies published in the DOE Standard Radiological Control (DOE-STD-1098-99, issued July 1999). In the rest of the world, most countries have adopted the 1990 recommendations of the International Commission on Radiological Protection, the ICRP. These recommendations have been published as ICRP Publication 60.

The earliest provisions of 10 CFR 20 were written into the code in 1954. A number of amendments have been made over the years to account for changing circumstances. A major revision of radiation protection philosophy was adopted by the ICRP in 1977 (and published as ICRP Publication 26 and ICRP Publication 30). This led to a world-wide reevaluation of national codes. The U.S. NRC decided to completely overhaul 10 CFR 20 to bring it substantially into compliance with the ICRP. In December, 1985, the NRC published for public comment the results of this effort.

Shortly thereafter, legal action was taken to block implementation. In October 1987, the NRC announced that the litigation had been settled and that it was moving ahead again. The schedule called for publication of the Final Rule in the Federal Register in October 1988 and an effective date of January 1, 1991. However, due to the huge volume of comments received, and petitions from industry groups, the Final Rule was published in May 1991, with an effective date of June 1991. Licensees were allowed to defer implementation until January 1, 1994 at which time implementation became mandatory.

In the Department of Energy sector, DOE Order 5480.11 went into effect in January 1989. It superseded DOE 5480.1A which had been published in 1981. The provisions governed radiation safety at all DOE facilities and also applied to DOE contractor operations. In order to provide a practical guide to DOE workers, the Department published a manual of policies and procedures which satisfied the requirements of DOE Order 5480.11. This guidebook was called the Rad Con Manual. It was superseded by the DOE Standard Radiological Control. There was a long-standing problem with this approach to standards by the DOE – the DOE Order did not carry the force of law. In order to rectify this situation, the provisions of DOE Order 5480.11 were rewritten as a proposed new Part 835 of Title 10 CFR, circulated for comment in 1991. Publication of the Final Rule took place on December 14, 1993 and Part 835 became effective January 13, 1994. This approval means that 10 CFR 835 has now replaced DOE Order 5480.11.

The 1990 recommendations of the ICRP as found in ICRP Publication 60 are an update of the 1977 ground-breaking work in Publications 26 and 30. Substantial work went into refining the biological risk estimates used in the previous publications. This involved taking the UNSCEAR risk factors based on high doses and high dose rates and applying dose and dose rate effectiveness factors to estimate risks at low doses and rates. Recent data from the Japanese survivors assisted this projection. Also, in contrast to the 1977 recommendations, the risk factors included genetic damage and nonfatal cancers in radiation detriment. Finally, The 1977 dose limits were set by comparing risk of radiation fatality with fatality risk in safe occupations. The 1990 recommendations broaden the picture to include the effects of loss of life expectancy, time lost if death occurs and annual risk of death as well. The result of these risk estimate changes is a lowered dose limit for occupational workers compared to the 1977 recommendations.

Members of the Public

Two of the standards mentioned above explicitly include dose limits for radiation exposures from licensed sources to members of the public. These are persons receiving this radiation exposure while not on the job as a radiation worker for the licensee. Thus, a nuclear plant worker who is exposed to an industrial radiography gamma ray field while passing a radiography hot cell on his way home would be a member of the public for purposes of dose limits at that moment in time. Both 10 CFR 20 and ICRP Publication 60 contain explicitly stated dose limits for members of the public. The 10 CFR 835 does list limits in the more restrictive case of a member of the public entering a controlled area on a DOE facility.

A second approach to protecting members of the public from risks caused by

radiation exposure is exemplified by the U.S. NRC's introduction of the policy of "Below Regulatory Concern" or BRC. The idea is that in view of the fact that natural background radiation levels vary by more than 1 mSv per year across the U.S.A., it is reasonable to cease to be concerned about any licensed facilities which add only a small fraction of a millisievert per year to the surrounding radiation environment. Both the NRC and the NCRP established a numerical value of 0.01 mSv per year as being so insignificant that no regulatory controls need be exercised to fully protect public health. However, an immediate, loud cry of protest was registered by a variety of public interest and environmental protection groups who felt that the NRC was delinquent in its responsibilities to control all radiation sources in the country, no matter how trivial a risk they presented. The original motivation for the NRC was to establish an acceptable BRC level in order to save limited regulatory resources for inspection and enforcement activities at licensees who presented a potential risk to public health. As a result of the public outrage, the NRC withdrew its BRC policy late in 1991, and no further proposals have been forthcoming.

Attention will now be given to the analysis of the major radiation protection standards themselves. First, 10 CFR 20 will be covered in detail. Each of the fifteen Subparts will be discussed. Next, the chief differences between 10 CFR 20 and DOE standards will be explored. Finally, the differences found in the ICRP recommendations in Publication 60 will be examined briefly.

10 CFR Part 20

10 CFR 20 is divided into fifteen Subparts. They will be addressed below. In addition to the major numerical limits and procedures, some interpretation of what is required by licensees will also be made. The interpretations given here are based on both the "Supplementary Information" published by the NRC in the Federal Register along with the final rule, and the five sets of **Questions and Answers on New Part 20** published by the NRC Office of Nuclear Reactor Regulation between 1991 and 1993.

Subpart A - General Provisions

The first Subpart begins with a statement of purpose. It clarifies the idea that dose limits should be enforced "to the extent practicable during emergencies" but that public health and safety (e.g., lifesaving actions) take precedence. The bulk of this subpart consists of detailed definitions - eighty-one of them! In contrast to the earlier version, all definitions have been grouped together in one section for ease of reference. Many of the definitions were taken from ICRP recommendations. In addition, to clarify things for licensees, some quantities not given unique names by the ICRP are also defined, such as eye dose equivalent and total effective dose equivalent (TEDE). Note that the roentgen is no longer defined or used in 10 CFR 20.

The fast neutron quality factor remains numerically identical to the "old" 1954 version. This is remarkable in view of the increased value recommended by the ICRU, the ICRP and the NCRP. The NRC's rationale is that the higher value used by the others was justified on the basis of Japanese survivor data where the neutron RBE was referenced to Co-60 gammas. However, earlier neutron RBE studies used lower energy x-rays as the reference. The NRC now feels that higher energy photons are less

CONTROLLED AREA	RESTRICTED AREA
Not required for a license	
Boundaries should be documented	Boundaries should be documented
Posting as a controlled area not required	Posting as a restricted area not required
Doses may be "Occupational" or "Public"	Doses are usually "Occupational"
Only if doses are occupational; training must be given	All persons entering must be trained
Only if doses occupational, persons must be informed occupational limits apply	All persons must be informed that occupational limits apply

Fig. 4 - A Controlled Area versus a Restricted Area

effective and that would artificially inflate the neutron RBE. Until further research clarifies this issue, the NRC is staying with Q = 10.

Note that there are separate and distinct definitions of restricted areas and controlled areas. In past practice, they were considered identical by most licensees. The current definitions are compared in Figure 4. The whole point of distinguishing a controlled area from a restricted area is that the controlled area allows for two different dose limit situations. If an occupational worker is performing assigned duties there, the occupational limits apply. If the same individual or another person (who is not a designated radiation worker) is just passing through and their activities in the controlled area "are not closely and frequently connected to the licensed activity," their doses are considered "Public" doses.

"Very High Radiation Area" is an important definition. This is defined as any humanly accessible location where the dose rate exceeds 500 rads (5 Gy) in one hour at one meter from the source. The dose is evaluated at a tissue depth of 1 cm, i.e., it is a deep dose equivalent. The NRC felt the need to distinguish high radiation area dose rates (1 mSv/hour) from really deadly dose rates, hence the "very high" designation. In high radiation and radiation areas, the defining dose rates are specified to be measured at a distance of 30 centimeters from the source.

The truly careful reader noticed that the definition of Very High Radiation Area included a dose rate measured in rads or gray. Why not rem or sieverts? A good question! Recall from Chapter 4 that RBE values from radiobiology were used as a basis for the Quality Factors that convert rad to rem. The RBEs were based on cancer incidence and on genetic mutational risks of radiation exposure when the radiation is delivered at relatively low dose rates. Cancer and mutations were identified in Chapter 4 as being stochastic effects (the chance of the effect, not the severity, is dependent on dose). Now, dose rates of over 500 rads in an hour are a different thing entirely. These extreme radiation fields can cause death, skin burns and acute radiation syndrome in a short time. These effects are non-stochastic (the severity depends on dose and there is a threshold). The NRC has not found it necessary to define Quality Factors for non-stochastic effects. Thus, rads are more "appropriate" for measuring Very High Radiation Areas.

The rest of this Subpart is devoted to radiation units. Both the SI metric units and the old, special units are introduced. The flux to dose equivalent factors for neutrons listed in Chapter 5 of this text appear in this Subpart. A short summary of Subpart A is given in Figure 5.

Subpart A Summary

Purpose:

Introduces the definitions for terms and units used in the remainder or 10 CFR 20.

Subjects Covered:

Purpose of 10 CFR 20, Definitions, Units of Dose, Units of Radioactivity, Implementation.

Chief Provisions:

ICRP 26 tissue weighting factors are used, i.e., gonads 0.25, breast 0.15, red marrow 0.12, lung 0.12, thyroid 0.03, bone 0.03 and remainder 0.30.

1 Sv = 100 rems. 1 Bq = 1 disintegration per second.

1 Ci = 3.7 x 10^{10} disintegrations per second = 2.22 x 10^{12} disintegrations/min.

Q = 10 for neutrons of unknown energy and high energy protons.

Q = 20 for alpha particles.

Q = 1 for x-rays, gamma or beta radiation.

Fig. 5 - Subpart A summary

Subpart B - Radiation Protection Programs

This next Subpart occupies only three paragraphs in the Code and requires that each licensee develop a formal ALARA program. In addition, a formal review or audit must be conducted annually of a) program content and b) extent of program implementation at the licensee. Records of the annual audit must be maintained. The ALARA program need not be a stand-alone document but can be incorporated in the license application.

In 1997, this Subpart of the Code was amended by the NRC to establish a limit of 10 mrem per year TEDE to any member of the public from air emissions of radioactive material from all licensed facilities except nuclear power reactors. This adoption by NRC made it unnecessary for licensees to annually satisfy Environmental Protection Agency regulations on radioactive airborne releases.

Subpart C - Occupational Dose Limits

Subpart C contains the numerical occupational dose limits for adults, minors and an embryo/fetus. All dose limits are annual so as to conform with ICRP recommendations regarding the time span over which the dose is limited.

As discussed at length in Chapter 9, the standards require summation of internal and external doses in the case that both internal and external doses are likely to exceed 10% of the annual limits. This sum, the Total Effective Dose Equivalent or TEDE, cannot exceed 0.05 Sv annually, or the dose to any organ or tissue (with the explicit exemption of the lens of the eye which is limited to 0.15 Sv) cannot exceed 0.5

Sample Problem 1

GIVEN:

A 16 year old high school student receives 3 mSv deep dose equivalent and 4 mSv CEDE while in a work-study program in a local radiation laboratory.

FIND:

Has she exceeded allowed levels under 10 CFR 20?

SOLUTION:

To compare with the standard, the TEDE must be calculated. TEDE = Deep Dose Eq. + CEDE = (3 + 4) mSv = 7 mSv. The student's assigned duties involved radiation exposure so she has received an occupational dose. However, due to age, the minor dose limit applies. The limit is thus 10% x 50 mSv = 5 mSv. So the student has exceeded the dose limit.

Sv. If annual limits are exceeded, the excess dose must be subtracted from the Planned Special Exposure (PSE) limits for that individual.

The Planned Special Exposure was launched with the 1991 10 CFR 20 version. The PSE allows a worker to exceed the annual dose limits under exceptional circumstances and after a licensee complies with the requirements in Figure 6. The PSE cannot be used just to reduce collective dose at a facility. One clear use would be for emergency lifesaving actions. The dose limits for a PSE are 0.05 Sv in a year and 0.25 Sv in a lifetime. Note that the NRC is willing to make an advance determination of an "exceptional situation" in certain cases.

The occupational dose limits for minors are set at 10% of the applicable adult limits. Occupationally exposed minors are persons under 18 years of age whose "assigned duties involve exposure to radiation as a necessary feature of those duties." Examples might include "candy stripers" volunteering in a nuclear medicine department of a hospital or student technicians employed at a university research accelerator lab. See Sample Problem 1.

Finally, this Subpart concludes with standards for exposure of an embryo/

Worker informed of purpose of operation
Specifically authorized in writing
Worker given a dose estimate for the operation
Worker instructed in ALARA measures
Worker's prior PSE doses are determined
Licensee reports to NRC within 30 days:
- **Exceptional circumstances**
- **Authorizing management official**
- **What actions were necessary and why**
- **How doses were kept ALARA**
- **Expected and actual doses received**

Licensee reports dose to worker within 30 days

Fig. 6 - Conditions on use of the Planned Special Exposure

FORM LETTER FOR DECLARING PREGNANCY

This form letter is provided for your convenience. To make your written declaration of pregnancy, you may fill in the blanks in this form letter, you may use a form letter the licensee has provided to you, or you may write your own letter.

DECLARATION OF PREGNANCY

To: _____

In accordance with the NRC's regulations at 10 CFR 20.1208, "Dose to an Embryo/Fetus," I am declaring that I am pregnant. I believe I became pregnant in_____ (only the month and year need be provided).

I understand the radiation dose to my embryo/fetus during my entire pregnancy will not be allowed to exceed 0.5 rem (5 millisievert) (unless that dose has already been exceeded between the time of conception and submitting this letter). I also understand that meeting the lower dose limit may require a change in job or job responsibilities during my pregnancy.

(Your Signature)

(Your Name Printed)

(Date)

Fig. 7 - Suggested form letter for declaration of pregnancy

fetus. The basic limit is 5 mSv during the entire pregnancy of a declared pregnant woman. In addition the licensee must attempt to keep the monthly dose rate uniform. The dose is the sum of the deep dose equivalent to the mother plus the internal dose from radionuclides in the mother and the embryo/fetus. If the limit has already been exceeded or is as high as 4.5 mSv at the time the pregnancy is declared, then a dose to the embryo/fetus of 0.5 mSv during the remainder of the pregnancy will comply with the regulations. Note that the woman is not obligated to declare her pregnancy. In that case, the licensee is not obligated to meet the 5 mSv dose limit. Also, the declaration can be withdrawn. However, in this case, the licensee must "continue to maintain the records of dose to the embryo/fetus that were prepared as a result of the woman's declaration of pregnancy." A declaration of pregnancy does not require documented medical proof. See Figure 7 for the Regulatory Guide sample declaration letter. Figure 8 summarizes the provisions of Subpart C.

Subpart D - Radiation Dose Limits for Individual Members of the Public

This is a relatively short Subpart. Members of the public cannot be exposed above 1 mSv of TEDE in a year from any single licensee. This limit applies to sources under control of the licensee except that authorized sewer releases are not counted. Figure 9 summarizes the provisions of Subpart D.

In addition to the annual limit, unrestricted area external doses (deep dose equivalent, eye dose equivalent and shallow dose equivalent) cannot exceed 0.02 mSv in any one hour time period. However, there is no control of the instantaneous dose rate which could be much higher for short periods. Further, this limit does not apply inside a controlled area.

Subpart C Summary

Purpose:

Lists the numerical limits for occupational radiation workers including adults, minors and an embryo/fetus

Subjects Covered:

Limits for Adults, Summation of Internal & External, External Dose from Airborne Material, Internal Dose, Planned Special Exposures, Limits for Minors, Dose to Embryo/Fetus

Chief Provisions:

Annual limit, TEDE, of 0.05 Sv or dose to any single organ of 0.5 Sv [except the eye limit of 0.15 Sv]

Annual limit of 0.5 Sv to skin or extremity [hand, elbow, arm below elbow, foot, knee, leg below knee]

Weekly limit on soluble uranium of 10 milligrams

NOTE: Limits for minors are 10% of all the above

Planned Special Exposure limits are equal to annual limits in a year and five times the annual limits in a lifetime

Embryo/fetus shall not exceed 5 mSv during pregnancy

Fig. 8 - Subpart C summary

There is a provision for temporary relief in the case of undue hardship for a licensee. By application to the NRC, an exemption may be granted to limit the annual dose to an individual member of the public to 5 mSv (0.5 rem). The licensee must show the need for and the expected duration of operations above the 1 mSv limit, must have an acceptable program to measure the dose, and must have procedures in place to keep the public dose ALARA.

Finally, this Subpart deals with concentrations of radioactive effluents released from licensed facilities (see Figure 9). Compliance can be shown by keeping releases

Subpart D Summary

Purpose:

Introduces external and internal dose limits for members of the public

Subjects Covered:

Dose Limits, Compliance Demonstration

Chief Provisions:

TEDE not to exceed 1 mSv in a year

Special exemption up to 5 mSv in a year possible

Unrestricted area less than 0.02 mSv in an hour

Effluents in air and water less than Table 2 Appendix B concentrations

Fig. 9 - Subpart D summary

within the 10 CFR 20 Table 2, Appendix B concentrations of water and air. The concentrations are allowed to be averaged over an entire year.

Subpart E - Radiological Criteria for License Termination

Subpart E was added to the code in 1997. It lays down rules for decommissioning of NRC licensed facilities. In order to release a facility for "unrestricted use," it must be decontaminated to such a level that an average future occupant will not receive over 25 mrem TEDE per year and the levels are also ALARA. In order to verify that the dose to future occupants can be held to the 25 mrem limit, the NRC has provided a NUREG document, **Consolidated NMSS Decommissioning Guidance**, to assist licensees in choosing dose models that are appropriate to their site and in performing a final status survey to measure residual radioactivity. (See "Other Resources" for this Chapter.) As discussed earlier in this book, licensees can elect to use the NRC's "DandD" computer code to determine the residual contamination level that would correspond to the 25 mrem/yr limit. The NRC also accepts the survey methods described in the **Multi-Agency Radiation Survey and Site Investigation Manual** (MARSSIM) as suitable for the final status survey to demonstrate compliance with Subpart E.

In case unrestricted release is not practical, criteria are included that allow termination under "restricted conditions." This requires setting up long-term institutional controls over the site and the posting of bonds or insurance. It also requires input from the general public regarding the decommissioning plans. The NRC is authorized to notify State and local governments, Native American Nations, the EPA, the Federal Register and local community newspapers of the intent of the licensee to terminate. Furthermore, the licensee has to demonstrate to the satisfaction of both the NRC and the local community that, if the institutional controls fail, the future occupants would not exceed 100 mrem/year of TEDE. A summary of these provisions is shown in Figure 10.

Subpart E Summary

Purpose:
Provides criteria for release of license terminated nuclear facilities
Subjects Covered:
Dose limits, restricted vs. unrestricted release, public participation
Chief Provisions:
TEDE should not exceed 25 mrem/yr for unrestricted release
If institutional controls are set up, can terminate under restricted conditions
Public must be involved in decision to release
Financial bond must be posted

Fig. 10 - Summary of Subpart E

Subpart F - Surveys and Monitoring

Licensees are required to make surveys of radiation levels, concentrations or quantities of radioactive material and potential radiation hazards. All equipment used

for radiation monitoring must be calibrated "periodically for the radiation measured." As a special case, personnel dosimeters needing processing (in other words, TLD, OSL and film badges) are singled out. These devices must be evaluated by a NVLAP accredited processor.

Next, this Subpart lays out the conditions under which external and internal personnel monitoring are mandatory. The basic criterion is that each is required if a quantitative evaluation shows that an adult worker is likely to receive a dose over 10% of the respective limit annually from the licensee's activities. That is, personnel badges are required if the likely deep dose equivalent is 10% of 50 mSv or any individual organ or tissue might receive 10% of 500 mSv deep dose equivalent. Internal dosimetry (e.g., bioassay or *in vivo* count) is required if intakes are likely to exceed 10% of the ALI. Similar 10% levels are established for minors and declared pregnant women.

One final requirement is specified. External personnel monitoring is required for all individuals entering either a high or a very high radiation area. See Sample Problem 2.

Sample Problem 2

GIVEN:
A nuclear plant contract worker receives 12% of annual dose limits at Plant A in the 1st quarter. He will be employed at Plant B for 6 months and then sent to Plant C. Based on duties at Plant B, it is unlikely that he will exceed 10% of annual limits while at that site.
FIND:
Does this worker need to be monitored at Plant B?
SOLUTION:
No. Even though the worker has exceeded the 10% threshold for monitoring while working at Plant A, work at Plant B is "a new ballgame!" Based on the NRC's interpretation, "doses that may have been received, or that may be received in the future, at another licensee's facility are not included in the determination of the monitoring requirement."

Subpart G - Control of Exposure From External Sources in Restricted Areas

There are two sections to this topic. Special precautions must be taken to prevent entry to high radiation areas and even more stringent precautions are mandated for very high radiation areas. For high radiation areas, one or more designated features must be present. These features include a device to reduce, upon entry, the dose rate below 1 mSv in an hour at 30 cm, a "conspicuous" visible or audible alarm, and locked entrances. Continuous surveillance may be substituted for the preceding "features." Alternate methods of control might be approved on a case-by-case basis.

A very high radiation area requires the above controls and, must be protected against unauthorized or inadvertent access. The NRC admits "that it is virtually impossible to prevent determined, willful circumvention of physical barriers." But steps must be taken to make such entry very difficult and detectable. This means that cut locks or holes in fencing would alert personnel to the unauthorized entry situation. Figure 11 summarizes the provisions of Subpart G.

Subpart G Summary

Purpose:
Entry control to large sources in restricted areas
Subjects Covered:
Access control to both high radiation areas and very high radiation areas
Chief Provisions:
Entrance needs device to reduce rate below 1 mSv an hour on entry
or activate conspicuous alarm
or it must be fitted with a lock
or be under continuous surveillance

Fig. 11 - Summary of Subpart G

Subpart H - Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas

Possibly in view of the fact of increased use of respirators in the nuclear industry in the 1980s, a new Subpart was dedicated to them in the 1991 revision. In effect, it begins with the admonition that respirators are a last resort solution. Before choosing to use respirators, the licensee "shall use, to the extent practicable, process or other engineering controls (e.g., containment or ventilation) to control the concentrations...."

As a word of caution here, there has been a tendency for licensees authorized to handle significant quantities of volatile radioiodine to have workers routinely take potassium iodide, KI, as a blocking agent to reduce thyroid doses. The NRC says that the use of KI is neither a process control nor an engineering control, that licensees are not authorized to require routine KI use. The NRC then stresses that KI use is not a substitute for preventative measures. These measures, incidentally, do not include the use of a respirator with iodine sorbents. With currently available respirators, the wearer has no warning when the service life of an iodine sorbent is reached. Thus, the respirator protection factor for iodine vapors is declared to be 1.0 in this case. However, the NRC does accept proposals, on a case-by-case basis, for higher protection factors for iodine canisters.

If process or engineering controls are out, respiratory protection is still not the next choice. Before choosing respirators, the licensee is required to consider limiting access or limiting exposure time to maintain the total effective dose equivalent ALARA. Incidentally, consistent with maintaining the TEDE ALARA, the standards clarify the point that even though the air concentration is below 1 DAC, the voluntary use of respirators is desirable and allowed.

A significant shift in philosophy has taken place by emphasizing that the TEDE be kept ALARA. In the past, licensees were encouraged to not allow workers any internal intakes. Under the new rules, external plus internal dose must be ALARA. Thus, a TEDE ALARA determination may well show that respirator use will increase the dose in a given situation and the licensee should specify that they will not be used. This has led to problems with some workers who are uncomfortable when entering an airborne radioactivity area without a respirator. In this case, "the licensee may exercise

discretion on a case-by-case basis in determining whether to grant approval to a worker's request for using a respirator." The NRC hopes for a transition period after which, through training and experience, workers will be able to accept the possibility of not using respirators in all airborne area entries.

A closely related issue also affects respirator usage. In certain cases, the added efforts needed with respirator use increase risk of heat stress. Or the vision-limiting nature of a respirator might make climbing very risky. In the situations where the increased risk of industrial accident exceeds the benefit of decreased radiation dose through respirator use then the respirator use is not reasonably achievable and thus not required.

Having disposed of those cases when respiratory equipment should not be used, the discussion turns to proper care and use when it is appropriate. A number of conditions on use and considerations apply if a licensee elects to establish a respiratory protection program. Figure 12 summarizes these conditions. It should be noted that the requirements of Figure 12 apply even if the respiratory protective equipment is maintained only for emergency use.

Finally, Subpart H provides that a licensee notify the NRC in writing at least 30 days before using respirators.

NIOSH/MSHA approved equipment only
Air sampling prior to personnel entry
Bioassays to evaluate actual intakes
Testing of respirators immediately prior to each use
Written procedures on: selection/fitting, maintenance, training, record-keeping
Annual determination of medical fitness of all wearers by a physician
Written policy statement on: respirator alternatives, use, periods of relief
Advise each user that they may leave area any time for relief
Respirators must provide proper visual and speaking capability

Fig. 12 - Requirements for a respiratory equipment program

Subpart I - Storage and Control of Licensed Material

This short Subpart consists of two brief paragraphs. Radioactive material in quantities above the "exempt quantity" limits must be secured to prevent unauthorized removal or access when the material is stored in either a controlled or in an unrestricted area. Furthermore, when not in storage, such material must be maintained under constant surveillance as long as it is in an uncontrolled or unrestricted area. Practically, this means that each time a radiotracer laboratory is unoccupied, the doors must be locked. The NRC feels this is a small price to pay for preventing unauthorized removal and subsequent contamination of unrestricted areas with microcurie amounts of radioactive materials. In view of the public and media hysteria that has accompanied past incidents of this nature and considering the astronomical costs of decontamination these days, this is certainly a reasonable and prudent way to do business.

Subpart J - Precautionary Procedures

A selection of procedures for day-to-day operations makes up this Subpart. The radiation symbol appearing on warning signs is defined here. Note that black is an acceptable color to use in place of magenta or purple for the three-bladed symbol on signs. The background color must be yellow. (The color requirements are removed if the source device is subject to high temperatures.) Posting of warning signs is required under five conditions. Proper warning signs must be posted for a very high radiation area, a radiation area, a high radiation area, an airborne radioactivity area and a radioactive materials area, The sign for the "very high" case must contain the wording "GRAVE DANGER, VERY HIGH RADIATION AREA." The word DANGER is not allowed to be used on a radiation area sign. This word is reserved for areas of greater hazard than the relatively benign radiation area.

There are some generic exemptions to the above posting requirements. Under certain conditions, rooms need not be posted as a radioactive materials area if the material is there for less than 8 hours at a time. As in the past, hospitals are exempted from some posting requirements. Rooms for radioactive patients or therapy rooms containing external sources for treatment can be exempted. Unless a room containing sealed sources meets the definition of a radiation area (over 0.05 mSv/hr at 30 cm) it does not need to be posted with a caution sign.

Containers of radioactive material must be so labeled. Also, the nuclide, quantity and date should appear. If the container is later decontaminated and removed to an unrestricted area, the labels must be removed or defaced. If the container holds quantities less than the Appendix C amounts or has a concentration less than Appendix B, Table 3 (sewer release concentrations), then it is exempted from labeling.

Procedures for receiving and opening radioactive packages have been tightened up. Packages with more than Type A quantities must be received when offered by the carrier or picked up expeditiously from the carrier. External package surfaces must be monitored, within three hours, by wipe test and for radiation level if the package is carrying a White I, Yellow II or Yellow III label. Monitoring is also required irrespective of the quantity contained if a radioactive package arrives crushed, wet or damaged. The carrier and NRC must be notified immediately if surface contamination is over 2200 dpm/100 cm^2 beta-gamma or 220 dpm/100 cm^2 for alpha emitters. External levels above 200 mrem/hr contact are also grounds for the notifications. Finally, licensees are required to have written procedures for safely opening radioactive packages.

Subpart K - Waste Disposal

Subpart K begins with a list of options for disposing of radioactive waste – transfer to an authorized recipient, decay in storage, or release in effluents, release into the sewer, or incineration. Four of these deserve further comment. Decay in storage is a relatively new feature of Part 20. It was added to clarify that this option is legally allowed. It can be applied by licensees that have short half-life nuclides in their waste. It is not a practical method for nuclear reactor waste.

If radwaste is released in effluents, the licensee must be able to demonstrate that the dose limits for members of the public in Subpart D have not been exceeded. For sewer disposal, specific conditions must be met. The material must be water soluble (a tightening of past practices) or "readily dispersible biological material" (such as ground up animal carcasses). The average monthly concentration cannot exceed the

Appendix B, Table 3 limits. Finally, the total activity released in a year is limited to 5 curies of H-3, 1 curie of C-14 and 1 curie of all other nuclides combined. Although incineration of rad waste is permitted, the incinerator must have an NRC permit which may be granted on a case-by-case basis after the NRC receives an application.

An actual example of the *de minimus* concept made its way into the rule book. The organization of radiation safety officers on university campuses successfully lobbied the NRC to exempt liquid scintillation wastes. These had been a big problem at research facilities for years as the solvents in the cocktail were often EPA listed and so the resulting brew was "mixed waste" meaning it presented both a radiological and chemical hazard. Virtually no disposal options existed for years for mixed waste. The final rule allows concentrations of 0.05 μCi/g of H-3 or C-14 in liquid scintillation cocktails or animal tissue to be treated as NOT RADIOACTIVE! Records of such disposal must be kept, however. See Sample Problem 3.

Sample Problem 3

GIVEN:

A nuclear medicine lab plans to dispose of 35 mCi Tc-99m pertechnetate down the sewer each day. Records indicate 1.7×10^9 ml of water released to the sewer annually by the facility.

FIND:

Does this proposed disposal method satisfy 10 CFR 20?

SOLUTION:

Monthly water volume down the sewer is 1.7×10^9 ml /12 = 1.4×10^8 ml/mo. If we assume 30 days per month, the activity dumped = 30 x 35 mCi x 1000 μCi/mCi = 1.05×10^6 μCi/month. Thus, the monthly average concentration will be (1.05×10^6 μCi/month) ÷ (1.4×10^8 ml/mo) = 7.5×10^{-3} μCi/mo. From Appendix B, Table 3 of 10 CFR 20, under Tc-99m, the allowed value is 1×10^{-2} μCi/ml. Thus, the proposed disposal method satisfies the monthly requirement. But the annual total dumped will be 0.035 Ci/day x 365 days = 13 Ci! Hence, the proposed method fails the second criterion which imposes a 1 Ci annual limit.

Subpart L - Records

As always, federal regulation means paperwork! Before getting into the details, a word on retention is in order. The NRC has established only two record retention periods throughout Title 10 CFR. They are summarized in Figure 13.

This Subpart begins with a requirement that licensees use only the "old" system of units – curie, rad, rem. As of 2011, this requirement is still in force. NRC staff state that since reports from licensees are kept electronically (computer data base) the units all have to be consistent and the old units have been chosen for reporting purposes. However, this does not prevent a licensee from keeping records in the new SI units or dual records (both old and new units), as long as only old units are used when reporting to NRC. The new SI units can be included as long as they are in parentheses.

Records which must be maintained until license termination:
Effluent Releases
Waste Disposal
Personnel Doses
Records to be maintained for three (3) years after record is made:
All other required records not listed above

Fig. 13 - Record retention requirements

Note that 10 CFR 35 specifies that surveys by medical licensees for removable contamination must be recorded in units of disintegrations per minute per 100 square centimeters.

Figure 14 summarizes the main record-keeping requirements. Some confusion exists around the requirement for obtaining the prior exposure history of workers. First, this is required only if the worker is likely to exceed 10% of annual limits at the current licensee. If the information is unavailable for any quarter, it should be listed

Subpart L Summary

Purpose:
Details the types and content of required records by a licensee
Chief Provisions:
The following records are required:
Radiation Protection Programs
Program provisions, program audits
Radiation Surveys and Calibrations
Dose from external sources, individual uptakes, air sampling, effluents released
Prior Occupational Dose
Current year and lifetime
Planned Special Exposures
Exceptional circumstances, authorized by?, actions taken?, why PSA necessary, ALARA steps taken, expected doses, actual doses
Individual Monitoring Results
Deep dose equivalent, eye, skin, extremities, CEDE, TEDE, high organ
Individual Members of the Public
Sufficient to show compliance with dose limits
Waste Disposal
Tests of Entrance Controls at Very High Radiation Areas
Date, time, results

Fig. 14 - Summary of Subpart L

as unavailable (NOT recorded as an assumed numerical value) in the personal dose history. But, for administrative purposes, 1/4 of all annual limits must be subtracted from the dose allowed the worker in the present calendar year. Sample Problem 4 clarifies this.

Before a worker can participate in a planned special exposure, PSE, the lifetime dose history must be available. This includes all doses for which there are annual limits - TEDE, organ dose, eye dose, skin dose, extremity dose. The rules state the licensee must "attempt to obtain" this information. To comply, the licensee must ask the worker to provide a signed statement or can request the information from the most recent radiation employer. If the requests are denied, no further action is required except that the worker is not available for PSEs. If the worker falsifies the dose and that leads to an overexposure, the dose is carried as an overexposure on the worker's record BUT the licensee is not held responsible! No penalty is assessed. The appropriate Regulatory Guide cautions "Although not required by the regulations, it is considered good health physics practice to verify the information provided by the individual."

Sample Problem 4

GIVEN:

A new employee begins July 1st. Records for the first two quarters of the year of TEDE and eye dose equivalent are not available. Skin and extremity dose were both reported to be 100 mSv.

FIND:

What dose limits are in effect for this worker through December 31st?

SOLUTION:

One quarter of the annual limit must be subtracted for each unreported quarterly value. Hence, the remaining TEDE = 50 mSv - 2 x 12.5 mSv = 25 mSv. Similarly, the remaining eye dose is 150 mSv - 2 x 37.5 mSv = 75 mSv. Finally, the skin and extremities will each be allowed 500 mSv - 100 mSv (reported) = 400 mSv. *Note that the 25 mSv TEDE and 75 mSv eye doses assumed in this calculation are not recorded in the personal dose history of this worker!*

Subpart M - Reports

From time to time, licensees are required to submit written reports or are required to notify the NRC of specified occurrences. As mentioned in Chapter 14, an immediate telephone report is mandated when a licensee loses (or has had stolen) sources exceeding 1,000 times the Appendix C quantities. This is relaxed to a 30 day phone report when the quantity is 10 times Appendix C. Both reports must be followed up with a written report within 30 days.

Immediate notification is also required if some event involving radioactive material possessed by a licensee causes or "threatens to cause" any of the consequences listed in Figure 15. Note that the telephone report must be followed up with a written report within 30 days.

Besides the reporting of "incidents" the NRC needs to receive a report within 30

Immediate Notification:
TEDE 250 mSv or more
Eye dose of 750 mSv or more to the lens
Skin/extremities dose of 2.5 Gy or more
Release of material in a quantity of 5 ALI

Twenty-four Hour Notification:
TEDE 50 mSv or more
Eye dose of 150 mSv or more to the lens
Skin/extremities dose of 0.5 Sv or more
Release of material in a quantity above 1 ALI

Thirty Day Written Notification:
All of the above events
Doses in excess of occupational limits
Doses in excess of public limits
Dose rates or concentrations > restricted area limits
Dose rates or concentrations > 10 x unrestricted area limits

Fig. 15 - Notification of incidents requirements

days of any planned special exposures. Also, seven different license categories have been singled out for personnel dosimetry reporting. They include nuclear power and research reactors, industrial radiographers, spent fuel facilities and processors of large quantities of radioactive material. These licensees must annually submit a report of individual monitoring for all persons requiring it (likely to receive > 10% of a limit). Note that facilities which voluntarily monitor workers who are not required to be monitored do not have to comply with this rule for those workers. They can partially complete NRC Form 5 and state "monitoring was not required" in the comments section.

Subpart N - Exemptions and Additional Requirements

Consistent with past practice, the NRC reserves the right to grant exemptions from any of the 10 CFR 20 requirements or they may impose additional requirements whenever they feel public safety is at risk.

Subpart O - Enforcement

Finally, the NRC warns us that they are empowered to take enforcement action as a deterrent and to encourage prompt compliance. They can issue a Notice of Violation, obtain a court injunction to prevent a violation of any provision of the standards, or they may obtain a court order for the payment of a civil penalty against licensees. As if this weren't enough, the NRC can also revoke a radioactive materials license and can fine or imprison individuals following their conviction of violations.

10 CFR Part 835

This Part to the Code of Federal Regulations became effective in 1994 and applies to persons who are either "a Department of Energy or DOE contractor employee; an employee of a subcontractor to a DOE contractor; or a visitor who performs work for or in conjunction with DOE or utilizes DOE facilities." However, the regulation specifically states that it is not applicable to the Naval Nuclear Propulsion Program. The format and content of Part 835 rather closely follow those of Part 20. This makes life much simpler for the radiation protection technologist who moves between the DOE and NRC sectors of the business. The Table of Contents for Part 835 appears in Figure 16.

PART 835

Fig. 16 - Table of Contents of Part 835

Except for some definitions, there are few differences from Part 20 in the General Provisions. Ambient air is specifically defined. "Background" radiation includes radiation from consumer products. Collective doses, measured in person-sieverts, are included. In contrast with Part 20, the DOE does not include the elbow or knee in the definition of "extremity." Also, the shallow dose equivalent is not averaged over an area of one square centimeter.

Requirements for radiation protection programs are similar. The chief difference is the more lax DOE specification of 3 year internal audits versus the NRC mandated annual internal audit. Occupational limits are also very similar. The DOE adds a section on determining nonuniform exposure of the skin. There is a factor of five difference in allowed limits for minors - DOE allows only 0.1 rem in a year while NRC allows 0.5 rem. Part 835 is stricter in regard to internal dosimetry. The NRC permits internal doses to be determined from air concentration measurements while the DOE internal dose is "based on bioassay data rather than air concentration values," although some exceptions are listed.

The factor of five mentioned above also applies to the "trigger" level for individual monitoring. The DOE requires monitoring if a worker is likely to receive 0.001 sievert (0.1 rem) annually while Part 20 uses a TEDE of 0.005 sieverts (0.5 rem). DOE facilities are required to do air sampling in areas where a worker might exceed only 2% of the ALI. Real-time

continuous air monitors must be used if the airborne concentration is likely to exceed one DAC value. These monitors are additionally required to have alarm capability. None of these air sampling procedures are spelled out in 10 CFR 20.

The DOE has provided practical information for handling the ever thorny problem of radioactive contamination. First, actual numerical values are given in the regulation for surface contamination levels. The closest that NRC comes to this is a table of values in a Regulatory Guide. Part 835 requires protective clothing be worn to enter areas where removable contamination exceeds the numbers given in Appendix D, listed here in Figure 17. These values "apply to radioactive contamination deposited on, but not incorporated into the interior of, the contaminated item. Where surface

Nuclide	Removable dpm/100 cm^2	Total (Fixed + Removable) dpm/100 cm^2
U-nat, U-235, U-238, and associated decay products.	1,000	5,000
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129.	20	500
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133.	200	1,000
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above*.	1,000	5,000
Tritium Organic Compounds; surfaces contaminated by HT, HTO and metal tritide aerosols.	[Reserved]	[Reserved]

As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency and geometric factors associated with the instrumentation.

***This category includes mixed fission products, including the Sr-90 which is present in them. It does not apply to Sr-90 which has been separated from the other fission products or mixtures where the Sr-90 has been enriched.**

Fig. 17 - Part 835 Appendix D - Surface Radioactivity Values

contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently." Averaging over one square meter is permitted as long as the maximum in any 100 cm^2 section is less than three times the listed value. DOE licensees can also use Appendix D for releasing tools and equipment that have become contaminated. Both the Removable and Total contamination limits must be met. The inaccessible surfaces must likely meet the limits and records describing the property, date, identity of surveyor, ID number of the survey meter and the survey results must be kept. It is also acceptable to release items with fixed contamination above Appendix D levels provided they are used in a controlled area, the removable contamination meets Appendix D and they are "clearly labeled, or tagged to alert personnel of the contaminated status."

Subpart F sets up a program for controlling entry to radiological areas, high radiation areas and very high radiation areas. Methods of control can include entrance control devices, signs, visual alarms, locks and administrative procedures. Part 20 requires posting for these areas but does not address methods for controlling access to radiation areas.

There are some minor differences in posting requirements (Subpart G) between the two Parts. DOE requires an airborne warning sign at 10% of the DAC value while the NRC uses 100% of DAC. Contamination warnings, not addressed by 10 CFR 20, are specified. An area must be posted as a Contamination Area if levels fall between 10% and 100 times the Appendix D values. If the levels are higher than 100 times Appendix D, the area is posted as a High Contamination Area.

Subpart N on accidents and emergencies has no Part 20 counterpart. It sets up a process whereby an overexposed employee can become reinstated to radiation work. Guidelines for control of emergency exposures are presented as a table (Figure 18). Risk/benefit judgements must be made. Rescuers must be volunteers. Doses to 10 rem are specified for protecting major property. More than 25 rems is specified for lifesaving or protection of large populations if the person is "fully aware of the risks

Dose Limit (Whole Body)	Activity Performed	Conditions
5 rems	All.....	.....
10 rems	Protecting major property	Where lower dose limit not practicable.
25 rems	Lifesaving or protection of large populations	Where lower dose limit not practicable.
>25 rems	Lifesaving or protection of large populations	Only on a voluntary basis to personnel fully aware of the risks involved.

Fig. 18 - Part 835 Guidelines for Control of Emergency Exposures

involved." Nuclear accident fixed dosimeters must be installed in areas where workers might receive a dose in a criticality accident. Individual personal criticality dosimeters must be worn by anyone entering these areas.

Finally, there are some areas of radiation protection addressed by Part 20 which have no corresponding equivalents in Part 835. These include internal intake through wounds or skin absorption, hospital procedures, labeling of containers, procedures for receipt and opening of packages, waste disposal and regulatory enforcement.

In addition to the two Parts of 10 CFR discussed in detail (Part 20 and Part 835) radiation protection technologists should have a passing acquaintance with the following sections:

- 10 CFR 19, Notices, Instructions and Reports to Workers
- 10 CFR 21, Reporting of Defects and Noncompliance
- 10 CFR 34, Licenses for Radiography and Radiation Safety Requirements for Radiographic Operations
- 10 CFR 50, Domestic Licensing of Production and Utilization Facilities
- 10 CFR 61, Licensing Requirements for Land Disposal of Radwaste
- 10 CFR 71, Packaging and Transportation of Radioactive Materials

ICRP Publication 103

Radiation workers outside of the United States frequently fall under the authority of the International Commission on Radiological Protection recommendations. These become legally binding in a given country through adoption outright by that country's legislative body or by rewriting the recommendations and then placing them in the national code.

The most recent ICRP-developed comprehensive radiation safety program is found in their Publication 103. These recommendations were adopted by the Commission in 2006 and published in 2007. The revision

Occupational Limits

Effective dose of 20 mSv/yr averaged over 5-year periods, no year > 50 mSv
150 mSv annual equivalent dose to lens of eye
500 mSv annual equivalent dose to skin
500 mSv annual equivalent dose to hands and feet
Equivalent dose of 2 mSv to abdomen of declared pregnant worker
Intake < 1/20 ALI for declared pregnant worker

Public Limits

Effective dose of 1 mSv in a year averaged over 5 years
15 mSv annual equivalent dose to lens of eye
50 mSv annual equivalent dose to skin

Fig. 19 - Summary of ICRP dose limits

to ICRP 60 came after a 17 year interval.

The newest version utilizes the same dose limits for occupational workers and for the members of the public as the previous ICRP 60. See Figure 19. The whole body occupational dose average of 20 mSv/year compares to 50 mSv/year in the U.S. regulations, 10 CFR 20. Based on the most current data on biological risk, the Commission reached the conclusion that the lifetime dose to a radiation worker should not exceed 1 Sv. At that dose, the worker was estimated to experience an average loss of life expectancy of 6 months. Dividing the 1 Sv by a 50 year working life gives the ICRP occupational dose limit of 20 mSv in a year. The actual limit is stated to be "20 mSv per year, averaged over defined periods of 5 years with the further provision that the effective dose should not exceed 50 mSv in any single year." In the case of a declared pregnant worker, the ICRP recommends limits of 2 mSv to the woman's abdomen and 1/20 of the ALI for the remainder of the pregnancy.

Publication 103 continues the use of a Dose and Dose Rate Effectiveness Factor with a numerical value of 2. Also, carried over unchanged was the estimate of radiation induced fatal cancer risk. The value is still estimated by the ICRP to be about 5×10^{-4} per rem (or 5% per Sv). It is interesting to note that the U.S. NRC used an estimated fatal cancer risk of 1.25×10^{-4} per rem to establish the limits in the 10 CFR 20 regulations. The International Commission has been following the controversy over the Linear-Nonthreshold hypothesis. They recognize the shortcomings of the LNT idea, but feel that more research needs to be done before an alternative hypothesis has sufficient support to qualify for their recommendation.

One of the major changes in the new ICRP report was recommendation of new tissue weighting factors, w_T. The new values introduced now have explicit values for 14 organs (versus the 6 given in the current 10 CFR 20). The ICRP 103 values were listed in Chapter 5, but, for the sake of completeness, they are reproduced here in Figure 20.

Another change that should be noted is that the ICRP introduced new radiation weighting factors, w_R. The explicit values currently recommended were tabulated in Chapter 5, Figure 8.

The basic ICRP system of protection is based on three overriding principles. Justification means that every radiation exposure must have a benefit larger than the detriment it caused. Optimization has the same

Tissue	w_T Value	Tissue	w_T Value
Bone Marrow	0.12	Breast	0.12
Colon	0.12	Lung	0.12
Stomach	0.12	Gonads	0.08
Bladder	0.04	Esophagus	0.04
Liver	0.04	Thyroid Gland	0.04
Bone Surfaces	0.01	Brain	0.01
Salivary Glands	0.01	Skin	0.01
Remainder	0.12		

Fig. 20 - The 2007 ICRP Publication 103 tissue weighting factors

definition as ALARA in the U.S.A. Limitation means that exposure of individuals must be subject to numerical limits to prevent unnecessary risk.

The system of protection is applied in three types of exposure situations. Exposures in the licensed occupational setting are designated Planned Exposure Situations by the ICRP. Radiation accidents are designated Emergency Exposure Situations. Finally, situations such as public exposure to radon gas are classed as Existing Exposure Situations. Consistent with current U.S. practice, the ICRP only recommends dose limits for the occupational and public categories. They state that medical exposure provides a direct benefit to the patient and that numerical limits beyond the recommended ALARA practice might be detrimental to the patient.

However, ICRP 103 did break ground in a new area. They suggest a framework for future recommendations in radiation protection of our environment. This could conceivably lead to definition of "Reference Animal" or "Reference Plant" at some future date. No numerical values were put forth in Publication 103.

NORM Regulations

Interest continues to grow regarding the regulation of activities involving naturally occurring radioactive material. Louisiana (LA) and Texas (TX) already have statutes on the books which regulate NORM in those states. Model regulations have been proposed by the Conference of Radiation Control Program Directors (CRCPD), a group of state radiation control regulatory professionals. These do not carry the force of law.

All three of the documents just mentioned have many similarities. They all begin with a statement of applicability and a list of exemptions to the standards. The LA regulation specifically mentions that the rules apply to pipe scale and soil contaminated with scale during oil field operations. Texas has chosen not to adopt the exemption for Brazil nuts found in the LA and CRCPD versions. Both TX and LA have included a numerical exposure rate limit for determining if NORM contaminated equipment or materials are regulated. Equipment and materials showing a maximum exposure rate at "any accessible point" less than 50 microroentgens per hour including background are exempt from all the NORM regulations. The measurement must be made with a survey meter accurate to ± 20% and calibrated within the last 6 months (in LA) or 12 months (in TX). The TX and CRCPD versions also include a table of acceptable surface contamination levels for NORM. Equipment exceeding the table values cannot be released for unrestricted use. Texas allows surface contamination to reach 5,000 dpm/100 cm^2 average, 15,000 dpm/100 cm^2 maximum and 1,000 dpm/100 cm^2 removable. The CRCPD version uses these numbers for beta-gamma emitters and all uranium isotopes. Lower values are proposed for transuranics and other alpha emitters. Texas has also added one other unique dose rate limit. Even if the above mentioned 50 μR/hr is exceeded, handling or processing of NORM contaminated materials is authorized if the dose rate is less than 2 millirem/hr at 18 inches.

Although the LA version does not address specific licensure for NORM activities, the other two versions do so in a very similar manner. A specific state license is needed to manufacture or commercially distribute any material or product containing NORM above the exemption limits. An applicant must meet requirements similar to

those specified for persons applying for an NRC or agreement state byproduct radioactive materials license. For products containing NORM, it must be shown that routine use of the product will not deliver a whole body dose above 0.05 mSv (5 mrem) in a year. Extremities can receive 75 mrem in a year. Doses to personnel servicing or distributing the product are allowed to be 100 times larger.

Epilogue

This concludes the main body of the text. Three Supplemental Chapters are included for the more advanced reader. Appendices and indices are also found at the end of the book. It is the author's sincere hope that the book has met the expectations of the reader. If this is not the case, he would gratefully receive written comments to that effect (dr_dan@pacificrad.com). Best of luck in your practice of radiation protection technology, and always keep your doses ALARA!

Problem Set

1. Describe the differences in the purposes of the ICRU and the ICRP.

2. Are the recommendations of the ICRP and the NCRP legally binding?

3. What is the role of the Nuclear Regulatory Commission?

4. Describe the difference between a regulation and a regulatory guide.

5. How is an agreement state formed? What functions does this transfer to the state?

6. Are ALARA and the "Below Regulatory Concern" concept logically compatible? How does the idea of a "radiation deficiency" at low radiation levels affect the BRC concept?

7. Briefly summarize the concept of "comparable risk" as applied to radiation standards setting. Has it been achieved?

8. A twenty-three year old radiation worker with a cumulative lifetime radiation dose of 3.4 rem is exposed during one calendar quarter to 2.8 rem TEDE. Does this constitute a technical overexposure for this worker? Why or why not?

9. Is a radiation badge required for a U.S. worker who will likely receive 120 mrem per month on a job scheduled to take four months to complete?

10. What was the average civil penalty (fine) imposed by the NRC on U.S. licensees in 1992?

11. Briefly list some topics that should be included in a talk given to new

employees that will occasionally be working in a restricted area having some sections designated as “radiation areas.”

12. How much dose can a person under 18 years of age receive under the regulations?

13. How is a “very high radiation area” protected differently from a high radiation area?

14. A university plans weekly sewer disposal of 15 mCi of C-14, 15 mCi of P-32 and 10 mCi of S-35. All are in water soluble form. Is this permissible under 10 CFR 20?

15. What notification category would a release into an occupiable area of 11 MBq of Cs-137 fall under according to 10 CFR 20?

16. How long must records of the testing of the entrance controls at a very high radiation area be kept by a licensee?

17. A technician measures the dose rate to be 85 microsieverts per hour at 1 meter from a point source. Does this area have to be posted as a high radiation area?

18. A worker has a lifetime PSE dose of 120 mSv and a complete dose history on file with the current employer. Her current year TEDE is 65 mSv. Is this worker available for another PSE this year? If so, what would be her dose limit for the PSE?

S-1. A radiation worker receives 1 rem of deep dose equivalent from external exposure and also has an internal uptake of 0.9 μCi of ^{131}I and 6 μCi of ^{137}Cs in an accident. Calculate the percentage of applicable limits these doses correspond to. The ALI’s for I-131 and Cs-137 are 50 μCi and 200 μCi respectively.

S-2. A DOE survey meter is contaminated with I-131. Measurements show the removable contamination at 180 dpm/100 cm^2 and the total contamination at 1800 dpm/100 cm^2. Can this instrument be released for unconditional use? Can it be transferred to another controlled area in the plant?

S-3. Under what conditions would the worker in Problem 8 above meet ICRP recommended dose limits?

Other Resources

1. “Title 10, Energy, Code of Federal Regulations, Part 20,” Office of the Federal Register, U.S. Government Printing Office, Washington, D.C.

2. “Title 10, Energy, Code of Federal Regulations, Part 835,” Office of the Federal Register, U.S. Government Printing Office, Washington, D.C.

NOTE: Both of these CFR titles are now available free online at http://www.nrc.gov/reading-rm/doc-collections/cfr/.

3. “The Recommendations of the International Commission on Radiological Protection,” ICRP Publication 103,” ICRP, Elsevier Ltd., Oxford, UK, 2007.

4. “Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM),” NUREG 1575, Revision 1, U.S. Nuclear Regulatory Commission, Washington, DC, August, 2000.

5. “Consolidated NMSS Decommissioning Guidance,” NUREG-1757, Volume 2, U.S. Nuclear Regulatory Commission, Washington, DC, 2003.

NOTE: This document is the “instruction manual” for the MARSSIM Manual, Other Resources Item 4. It is 524 pages long.

6. “Bulletin on Management of Naturally Occurring Radioactive Materials (NORM) in Oil & Gas Production,” API Bulletin E2, First Edition, American Petroleum Institute, Washington, DC, 1992.

SUPPLEMENTS and APPENDICES

S-1

Reactor Health Physics

Outline of this Chapter

Chapter Summary

This first supplemental chapter covers radiation protection practices at reactor sites. It provides considerably more detail than the main text.

The chapter begins with material from the closely related areas of reactor physics and reactor engineering. The fission process is covered, and then reactor control follows. These principles are then illustrated by example. The design features and limitations of PWRs, BWRs, CANDUs and RBMK-1000s are covered in detail.

Next, the actual health physics aspects of reactors are introduced. The initial material focuses on the power reactor setting. Administrative functions such as staffing, worker training, self-assessment programs and use of performance indicators are covered first. The focus then shifts to controlling doses to personnel through ALARA design, access control, surveys and dose measurement, and work-planning. Control of radioactive materials is then covered. Finally, some observations and predictions are made on the future directions of the nuclear power industry, particularly as a result of deregulation trends.

Finally, the chapter concludes with a look at the “other” side of reactor health physics. Research and test reactors are still active at numerous sites around the USA and the rest of the world. Some of the unique health physics problems associated with these reactors are discussed along with routine health physics practices.

Reactor Physics

Physics of Fission

Fission can occur if the mass-energy of the fissioning nucleus is greater than the sum of the mass-energies of the fission products and emitted neutrons. In practice, this limits fission to the heavy end of the periodic table of the elements. As discussed earlier, fission is accompanied by the release of a large amount of energy, typically over 200 MeV per atom. Most of the energy from fission is absorbed in the core and primary shielding of the reactor. The exception is the energy from neutrinos, which only accounts for about 5% of the total energy release.

If we assume an average value for all fissile nuclides of 190 MeV per fission, then the fission rate to produce one watt of thermal power can be calculated as follows:

$$1\ W \times 1\ J/W\text{-sec} \times 10^7\ erg/J \times 0.62 \times 10^6\ MeV/erg \times 1\ fission/190\ MeV$$
$$= 3.3 \times 10^{10}\ fissions/sec\ per\ watt.$$

The fission of 1 gram of ^{235}U produces about 1 $MW_{thermal}$-day of power – an energy release equivalent to 21,000 tons of TNT! On the average, about 2.5 neutrons are released per fission, with an average energy of about 2.5 MeV. More than 99% of these neutrons are released at the instant of fission. They are given the name "prompt neutrons." The remaining neutrons are released by the radioactive fission fragments a short, but incredibly important, time later. These "delayed neutrons" account for 0.65% of the total for ^{235}U (but only 0.21% for ^{239}Pu). The decay half-lives range from 0.2 second to 56 seconds for these neutron emitters.

Criticality in a reactor is attained when at least one of the neutrons released in fission causes a second nucleus to undergo fission. The effective multiplication factor, k_{eff} discussed in Chapter 8 is a measure of criticality. The amount by which k_{eff} exceeds 1.000 is termed the excess reactivity, Δk. If there are n neutrons in the core in one generation, there will then be n Δk additional neutrons in the next generation. The algebraic relation which calculates the growth of neutrons with the passing of time is

$$\mathbf{n = n_o\ e^{t/T}.} \qquad \textbf{[Eqn. 1]}$$

In this equation, T is called the reactor period. It is the time to increase reactor power by "e" times, i.e., by a factor of 2.718 times.

Reactor Period

It is possible to calculate T rather simply; $T = L/\Delta k$. In this relationship, L is the average lifetime of the neutrons released in fission. For the prompt neutrons, L =

0.001 second. Thus, for example, if the excess reactivity in the reactor core was 0.005, then the prompt neutrons would cause the reactor to have a period of T = 0.001/0.005 = 0.2 second.

Under these conditions, in a time span of only one second (i.e., t = 1), the power level (neutron flux) would increase by $e^{1/.2} = e^5$ = 150 times!! The reactor would be extremely hard to control. However, this calculation ignores the delayed neutrons. They have an effective lifetime of 0.082 seconds. Thus, using the same example of excess reactivity of 0.005, the period now is 0.082/0.005 = 17 seconds. Clearly, the delayed neutrons are vital to the control of the reactor.

In the event the excess reactivity ever exceeds 0.0065, the reactor is said to be "prompt critical." In this case, the chain reaction is dependent only on the prompt neutrons. There are enough of them that the reactor is critical without any delayed neutrons. This causes the reactor period to be very short (as seen in the example in the previous paragraph). This condition must always be avoided. Prompt criticality led to the explosion of the Chernobyl reactor.

A convenient way to measure excess reactivity is in dollars and cents. The "conversion factor" is that \$1 = a Δk of 0.0065, exactly the amount to make the reactor prompt critical. Of course, this definition is numerically valid only for ^{235}U fueled reactors, since the delayed neutron fraction is different for other fuels. For example, in a ^{239}Pu fueled reactor, \$1 = Δk of 0.0021. Thus, if a control rod extraction inserts 10¢ of reactivity into a reactor running with k_{eff} = 1.0000, the reactor is 1/10 of the way to operating prompt critical.

Reactivity Coefficients

The rate at which the core temperature or volume of voids caused by boiling water affects the reactivity is measured by the temperature coefficient and void coefficient, respectively, of the reactor. The temperature coefficient of the core is the ratio of the change in reactivity produced by a temperature change of one degree. It is the sum of all of the individual temperature coefficients of the fuel, moderator and coolant. The overall temperature coefficient can be positive or negative, depending on reactor design. Obviously, a positive coefficient is very hazardous - as the reactor heats up, the reactivity would increase correspondingly, thus, raising the temperature even further. This "positive feedback" situation could cause the reactor to go out of control. All licensed reactors in the United States have negative temperature coefficients.

The void coefficient is the reactivity change produced by a one percent change in the liquid void volume. A boiling moderator or coolant clearly has less density than when in liquid form. The lower density is a poorer neutron moderator so this tends to reduce k_{eff}, shutting the reactor down ("negative feedback"). In some types of reactor design, it is possible to actually have a positive void coefficient. Remember that this was the physical cause of the explosion of the Chernobyl reactor. If less graphite is

used, the void coefficient in the RBMK family of Soviet reactors can be made negative. At present, all remaining RBMK reactors have been modified so that this is the case.

Fission Product Poisons

As the reactor operates at power, a variety of different fission product nuclides build up in the fuel elements. Some of these nuclides are strong neutron absorbers, and so they are termed "poisons." The chief culprits are ^{135}Xe and ^{149}Sm. About 1.5% of the fissions of U-235 result in one of these poison atoms being produced. The Xe has a half-life of 9 hours, while the Sm isotope is stable. Even though they reduce core reactivity by capturing neutrons, the capture itself induces a (n,γ) nuclear reaction which removes the poison. Thus, the poison level reaches an equilibrium, or constant value under steady state reactor power conditions.

Upon shutdown, the poison concentration continues to increase because both the ^{135}Xe and ^{149}Sm are also formed by radioactive decay of other fission products, I-135 and Pm-149 respectively. The maximum poison concentration is reached about 10 hours after shutdown. As a result of the negative reactivity introduced by poison buildup, a delay period results following shutdown, during which time the reactor cannot be restarted due to excess Xe. In a nuclear power reactor, this "xenon precluded start-up" time is about 80 hours.

Reactor Engineering

Power Reactor Design Types

The PRESSURIZED WATER REACTOR, or PWR, is fueled with uranium enriched to between 4% - 5% with U-235. The uranium is in the form of ceramic oxide pellets enclosed in metal tubes (cladding). Water is used as both a coolant and moderator. The primary loop, consisting of water flowing continuously between the fuel inside the pressure vessel and a steam generator, is kept at high enough pressure (approximately 2,200 psi) to prevent boiling. The superheated water (about 600° F) flows in pipes through the steam generator where some of its heat energy is transferred by conduction to water in the secondary loop which flows between the steam generator and the turbine (Figure 1). Because the pressure is lower in the secondary loop, the water turns to steam and passes through the turbine. The typical thermal efficiency (fraction of the heat produced which is recovered in the form of electrical energy) is about 33%. Most of the recently constructed PWRs in the USA have a rated electrical capacity of about 1,100 MW_e. This means they produce about 3,300 MW of thermal power at full capacity. Fuel burnup is typically 15,000 MW-days/ton. A fuel loading is typically 100 tons. Figure 2 is a photograph of the Indian Point Nuclear Station on the Hudson River above New York City. The two operating units use the river

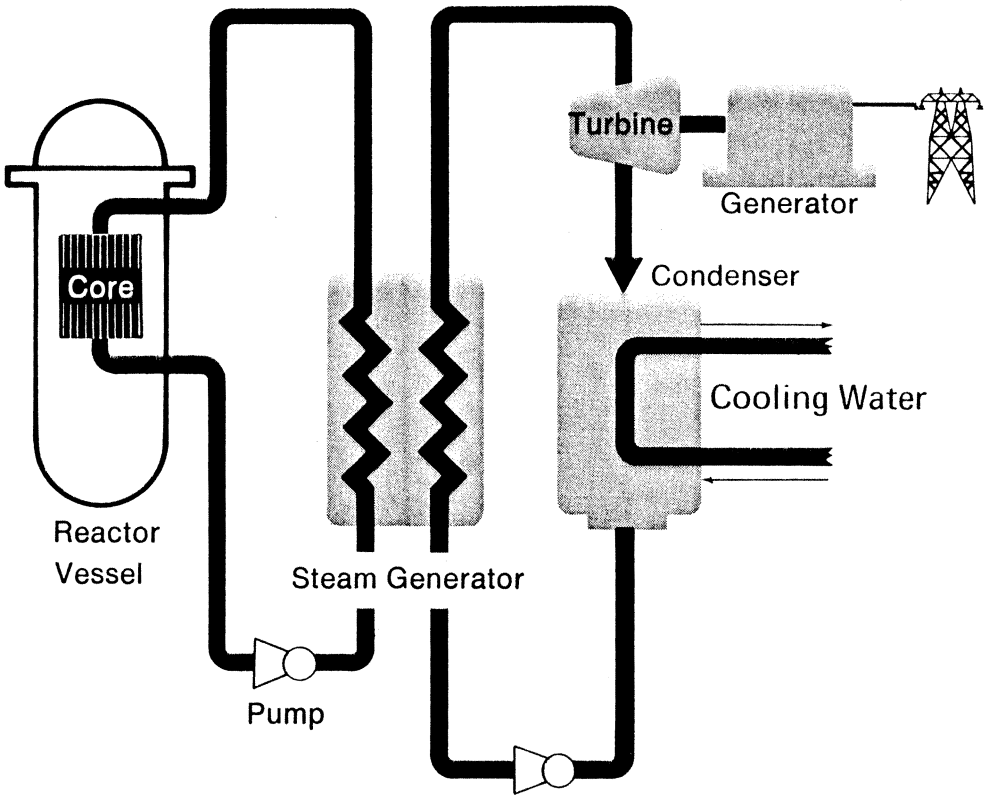


Fig. 1 - Design features of a PWR

Fig. 2 - The Indian Point PWR station on the Hudson River in New York

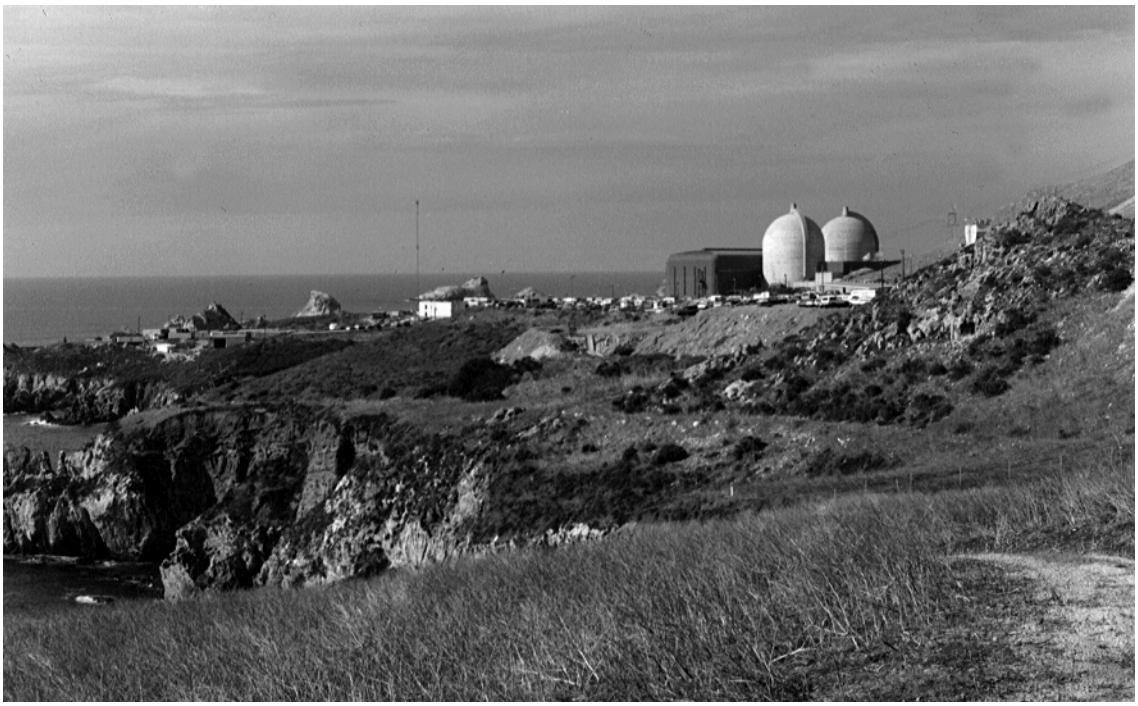


Fig. 3 - The Diablo Canyon PWR station on the Pacific Ocean in California

for cooling. Figure 3 shows the Diablo Canyon Station which uses water from the Pacific Ocean for cooling.

The BOILING WATER REACTOR is characterized by having a single coolant loop (Figure 4). The water which passes through the fuel in the core (at about 550°) is allowed to boil and the resulting steam directly drives the turbine. Typical thermal efficiencies for BWRs run 31-33%. Burnup of fuel is again typically 15,000 MW-days/ton. Recent U.S. BWRs have a rated electrical capacity of around 1,250 MW_e.

The HIGH TEMPERATURE GAS REACTOR, HTGR for short, is a design favored by European reactor engineers. In the U.S., no plants of this type are operational any longer. Fort Saint Vrain, near Denver, Colorado, was the last U.S. licensed HTGR power reactor. In contrast to PWRs and BWRs which use water as a coolant, an HTGR uses helium or carbon dioxide gas. The moderator is typically graphite. The fuel is a mixture of uranium and thorium carbide. The ^{232}Th in the fuel captures neutrons to produce fissionable ^{233}U. The cooling gas enters the core at about 600° F and leaves at about 1,400° F. This significantly higher exit temperature, compared to PWRs and BWRs, allows a greater fraction of the heat energy to be recovered usefully as electrical energy - typically HTGRs have a 39-40% thermal efficiency. Additional generic advantages of HTGRs include non-corrosive coolant, neutron transparency of the coolant gas and less risk of a loss of coolant accident.

The LIQUID METAL FAST BREEDER REACTOR falls into a completely different design family, fast reactors. Only reactors which utilize neutrons above 100 keV are

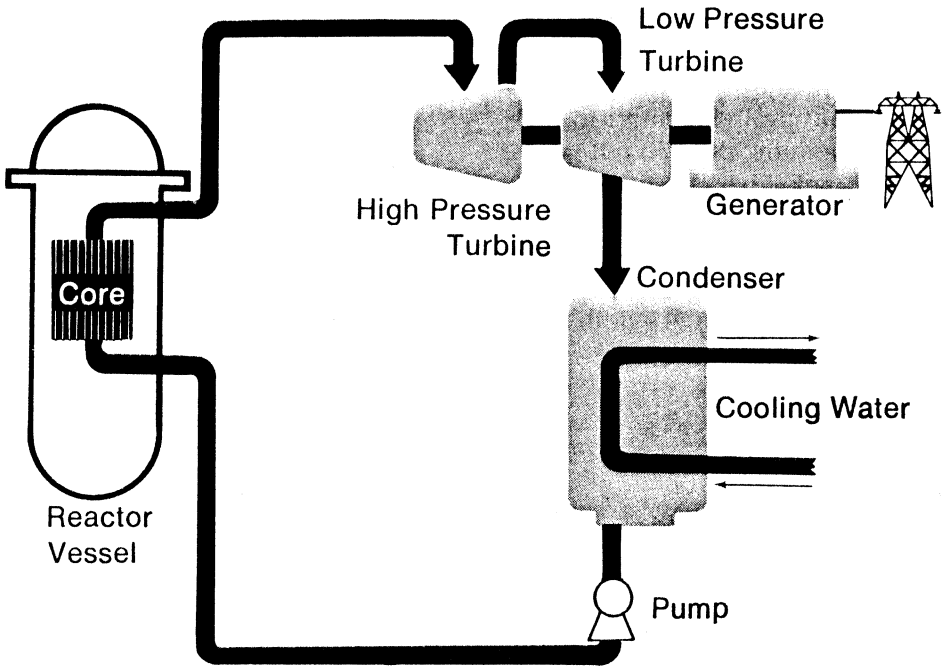


Fig. 4 - The main design features of a BWR

part of the fast reactor family. A three-loop design is employed. The primary loop contains liquid sodium metal circulating through the core as shown in Figure 5. The heat is transferred to sodium in a secondary loop which finally converts steam to water in the third loop. This three-loop design reduces the chance of releasing radioactive sodium in the event of a loop rupture. The high boiling point of sodium (880° C versus 100° C for water) makes it possible to operate the reactor at rather low pressure. This advantage is offset by the high chemical reactivity of sodium metal if it comes in contact with water. Special firefighting techniques must be used.

One generic advantage is the absence of a moderator. This allows construction of much smaller cores compared to thermal reactors. Also, the chance of a fast neutron being captured in the reactor is much less than for thermal neutrons in a conventional reactor. Practically, this means a much lower level of activation products. Also, the fission products themselves capture fewer fast neutrons so fission product poison production is reduced. The direct benefit of this property is that much higher burnup of fuel is possible in the fast reactor - typically 50,000 MW-days/ton versus only 15 - 20,000 MW-days/ton in a thermal reactor.

The capture probability of ^{238}U for fast neutrons is relatively high, leading to the production of Pu-239. This is what makes the design a “breeder reactor.” In principle, a fast reactor can economically recover up to 75% of the potential energy contained in uranium. This sharply contrasts with thermal reactors which “burn” only about 1-2% of the contained energy. The breeder reactor may prove viable at a point

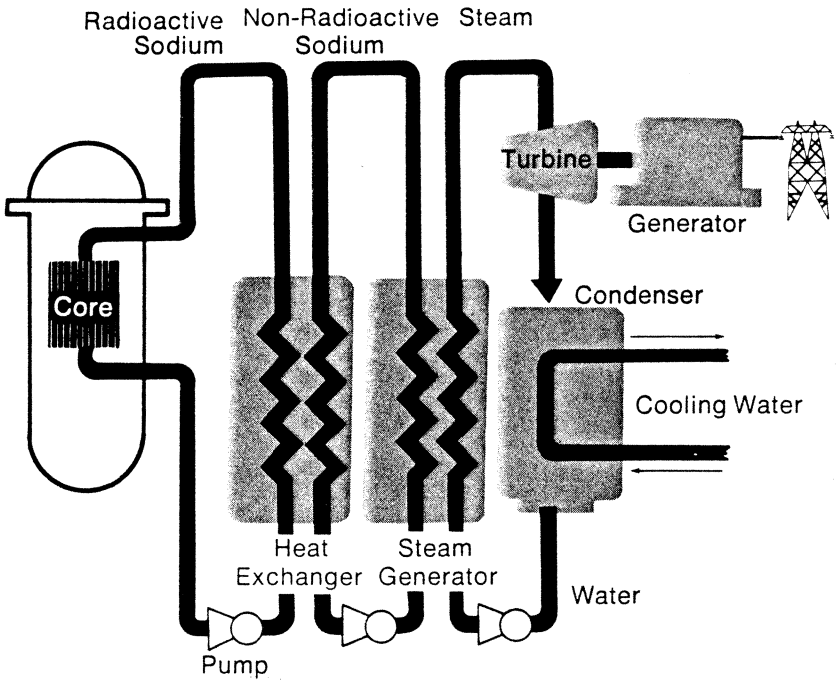


Fig. 5 - The main design features of a LMFBR

when the world's supply of uranium is close to exhaustion. Over a thirty year design life, a 1000 MW_e LMFBR would require 2300 kg of plutonium and would produce 7700 kg of plutonium. The thermal efficiency approaches 42%.

The world's first full-scale breeder reactor, the Superphenix located in France's upper Rhone valley, achieved full power operation in the fall of 1986. It was a 1200 MW unit that was built in about 9 years. When the project was conceived, the Superphenix was to be the prototype for a series of European breeder reactors. Unfortunately, the total project cost exceeded expectations. Electricity generated by the Superphenix was about 2.2 times more expensive than power obtained from a French light water plant. In addition, the continuing availability of low cost uranium has reduced the need for plutonium. Finally, the plutonium produced by the Superphenix required a special reprocessing plant which had not been built. In 1997, the plant was shut down permanently due to sodium leaks, litigation and low performance.

HEAVY WATER REACTORS using deuterium oxide (D_2O) as moderator constitute about 7% of the power reactors in use world-wide. None have been built in the USA but plants are operational in Canada, West Germany, India, Argentina, Romania, South Korea and The Czech Republic. Atomic Energy of Canada, Ltd. (AECL) manufactures the largest reactor of this type, a CANDU reactor of 790 MW_e capacity. The CANDU design is shown in Figure 6. The German heavy water plant runs at about 320 MW_e. The design details and the advantages and disadvantages of the CANDU plants will be covered later in this Chapter.

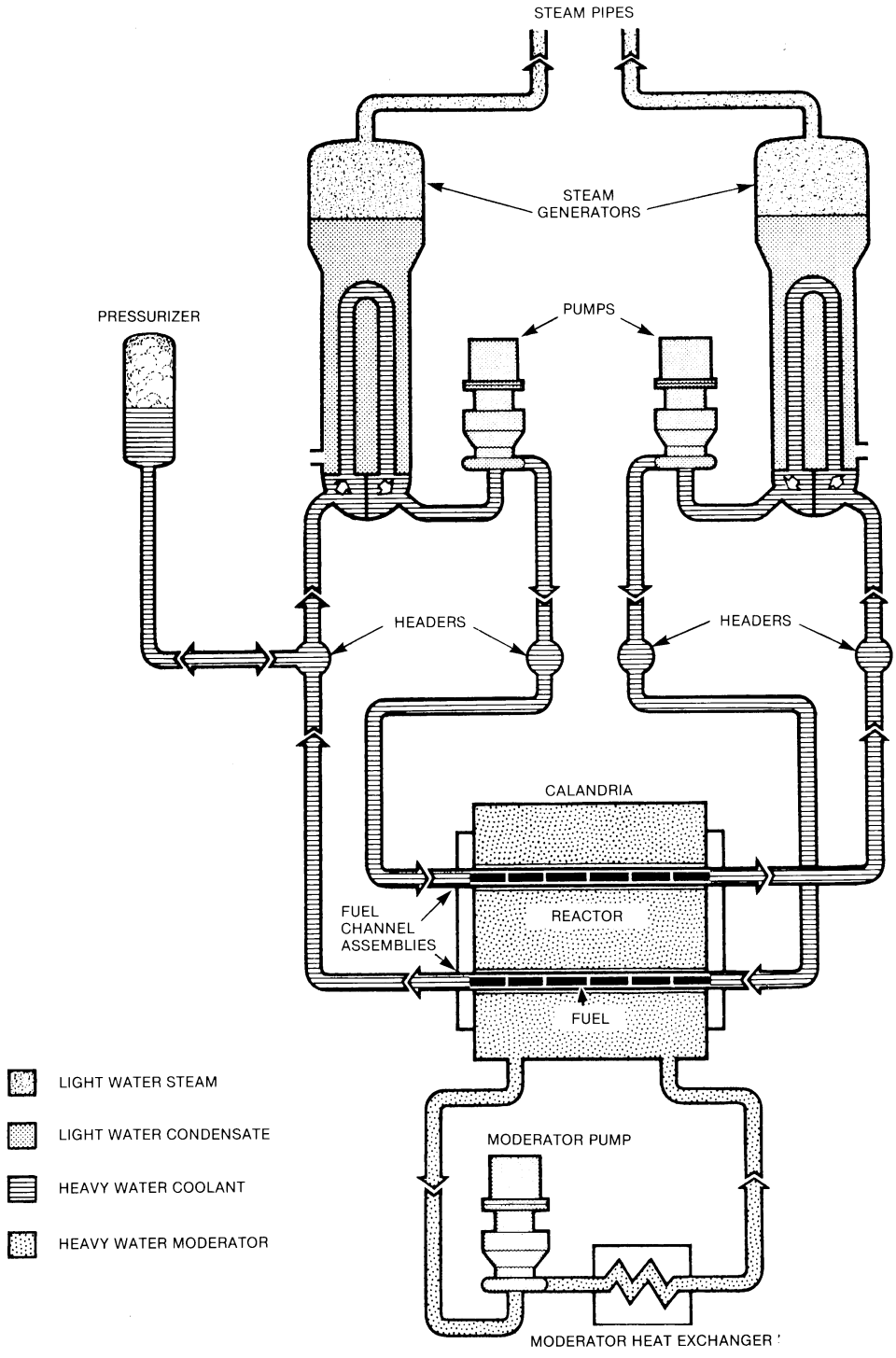


Fig. 6 - The main design features of the CANDU plant

PWR Features and Characteristics

About two-thirds of the U.S. power reactors are of the PWR type using light water. Although the Canadian and German heavy water reactors use the PWR principle, they have been covered elsewhere in this Chapter.

The pressure vessel of a PWR usually consists of a hemispherical bottom welded to a cylindrical shell as shown in Figure 7. It is sealed with a hemispherical

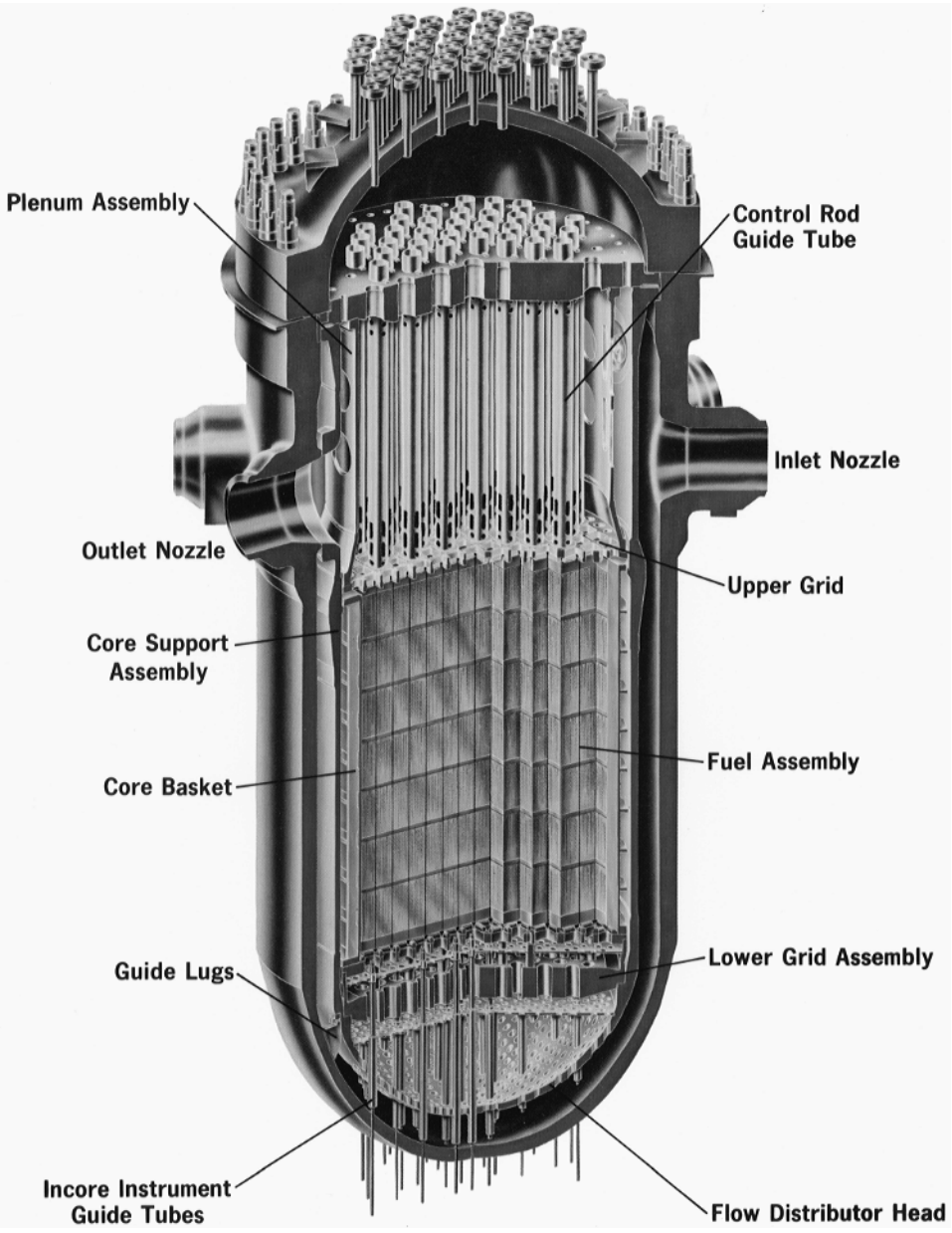
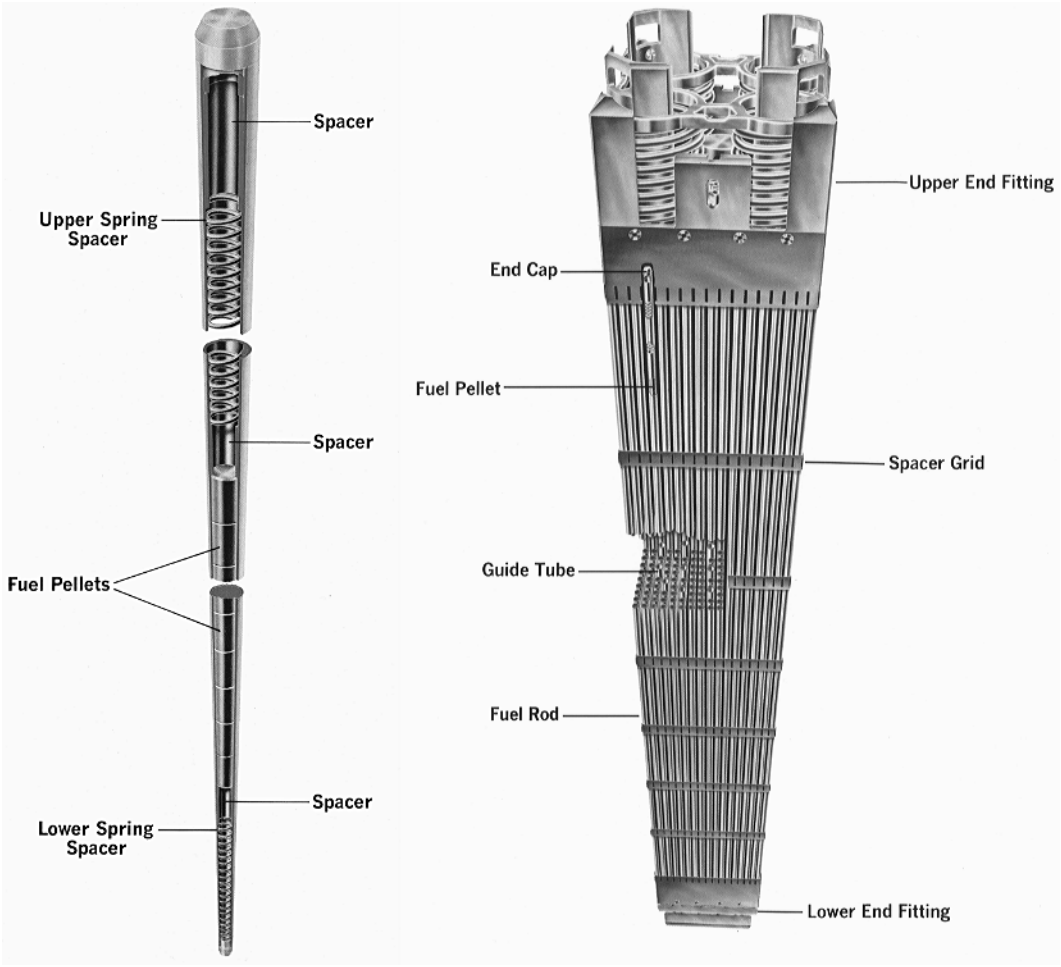


Fig. 7 - A PWR pressure vessel and internals

head that is bolted to the shell. Usually the vessel is made of steel with a stainless steel cladding. The wall is about 10 inches thick in a 1000 MW_e reactor and the assembly weighs several hundred tons. The main penetrations into this sealed vessel are for the inlet and outlet coolant lines, control rod drives and emergency cooling nozzles. These penetrations are all physically located above the top of the fuel so that a major penetration rupture won't necessarily expose the fuel as a result of a water leak.

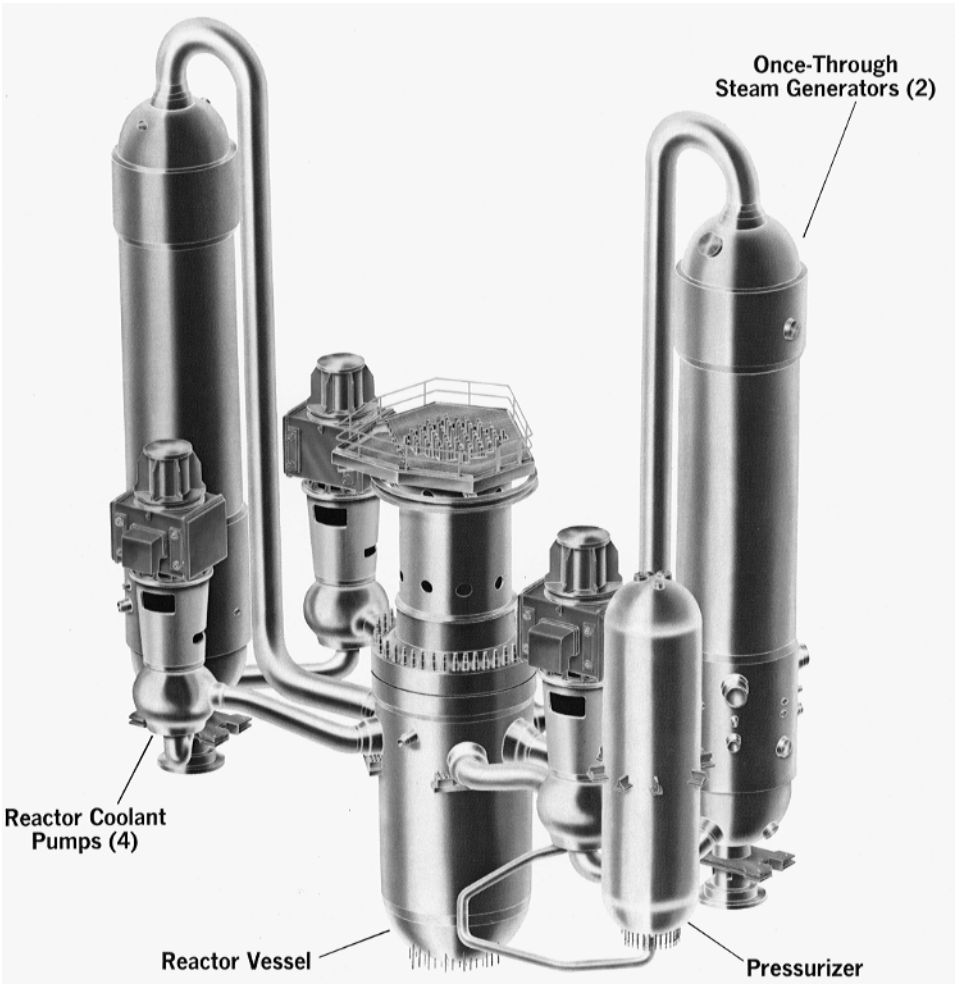
The fuel is typically contained in about 200 assemblies, each 10-14 feet long containing 150 to 250 individual rods (see Figure 8). The total weight of the uranium oxide is around 90 tons. Each rod contains a stack of pellets inside Zircaloy cladding (tubes made of a zirconium-aluminum alloy that don't absorb many neutrons).



Courtesy, Babcock & Wilcox

Fig. 8 - A PWR fuel assembly and fuel rod

The piping in the primary coolant system is as compact as possible to prevent excessive heat losses and to reduce the needed volume of coolant. Pipes are covered with thermal insulation. The coolant pumps (Figure 9) are typically vertical, constant-speed shaft-sealed units with water-cooled motors. The motor is designed for many years of continuous duty. A heavy flywheel makes sure that the "coast-down time" is



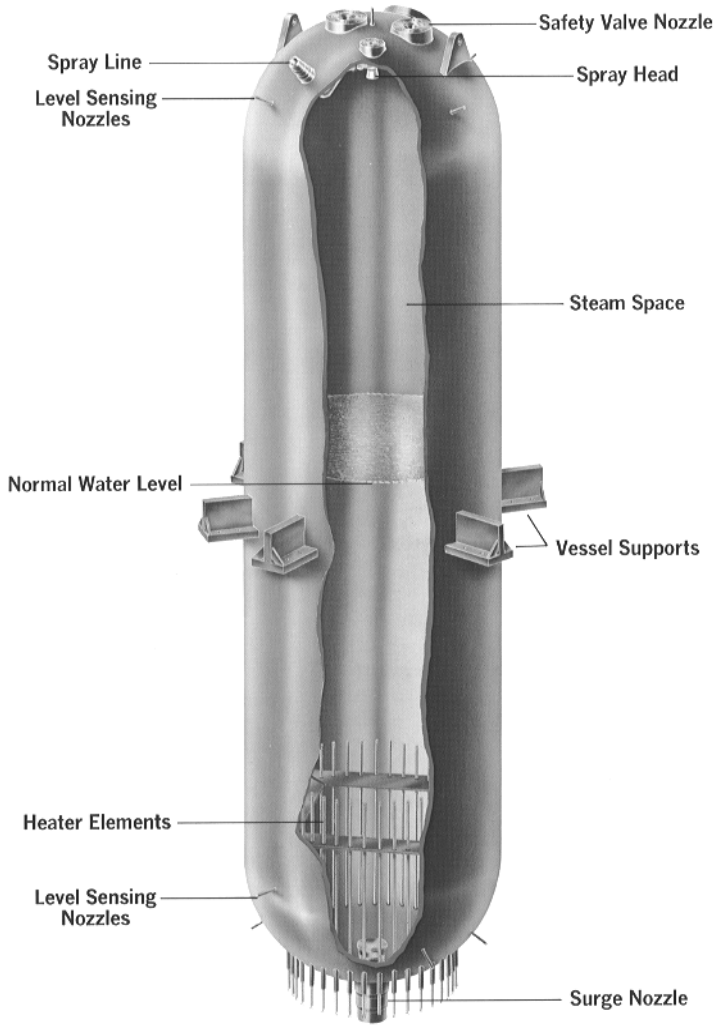
Courtesy, Babcock & Wilcox

Fig. 9 - Major PWR components

long in the event of power loss. Each primary pump motor is rated around 9,000 horsepower in a PWR plant.

The pressurizer is a tank structure designed to keep constant pressure on the coolant as it circulates. It has about 30 internal electric heaters and built-in spray nozzles to heat or cool the water as needed. Figure 10 is a cutaway view. In the case of overpressure, the top safety relief valve opens to vent into a quench tank. (Recall, it was the failure of this valve to close following a normal vent cycle that led to the emergency at Three Mile Island Unit 2.) The pressurizer is designed to hold the water at 600° F under 2250 psi pressure in the primary loop.

The steam generator (Figure 11) functions as a heat exchanger between the primary coolant loop and the secondary (steam) loop. Combustion Engineering and Westinghouse both use a "U-tube" design while Babcock & Wilcox uses a once-through design. Feedwater is pumped, under pressure, from the main condenser on the turbine to a distributing manifold in the steam generator. It is heated by primary



Courtesy, Babcock & Wilcox

Fig. 10 - A PWR pressurizer

coolant water flowing through tubes and is vaporized to steam. The water/steam mixture passes through a maze of moisture separators and finally a dryer stage which removes 99.7% of the water. The dry steam then is directed to the turbine and main condenser which is cooled by a separate loop fed from an ocean, pond or cooling tower. Steam pressure is usually 600 - 1000 psi.

The steam generator is the chief "heat sink" for the energy output of the reactor during operation. It, also, must continue to remove heat from the core following normal or unintentional (scram) shutdown. Without this heat removal pathway, the core temperature would shoot up to unsafe levels. Emergency cooling systems are activated if the steam generator fails to perform its critical function.

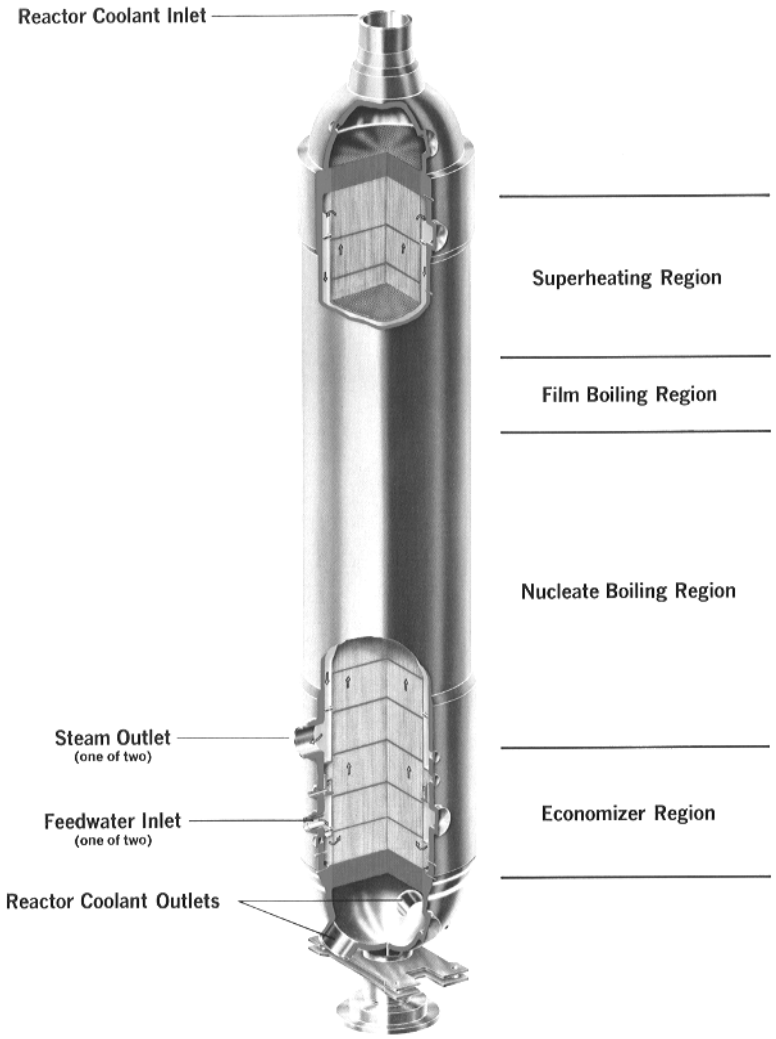
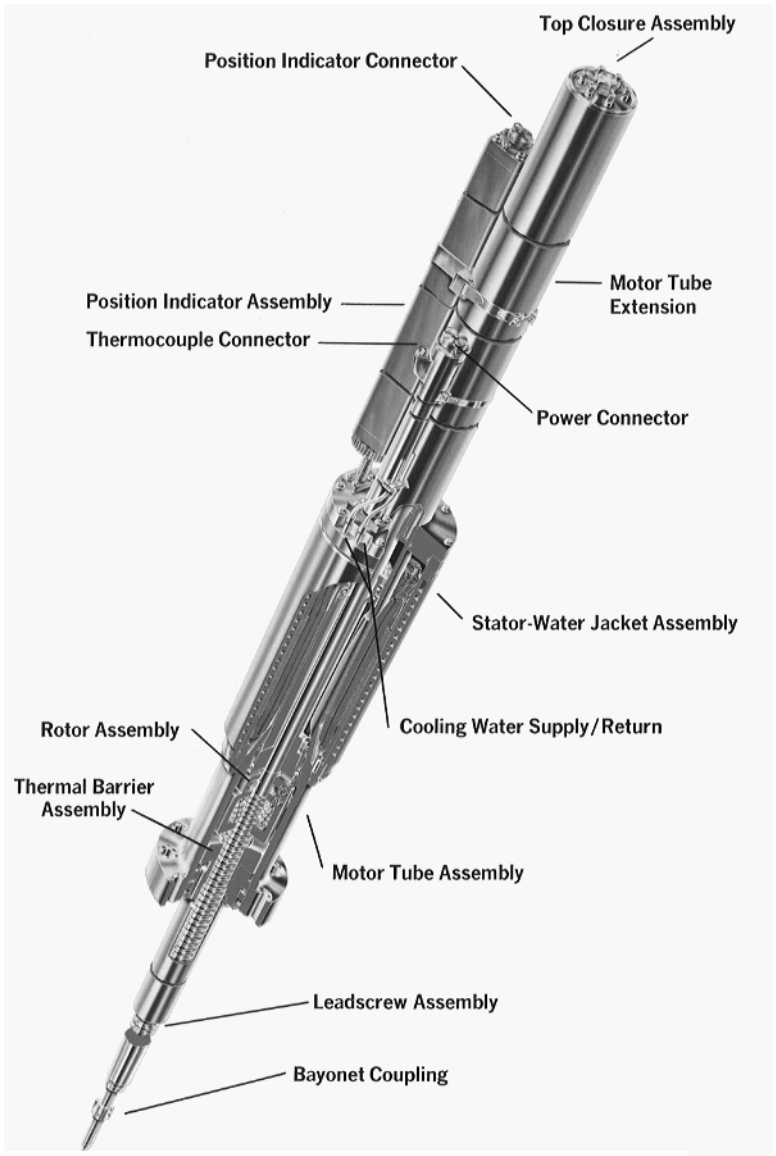


Fig. 11 - A once-through PWR steam generator

Control rods in a PWR enter through the top of the pressure vessel. They are held by a set of electromagnetic latches that enable a rod to be "stepped" up or down. Thus, the rods drop from gravity under a power failure. Normal movement is by way of a control rod drive assembly. Control rods must meet several criteria - have a high neutron capture probability, be machinable, resist radiation damage and corrosion, have low mass (so they can move fast) and be of reasonable cost. Commonly used materials include boron carbide (cheap, high neutron capture), cadmium (high neutron capture, cheap, low melting point), hafnium (corrosion resistant, poor machining characteristics, expensive), gadolinium (high neutron capture, cheap) as well as europium, silver and indium. The rod drive, as shown in Figure 12, uses a magnetic latch and a jack arrangement which allows small position changes to be made.



Courtesy, Babcock & Wilcox

Fig. 12 - A PWR control rod drive mechanism

A second, independent technique is used to adjust reactivity in a PWR. The controlled addition of boric acid (a strong neutron absorber from the boron) to the water in the primary loop is referred to as a chemical shim. A recovery system may be used to remove and store the boric acid for recycled use.

Inconel 600 is a nickel-based corrosion resistant material that was used for PWR steam generator tubes, for reactor head and pressurizer heater nozzles, and as part of many other primary system components. With time and experience, the alloy was found to suffer from primary water stress corrosion cracking (PWSCC). A combination of operating temperature, residual internal stress (caused by heat treatment plus cold

working), and chemical environment lead to PWSCC. Cracking is observed to occur within 1-27 years and is difficult to predict. Testing by EPRI showed that all Inconel 600 used in U.S. plants is susceptible to cracking at some future time. This is a larger issue for PWRs than for BWRs.

Steam generator (SG) replacements have been performed in many PWRs throughout the world. U.S. replacement experience began in about 1990 and continues through the present. Inconel 600 and poor chemistry controls are major reasons for steam generator failures. Steam generator replacement outages have ranged from about 41-125 days and exposure totals are in the 59-670 person-rem range. Palo Verde-2 completed a SG replacement for 41 days and 59 person-rem - the best in the world for a two-generator replacement job. Some plants have used Inconel 600 in their new steam generators and some have used Inconel 690.

CRDM nozzle cracking was first observed in European plants and later in U.S. plants. The first cracked reactor head nozzles (CRDM nozzles) were detected in a French plant in 1991. It was decided in 1992-3 that the entire French fleet of PWRs (54 units) would receive replacement reactor heads and that the new nozzles would be Inconel 690. The project began in 1994 and was completed in 2008. The replacement alloy has a long service history and is free of the PWSCC problem. A few U.S. PWRs have replaced reactor heads and most others plan to within the next few years. The alloy of choice for CRDM nozzles in U.S. plants is also Inconel 690.

Radiation exposure for reactor head replacement in the French plants has ranged from about 19 to 92 person-rem. The average for plants done in 1994 was about 41 person-rem and steadily declined to about 24 person-rem in 2001. Davis-Besse was the first U.S. plant to replace a reactor head and did so in 2002 for about 41 person-rem. North Ana 1 and 2 performed reactor head replacements for about 27 and 31 person-rem, respectively, in early 2003. Surrey-1 expended about 68 person-rem for a head replacement in mid-2003. The last reactor head replacement completed in 2004 was performed at Turkey Point and cost only about 6 person-rem. NATC/ISOE facilitated the exchange of reactor head replacement ALARA information among the French and U.S. plants and is a significant factor in the good U.S. plant performance.

The first couple of Inconel 600 pressurizer nozzle replacement jobs were completed in early 2005. More of these jobs should be anticipated. The radiation exposure cost was about 65 person-rem for replacement of 29 heater nozzles. The industry is on a steep learning curve and can be expected to apply lessons learned to significantly reduce the dose for this job in future replacements.

In terms of general operating characteristics, a PWR is inherently quite stable. It has a high negative temperature coefficient. The thermal neutron flux in the core is typically about 2×10^{13} n/cm^2-sec. The core excess reactivity typically runs 0.200 for a cold, new core to about 0.070 for a hot core with Xe and Sm poison concentrations at equilibrium values. Since U-238 makes up most of the fuel, the reactor will produce Pu-239 via the following nuclear reaction:

$$^{238}\mathrm{U} + \mathrm{n} \rightarrow {}^{239}\mathrm{U} \rightarrow {}^{239}\mathrm{Np} + \beta^- \rightarrow {}^{239}\mathrm{Pu} + \beta^-.$$

The amount of Pu produced depends on the degree of fuel enrichment. For a typical 1,000 MW_e plant, Pu production would be expected to run around 300 kilograms per year.

A major operational problem is maintaining the pressure in the primary loop at around 2230 psi. If the pressure falls, steam bubbles can form in the core which will damage the primary coolant pumps. Pumps are designed to move water, not a water/steam mixture. If the pressure climbs, risk of bursting a weld or seal is increased. Typically, there are three different "set points" for pressure control. At around 30 psi overpressure, the pressurizer water spray activates to cool down the primary coolant. If the pressure rises more than about 150 psi above normal, the reactor scrams. Finally, if the pressure exceeds about 250 psi over the design value, the pressurizer relief valve will automatically vent the excess pressure.

If the pressure falls about 25 psi below optimum, electric heaters in the pressurizer are energized which cause the pressure to rise. If the pressure drops to about 250 psi below the design value, additional heaters kick in to try to raise it to normal levels. A drop of about 400 psi will initiate a reactor scram.

Most PWRs have a pair of safety injection systems; one a low pressure system and the other a high pressure one. The low pressure injector is merely a tank or tanks of borated water (at about 2500 ppm) kept at a pressure below the normal operating point for the reactor. If the pressure in the primary loop drops below the injector tank pressure, water is sucked into the primary system by the pressure difference. The high pressure safety injection system makes use of pumps to refill the primary loop with borated water under accident conditions. Borated water is also collected in the containment building sump and is recirculated to the core and to a spray system.

One of the most characteristic external features of a PWR is the containment building, a reinforced concrete structure frequently topped by a hemispherical dome. Steel cables, held under tension, add strength to the containment. In the USA, the typical design will withstand an internal pressure surge of about 55 psi above ambient air pressure. This is the calculated overpressure within the containment building that would result if a major break in the primary loop released all the water and the water instantly vaporized to steam. The containment structure is isolated from the outside environment during normal plant operations. In the event technicians must enter, a personnel access airlock is used to maintain strict isolation of the containment atmosphere.

High inside the roof of the containment shell (typical floor to ceiling height is 240 feet) is a system of spray nozzles. During an accident they can activate to cool the air, following a steam release, which helps to lower the pressure and condense the steam back to water. The spray system can also discharge a sodium hydroxide solution. The resulting basic liquid keeps iodine in solution and greatly reduces the potential for a release of iodine into the environment. This solution is recirculated from the containment sump through the containment sprays to keep iodine washed out of the air. At some time after the accident, this liquid may be transferred by a sump pump to liquid radioactive waste treatment facilities. In many accident calculations, the iodine isotopes play a key role by accounting for the major fraction of the internal radiation doses received by the population. Thus, if the iodine can, so to speak, be "cut off at

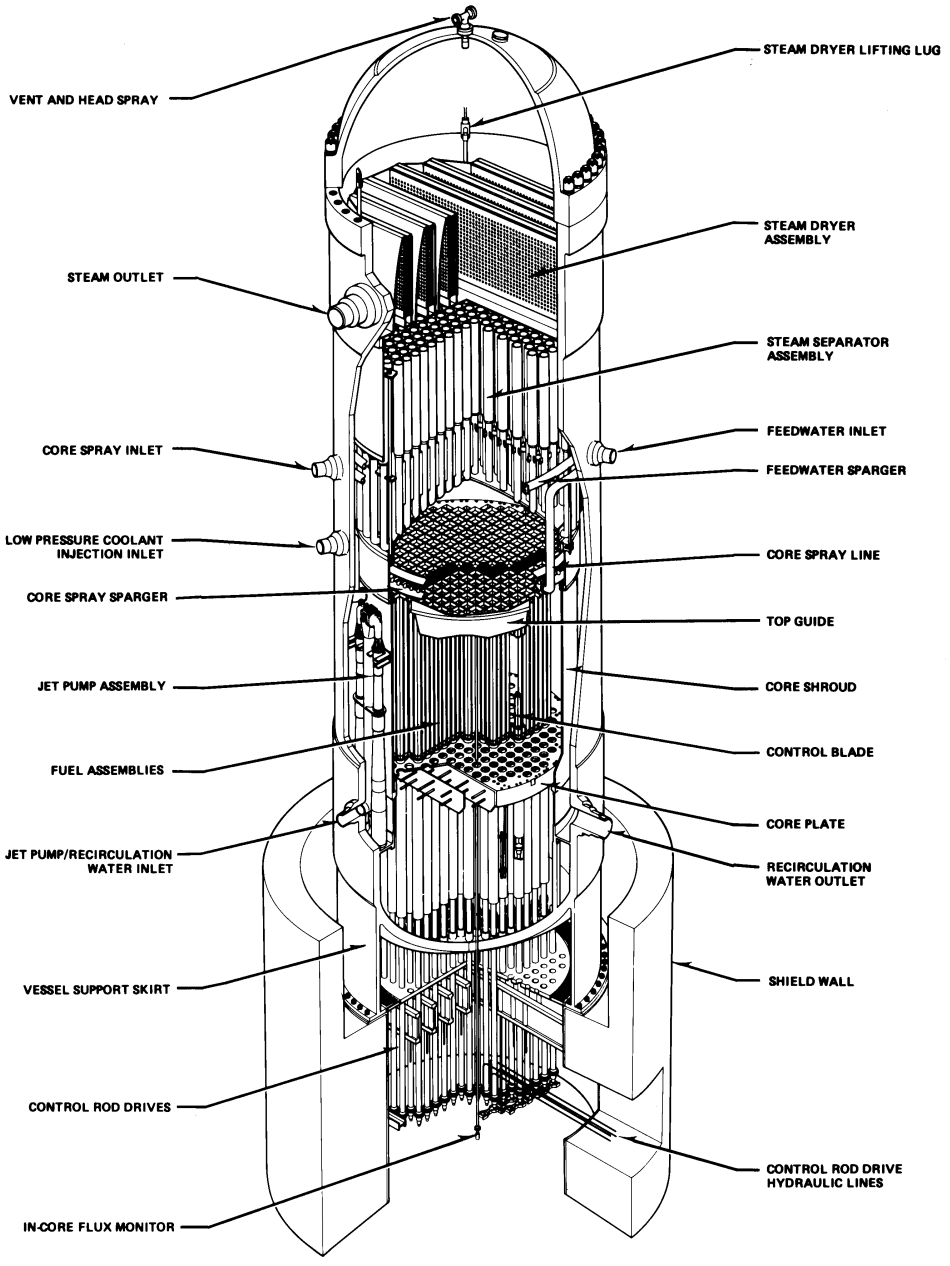
the pass" before it is released into the environment, the accident consequences will be greatly reduced. This was the case in the Three Mile Island accident. As is well known, iodine played a very minor role in terms of the radiological impact on the environment and on population doses.

Continuing on the subject of things that can go wrong, the opening of the circuit breaker on the electrical generator that produces the plant output is a major problem. If the breaker opens, the plant is subjected to a "turbine generator trip," a serious event. Disconnecting the electrical load from the generator means that almost no energy is required any longer to turn it. But prior to the trip, the entire reactor thermal energy output of about 3,000 MW was devoted exclusively to this task by energizing the turbine. The turbine responds to the situation by speeding up precipitously, and if unchecked, will literally self-destruct by throwing off turbine blades at high velocity. Various safety systems are designed to come into play to divert the steam away from the turbine and dump it to the main condenser or vent it into the atmosphere until the plant can be shut down.

BWR Features and Characteristics

The BWR design has some advantages over a PWR system. Of course, the plumbing system is much simpler, due to the absence of a steam generator and the accompanying secondary loop. There are some disadvantages, as well, to the BWR design. By having primary coolant flowing through all of the main pipes, including the turbine generator, any leak of a pipefitting or valve seal is a major contaminating event. Experience has proven that this is the case. As discussed in Chapter 13, a much greater volume of low level solid waste is shipped each year from a BWR compared to a PWR. Much of this volume is due to contaminated protective clothing, absorbent paper and other disposable goods used in routine cleanup.

Another difficulty with the BWR design relates to the fact that water is allowed to boil to steam in the core region. As mentioned above, steam has much lower neutron absorbing ability than water. The presence of steam bubbles in the core means that the reactivity is reduced in direct proportion to the void volume, i.e., the fraction of the water that has vaporized to steam. The moment-by-moment changes in void volume, caused by minor fluctuations in core pressure, make the reactor more difficult to operate at a steady power level compared to a PWR. The steam bubbles provide a "negative feedback" in terms of reactor control. If the electrical demands on the plant increase, the turbine draws more steam, thus, reducing the steam pressure in the pressure vessel. In turn, this allows more water to vaporize to steam, reducing the overall core reactivity, and leading to lower reactor power. Reactor control is also complicated by the presence of activation products and fission products circulating in the coolant. In order for the BWR to operate smoothly, the water must be much cleaner than in a PWR. Extensive water treatment is performed continuously on the water that is removed from the pressure vessel, cleaned by ion exchange resin beds and then recirculated back as feedwater.



Courtesy, GE Nuclear Energy, San Jose, CA

Fig. 13 - A BWR reactor assembly

The pressure vessel in a BWR is often constructed out of carbon steel with an internal liner of stainless steel (see Figure 13). The design core pressure is typically around 1,000 psi - only half that of a comparable PWR. This translates into a thinner vessel wall than in the PWR. The top of the pressure vessel, the vessel "head," is removable, but in contrast to the PWR, there are fewer penetrations in the head since

the control rods in a BWR enter the vessel from beneath. In order to refuel a BWR, it is merely necessary to remove the head and proceed. This is a big advantage over the PWR, which requires the disassembly of multiple control rod drives and instrument penetrations before access is gained to the fuel elements.

As was the case in a PWR, the steam driving the turbine must be relatively dry, i.e., free of water which would damage the delicate blades turning at high speed. The top of the BWR pressure vessel contains moisture separators and dryers to accomplish this task. Below these components is the fuel itself. A typical BWR core has about twice the mass of fuel as a comparable PWR, perhaps 200 tons versus 90 tons.

Fig. 14 - A BWR fuel assembly

Courtesy, GE Nuclear Energy, San Jose, CA

The typical BWR has in the neighborhood of 750 individual fuel element assemblies. An example is shown in Figure 14.

The coolant system has a number of unique features. At low power level, the natural circulation of the water due to convection heating is adequate to perform the required cooling of the core. When the reactor operates at high power level, above 40% of capacity, it is necessary to supplement the cooling. This is done by injecting recirculated water through a "jet pump" located physically inside the pressure vessel.

A 1200 MW_e BWR reactor has close to 200 control rods, the large number being due to the control difficulties related to steam bubbles in the core. They usually contain boron carbide and are constructed, in a cross-sectional view, with the shape of a + sign. The rods are inserted using a hydraulic principle.

The basic containment structure for a BWR is smaller than for the comparable PWR (there are no steam generators to enclose). Below the pressure vessel is a suppression pool filled with water that helps to condense steam and trap fission products in the event of a rupture of some vital component (see Figure 15). The overall smaller volume of coolant (single loop vs. two-loop design feature) and lower pressure are additional reasons for the simpler containment.

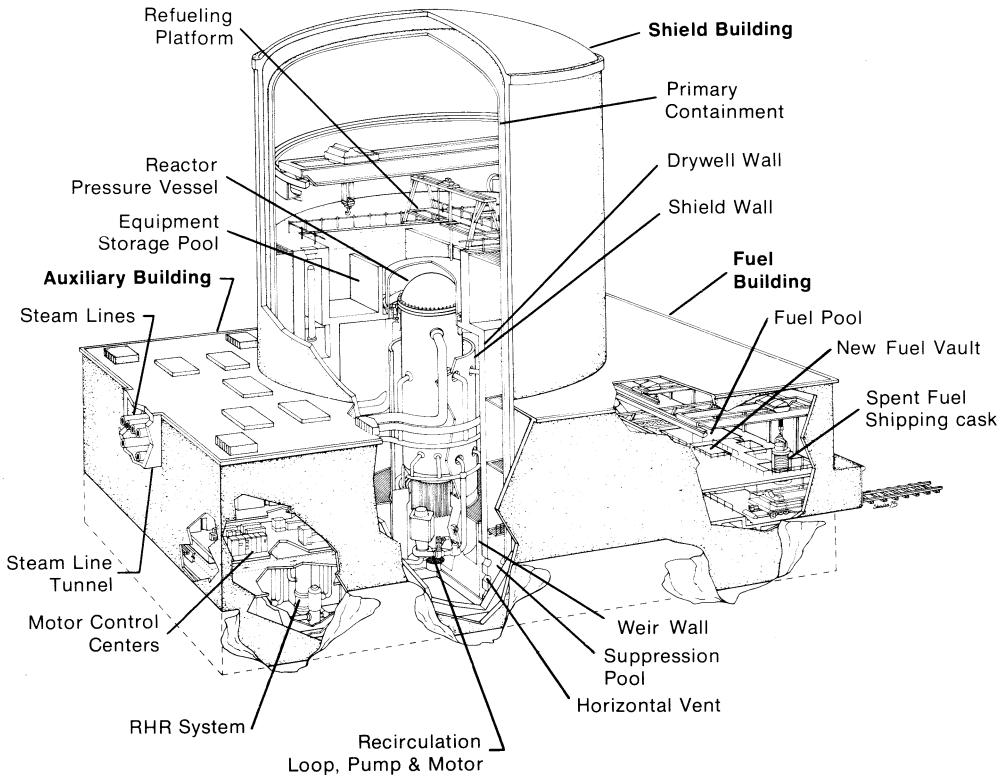


Fig. 15 - A Mark III BWR containment

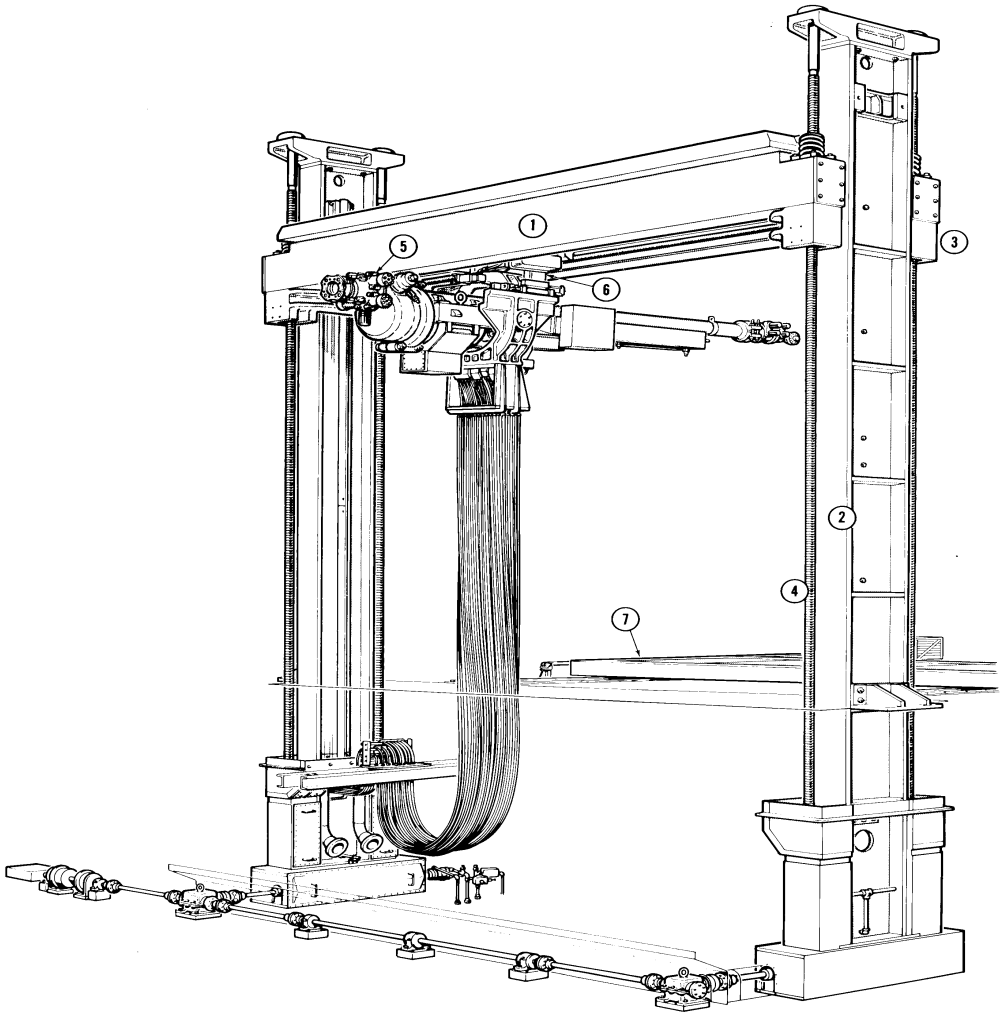
Courtesy, GE Nuclear Energy, San Jose, CA

CANDU Features and Characteristics

There are three generic advantages to a heavy water reactor:

- Natural uranium is used for the fuel.
- Refueling occurs during full power operation.
- The design is inherently safer to transients.

By choosing heavy water as the moderator, the thermal neutron absorption is reduced by 630 times compared to ordinary water. This means that enough neutrons will survive in the core to make $k_{eff} > 1$ with natural, non-enriched uranium as a fuel. Very few countries have the capability to enrich uranium in the U-235 isotope, but enriched uranium is needed for light water reactors. Thus, the heavy water design is favored in countries that have no enrichment facilities. The design of the heavy water reactor provides for shuffling (moving fuel elements from one location to another within the core) or for exchanging fuel bundles remotely while the reactor



Courtesy, Ontario Hydro

Fig. 16 - A CANDU refueling machine

operates at full power. Figure 16 shows the refueling equipment and Figure 17 provides details on the reactor assembly. Notice that, in contrast to U.S. light water reactors, the fuel channels are positioned horizontally in the core. This means that the periodic shutdown for refueling, characteristic of U.S. reactors, is not necessary. The reactor can operate continuously for very long periods. For example, the Bruce-1 unit in Canada was once connected to the power grid 96.6% of the time during a four-year period. Online refueling is possible due to large spacing (about 30 cm) between fuel bundles and by the lower reactivity per bundle due to use of non-enriched uranium. Removing a bundle (Figure 18) does not cause a large disruption of neutron flux in the core as would be the case for a light water reactor.

Finally, the heavy water design is safer under conditions of power transients - rapid changes in power level. This is again due to the use of natural uranium fuel with inherently lower reactivity.

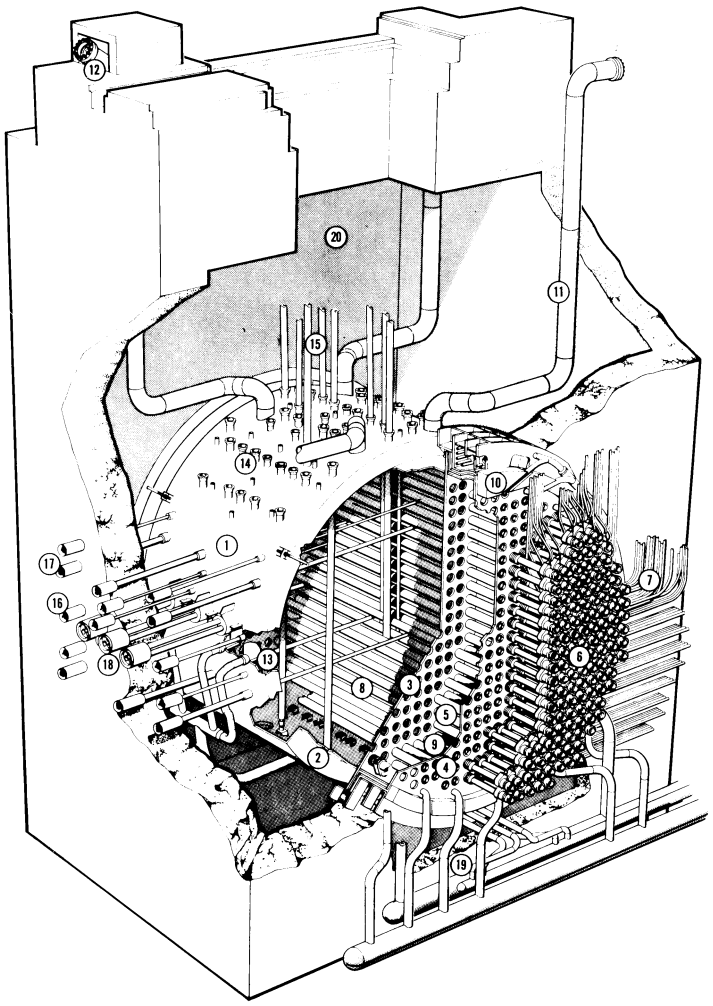
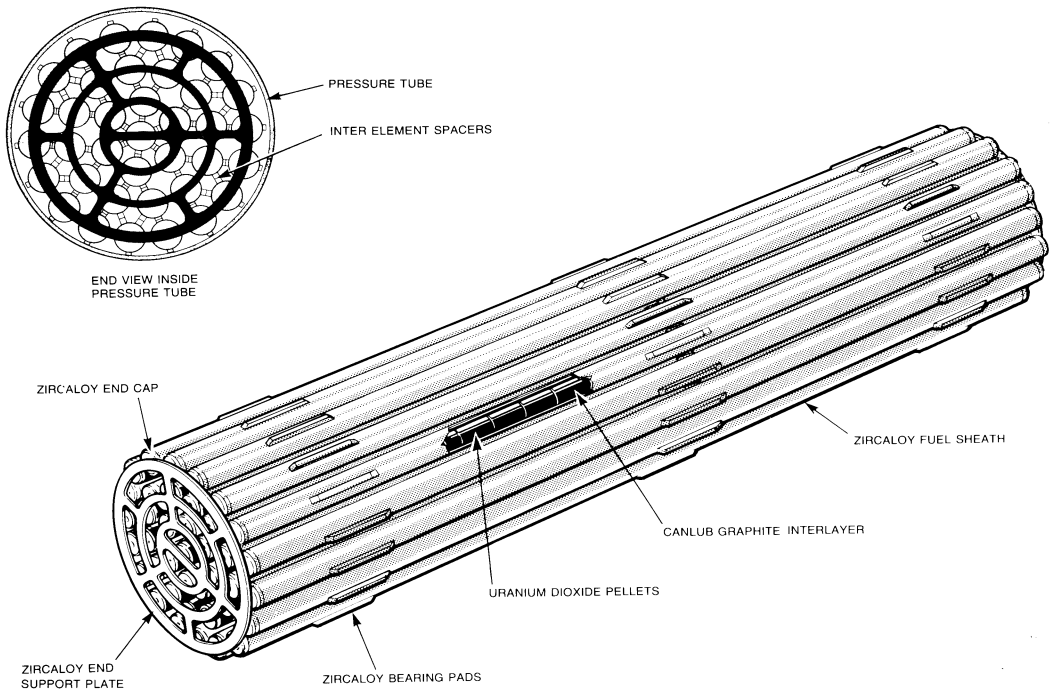


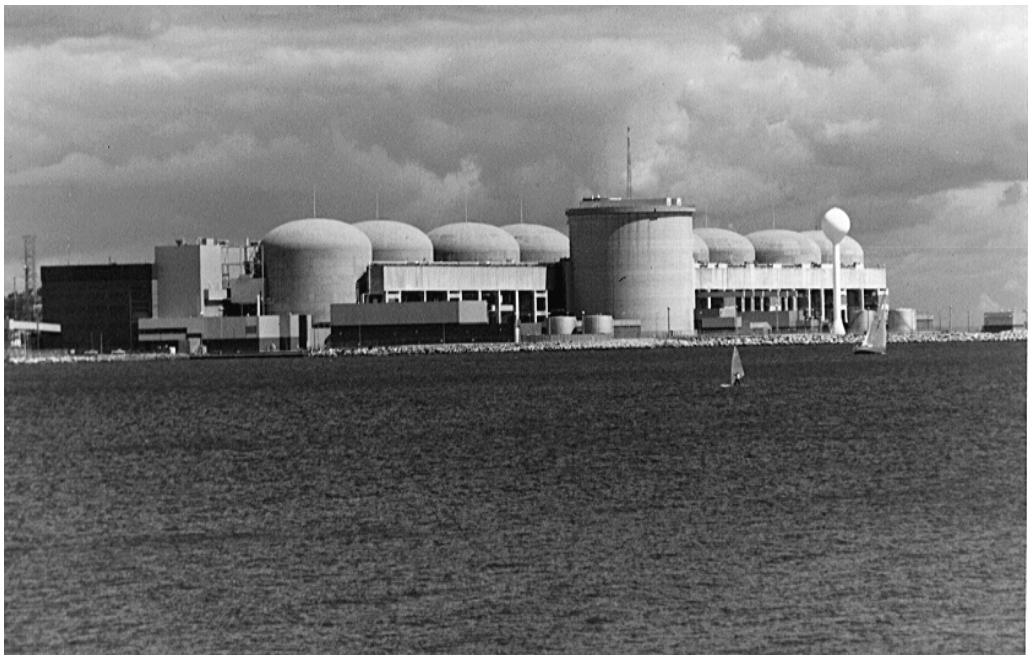
Fig. 17 - The CANDU reactor assembly

Courtesy, Ontario Hydro



Courtesy, Ontario Hydro

Fig. 18 - A fuel bundle for the CANDU reactor

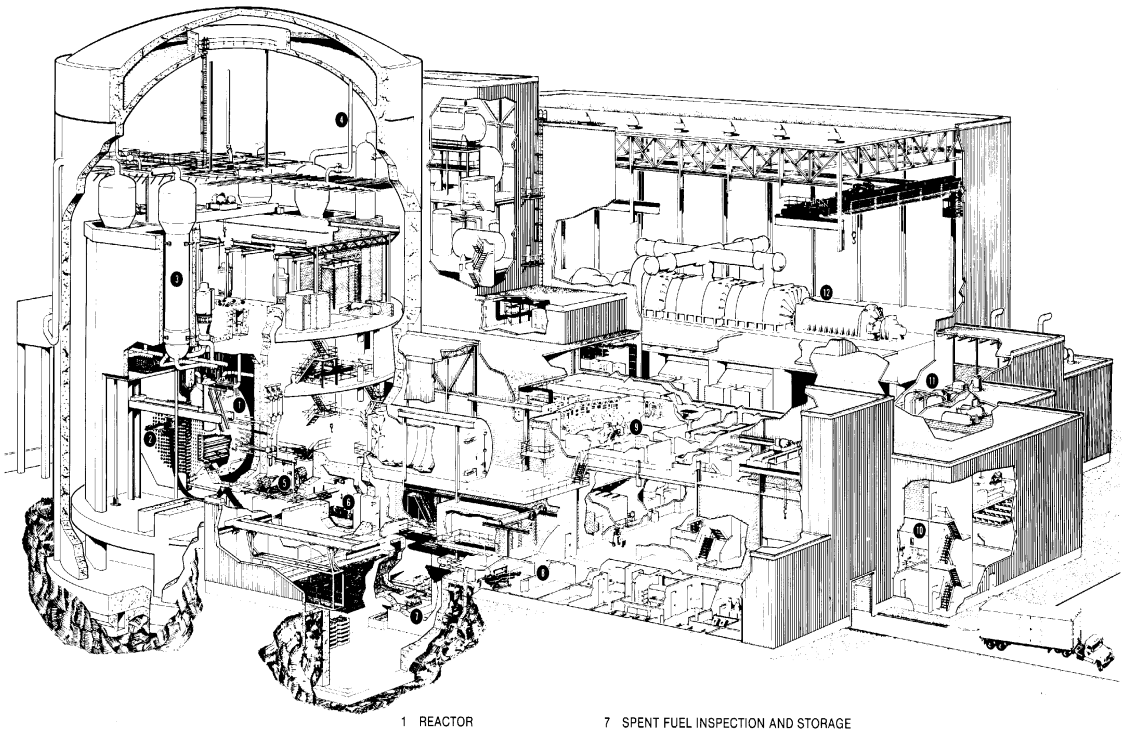


Gollnick

Fig. 19 - The eight reactor units at the Pickering site, Toronto, Ontario

Another difference between the Canadian and U.S. programs is evident in reactor siting. Although there were 103 U.S. nuclear generating stations operating in 2005, there were only six sites in the U.S. which had three reactor units on-site. The remaining sites had only one or two reactors per location. The Canadian approach is to place many units on a given site. For example, the Pickering Generating Station near Toronto, Ontario, has eight reactors, each of 540 MW_e (Figure 19).

Although each reactor is housed within a concrete reactor building, a single "Vacuum Building" (the cylindrical building closest to the lake in Figure 19) provides containment protection for all eight reactors in the event of a catastrophic accident. In the event of excessive pressure in any one of the reactor buildings, a pressure seal is ruptured which conducts the overpressure surge into the evacuated vacuum building to prevent release into the atmosphere. Figure 20 shows a cutaway drawing of a 600 MWe CANDU plant.



Courtesy, Ontario Hydro

Fig. 20 - A 600 MW_e CANDU plant

RBMK-1000 Features and Characteristics

The RBMK-1000 is a Soviet boiling water reactor design that was first built in 1973 at Leningrad. In 1986, there were 14 units of this type operating at 1000 MW_e in the Soviet Union, plus one unit scaled up to 1500 MW_e. The world's most serious nuclear power plant accident occurred on April 26, 1986 at the Chernobyl Unit 4 NPP. This event was precipitated by performance of an unanalyzed experiment that permitted the reactor to go prompt critical. See Chapter 14 for details. Today (2005) there are eleven units in operation at the Kursk (4), Leningrad (4), and Smolensk (3) sites. All of these units have upgraded to approach Western safety standards.

The original design was characterized by several unique features, which include the ability to refuel at power (five channels can be refueled per day), division of the graphite moderated core into two separate halves, a slow scram system and a positive void coefficient (as a result of excess graphite moderator). The fuel is of relatively low enrichment, about 2%. Figure 21 shows a cross-sectional view of the major

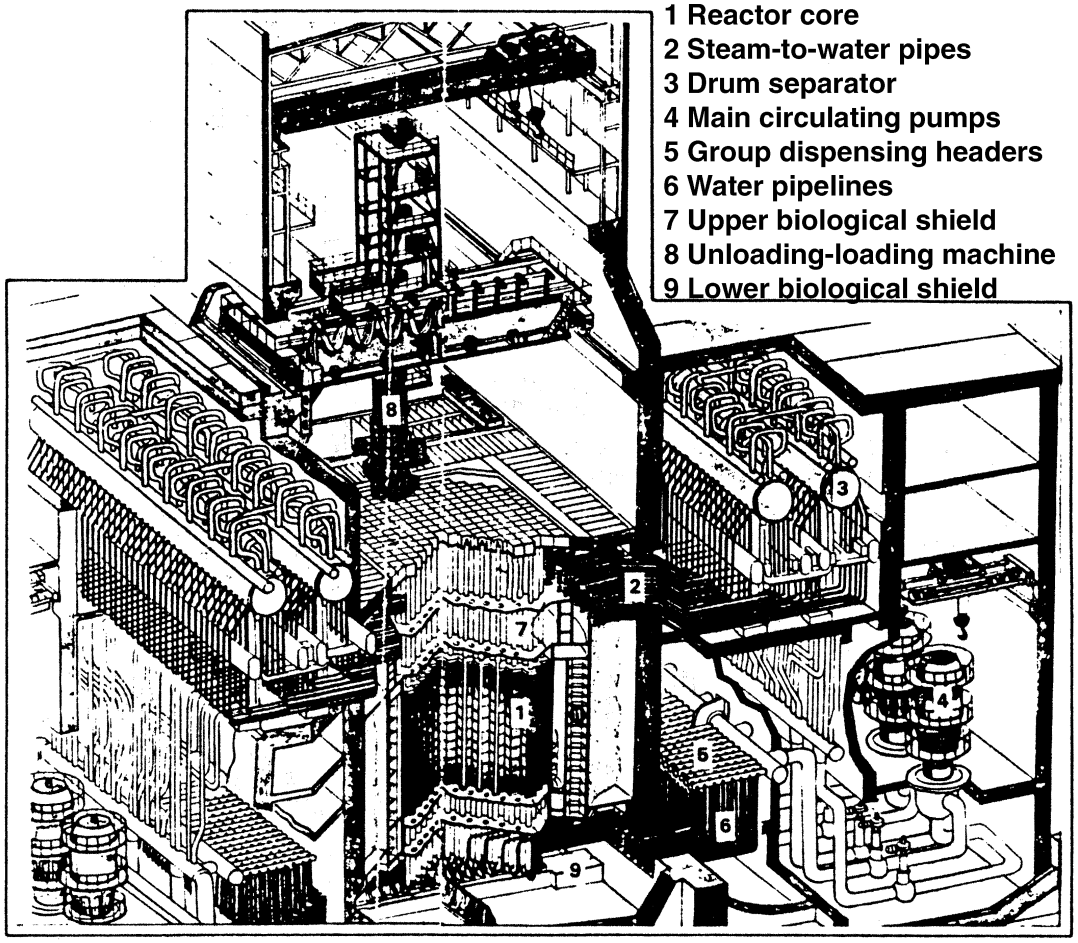


Fig. 21 - The design features of the Soviet RBMK-1000 reactor

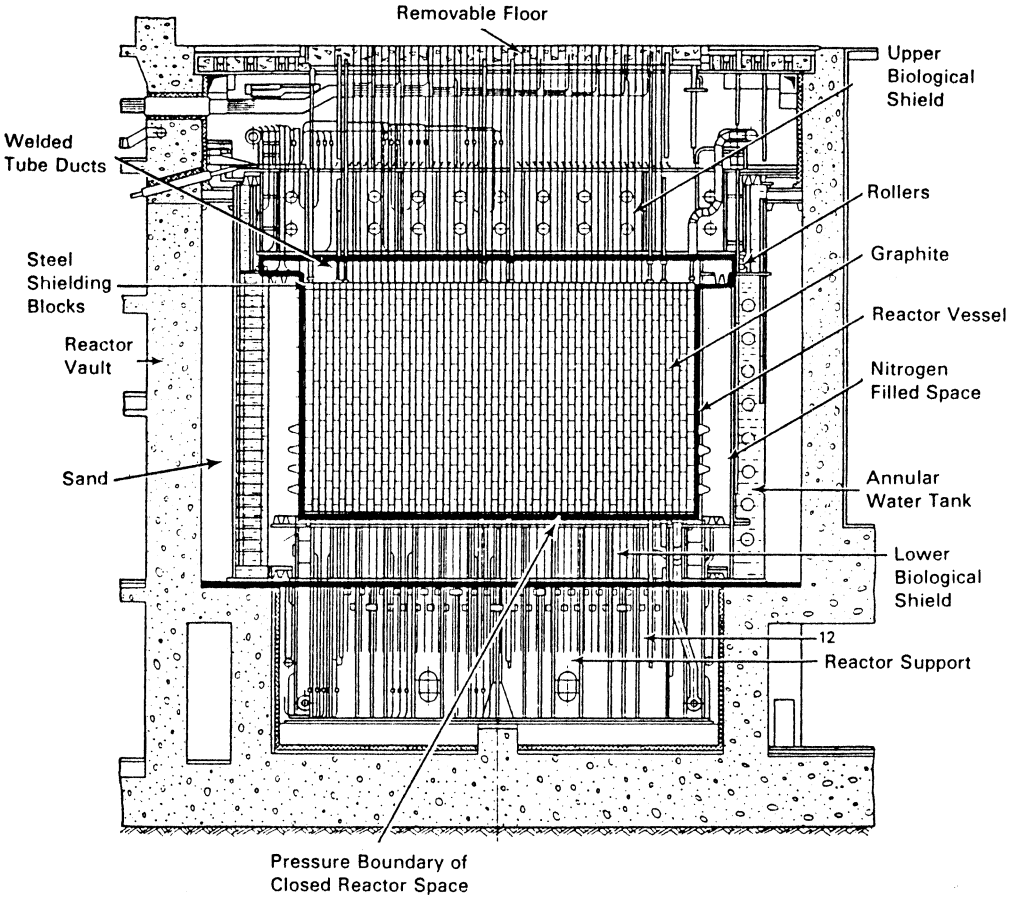


Fig. 22 - The RBMK-1000 reactor cavity and internals

components of the design, while Figure 22 gives details of the reactor cavity itself. Figure 23 is a side view of the Chernobyl plant.

Each unit at the Chernobyl site had two separate primary cooling loops, one for each half of the core. There were four primary circulating pumps in each loop. Unit 4 was equipped with 211 control rods. The graphite moderator weighed about 1800 tons and was built in the shape of a vertical cylinder. The reactor core was surrounded by a 5/8 inch thick vessel, about 50 feet in diameter by 32 feet high. This structure only has to contain the graphite cooling gas as steam pressure is confined within pressure tubes fitted into 1661 fuel channels running vertically through the graphite. The top of the core was dominated by an 1100 ton circular concrete upper biological shield 10 feet thick. A series of 4 foot thick water tanks plus a layer of sand provided radiation shielding to the sides of the core. Above the upper bioshield, a heavy concrete floor slab was poured in removable sections to enable refueling. The region above this floor slab held the refueling machine and an overhead traveling crane. Each pair of the 4 units at Chernobyl shared a common ventilation stack and a single room for the control boards.

In addition to the primary cooling system, the design includes a separate control rod cooling system, a gas cooling system for the 80% helium-20% nitrogen mix-

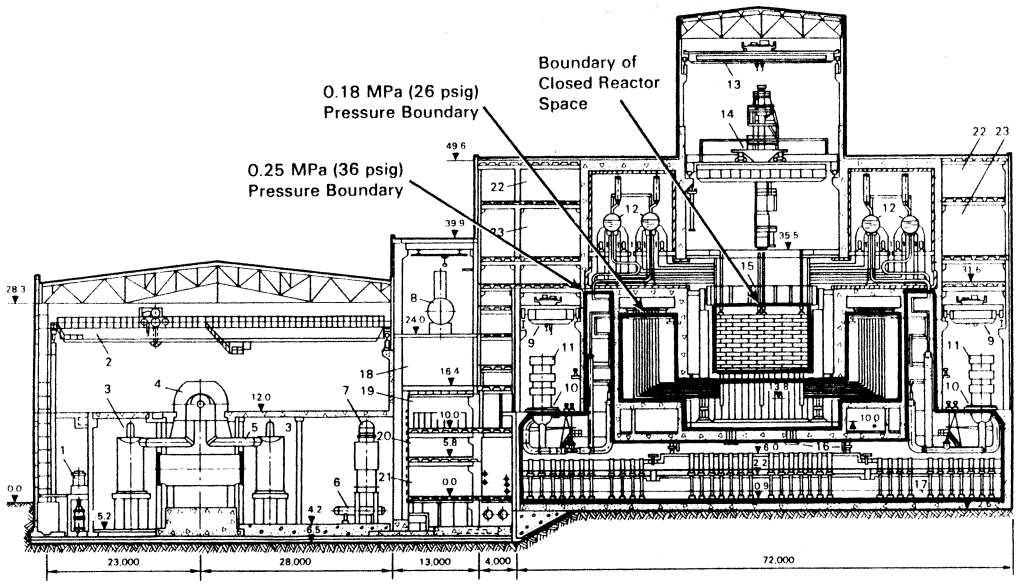


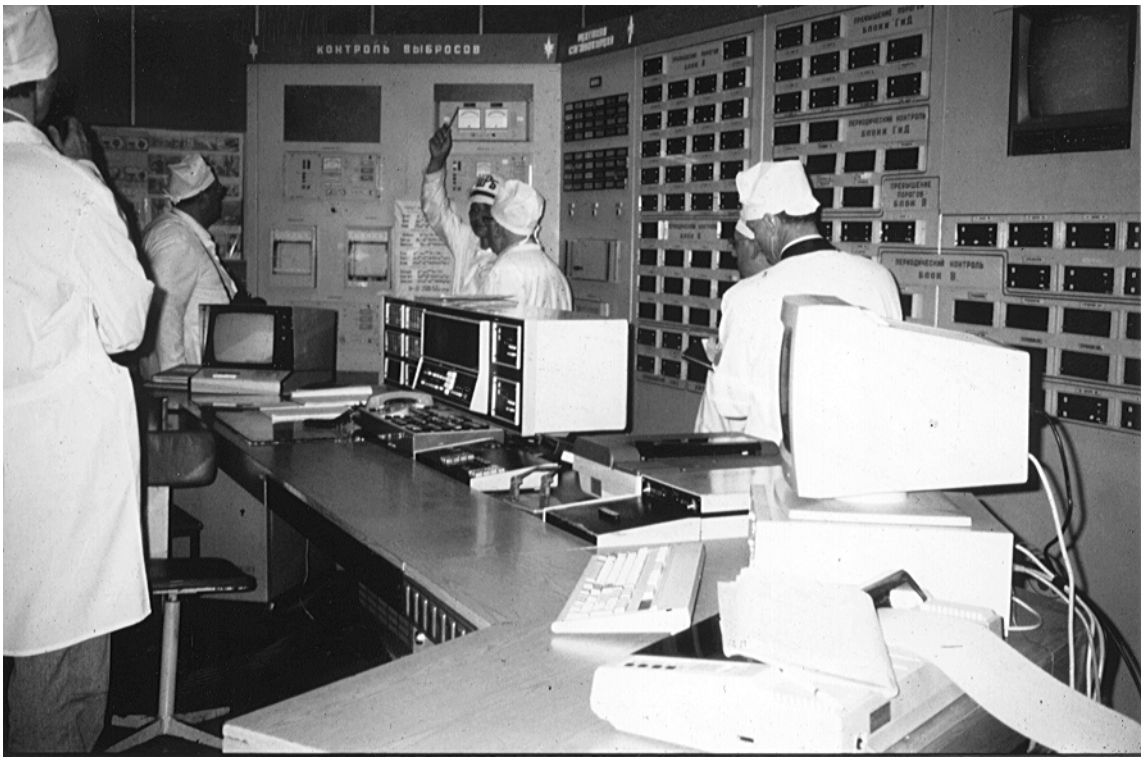
Fig. 23 - Cross sectional view of Chernobyl reactors

ture that circulates through the graphite and three independent emergency core cooling loops, any one of which can provide cooling for one of the halves of the reactor core.

Control rods are divided into groups, some of which are automatically controlled by computer and others which are manual. The automatic rods are linked to ion chambers which can insert these rods if excess neutron flux is detected by at least two chambers. Rods are raised or lowered by a motor driven drum connected via a cable. In the original design, it took 20 seconds to fully insert the rods, even in an emergency scram!

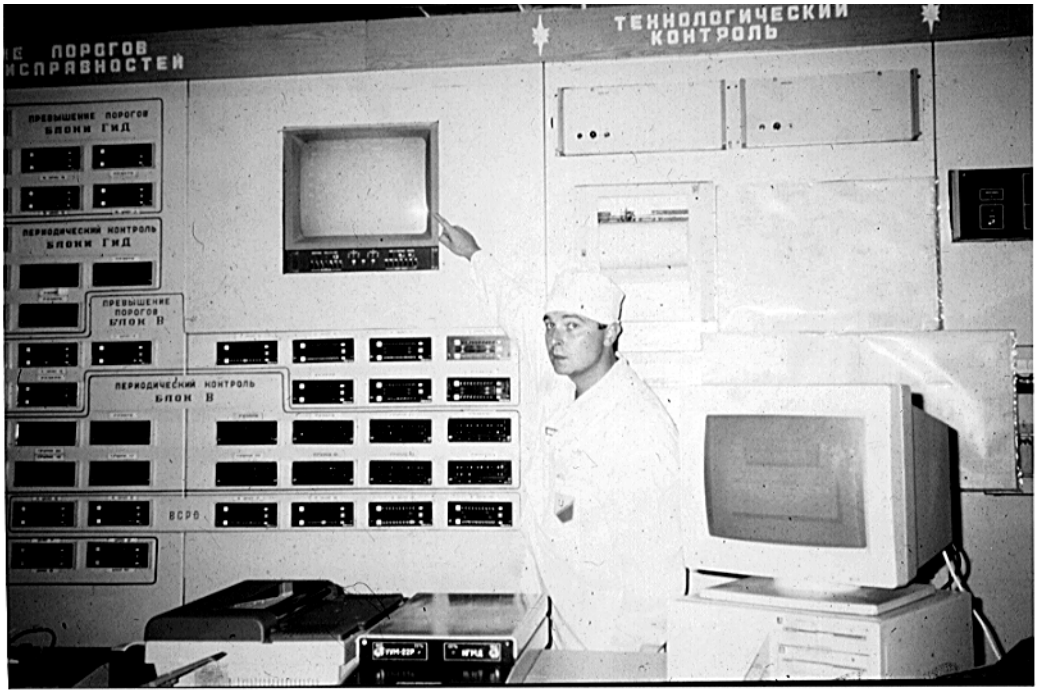
Although not employing a containment building, the RBMK-1000 design does provide for isolation of the reactor atmosphere in the event of accidents of various types and severities. If a single pressure tube ruptures during full power operation, the pressure burst is directed to a pressure suppression pool by escape valves. This emergency system can stand an overpressure of about 10 psi. Under conditions of overpressure above 30 psi, which might be caused by simultaneous breaks of more than one pressure tube, the upper bioshield will lift up. The fuel channels and control rod channels are welded to this structure, so the lifting action will rupture vast numbers of pressure tubes and at the same time, hoist the control rods out of the core!

Figure 24 shows the control room for Chernobyl's Unit 3. The control boards for the other three units at the site are identical. Figure 25 is the readout panel for the Unit 3 radiation protection area monitors. Finally, Figure 26 shows some of the portable instrumentation available at the site. According to the plant technicians, only a very limited number of portable meters were available during the 1986 accident. Due to its notoriety, at the time of closure, the Chernobyl plant had more instrumentation than all the other RBMK-1000 sites. Foreign donations helped considerably!



Courtesy, Michael Davidson

Fig. 24 - Chernobyl Unit 3 control room



Courtesy, Michael Davidson

Fig. 25 - Chernobyl Unit 3 radiation area monitor display panel

Courtesy, Michael Davidson

Fig. 26 - Portable radiation survey meters available at Chernobyl

Power Reactor Start-up

Before a nuclear power reactor can generate electricity, a series of steps must be completed. The initial start-up, prior to the first commercial operation of a plant, is the most complicated, but normal restarts after shutdown follow many of the same procedures.

For an initial start-up of a plant, a series of Preoperational Tests are conducted, usually over the course of a year or more before the actual uranium fuel is first loaded. These tests assure that the primary coolant systems and pressure vessel meet design specifications. The various high and low set points for temperature and pressure sensors are adjusted, and the vessel is filled with water. The water treatment systems can then be activated and subjected to testing. Remote sensors that can lead to a reactor scram are individually checked to see that they perform their designated duties. Hot, functional testing is demonstrated by raising the primary system to full temperature and pressure ratings and holding them there for 100 hours. The ability of the containment structure to retain its atmosphere is verified by pumping the structure to design pressure, typically about 60 psig, and measuring the leak rate. Finally, the various mechanical components that don't interfere with fuel loading are connected and run through qualification testing.

Fuel Loading is the next hurdle on the road to power operation. Isotopic neutron sources are first installed near the core. These are used to supply the initial neutron flux needed to bring the reactor to criticality. They also allow testing of the low range ("source range") neutron monitoring systems in the core. Next, the fuel elements are placed in the vessel, in a previously calculated arrangement. If the reactor has operated previously, then the presence of irradiated fuel elements in the vessel necessitates the use of additional radiation shielding, often in the form of waterpools near the vessel, during this refueling stage. Additional tests can then be conducted of the neutron detectors. Continual checks of k_{eff} are performed to verify that the core is subcritical.

At this point, Low Power Physics Testing commences. The reactor is brought critical with the power level not exceeding about 1% of the full power design limit. Criticality calculations and control rod positions are verified.

Full Power Physics Testing is usually done by raising the power level through a series of incremental increases, typically 25% of full power, then 50%, 75% and finally, 100% of rated power. The neutron detector systems that operate in this "power range" can be calibrated and tested. Before moving from one power step to the next, each of the various safety scram systems must be tested individually to assure their ability to shut down the reactor. Initial performance data can be obtained to see the effect of using different combinations of control rods and coolant flow rates, etc. Every reactor has its own "personality" and the full power testing gives the operators their first opportunity to get acquainted.

Very extensive radiation surveys are performed inside containment and in all other parts of the plant at each power level. This data is used to verify the shielding design.

The last stage of full power testing involves purposely causing a large disruption in normal operating parameters to make sure that the reactor can shut itself down safely. These disruptions might include reactor trips caused by the opening of the generator breaker, failure of a coolant pump or a turbine problem. Final acceptance of a new plant requires that the reactor operate continuously for a 100 hour period at 100% of its rated power.

Rad Waste Handling Systems

Sources: The largest routine volumes of liquid wastes come from primary system letdown and equipment drains. The on-site laundry for protective clothing only contributes a small volume. Nonroutine operations can, of course, produce large volumes of contaminated water on occasion, e.g., refueling operations including reactor cavity decontamination.

Radioactive gases are formed as fission products, chiefly xenon and krypton isotopes. The levels of these fission product gases is strongly dependent on defects in the cladding of the individual fuel pins, or the presence of "tramp uranium" (exterior surface contamination of fuel pins with uranium). In addition, oxygen dissolved in

coolant water can become neutron activated to produce nitrogen isotopes. ^{13}N is the most troublesome due to its 10 minute half-life. It is a beta emitter. ^{16}N only has a 7 second half-life, but this isotope has the distinction of being the highest energy gamma emitter known. Tritium, in the form of tritiated water vapor (T_2O or HTO) is a problem in PWRs because of the boric acid chemical shim mentioned earlier. The ^{10}B isotope in the boric acid captures a neutron. The resulting nucleus disintegrates into two alpha particles and a tritium nucleus. In heavy water moderators, such as the CANDU design, lots of ^{3}H is released as a result of neutron capture by deuterium nuclei in the heavy water. In a BWR, tritium is only produced by ternary fission. The tritium atoms become adsorbed on the zirconium in Zircaloy cladding, so these fuel pins release only about 1% of the tritium produced. If the fuel uses stainless steel cladding, instead, then about 32% of the produced tritium diffuses out into the coolant.

Normal operations, such as processing of liquid or gaseous wastes, and maintenance operations produce solid rad wastes. Typical low level solid wastes include absorbent paper and plastic goods. High level solids consist of such items as spent ion exchange resins, filters, evaporator bottoms and concreted liquid wastes.

Processing: In general, several procedures are used to process liquid rad waste. Suspended solids can be removed by filtration. Corrosion products can be removed with high efficiency by this method. Ion exchange works best on low concentrations of ions. It is a cheaper technique than distillation. Resin beds have a long life if properly cared for, but ultimately must be disposed of as solid waste. Evaporation is often used to reduce the volume of liquid waste.

Radioactive waste gases in a PWR are collected from vents on the surge tank, vents on the boric acid chemical shim system and gas scrubbers on the primary coolant loop. To prevent hydrogen explosions, the gases are diluted with nitrogen and then run through a catalytic recombiner which chemically converts the hydrogen to water (in the presence of atmospheric oxygen). The remaining gases are compressed into tanks and stored for decay. At a later time, they are discharged into the atmosphere.

In a BWR, most of the radioactive waste gases (xenon and krypton) are collected from the main condenser. They are diluted with steam, again to reduce the risk of a hydrogen explosion. Hydrogen is catalytically recombined to form water. The remaining gases are subjected to one of four different processes. Xenon and krypton are injected into a charcoal bed where these noble gases will adsorb onto the charcoal. As they slowly diffuse through the long bed, they have opportunity to undergo decay. The krypton isotopes take about 20 hours to travel the length of the bed, while xenon takes about 2 weeks. Kr and Xe can also be removed by cryogenic distillation (gases liquefied and then carefully heated to drive off the components one at a time) or by freon absorption in which the gas stream bubbles up against a freon liquid downflow that absorbs the noble gases but allows the other gases through. The remaining gases can be compressed for decay in gas cylinders.

Solid low level radioactive waste may be compacted into drums and stored until a full truck shipment is possible. Most U.S. plants now collect waste and ship it

to facilities in Tennessee for incineration. The ash is then shipped for burial. This greatly reduces the volume for final disposal. High level ion exchange resins and filters are sluiced into large shielded shipping casks and dried or they too can be incinerated.

Health Physics At Power Reactors

Introduction

The core functions of a radiation protection program at a power reactor are (1) to protect radiological workers by controlling their radiation exposure and (2) to control radioactive material such that it does not unintentionally leave the restricted area and cause exposure to the public. All segments of the radiation protection program are focused on assuring one or both of these core functions.

Two external organizations, the Nuclear Regulatory Commission (NRC) and the Institute for Nuclear Power Operation (INPO), provide the controlling regulation and guidance for all power reactor radiation protection programs. The NRC licenses power plants to operate, regulates personnel radiation exposure through 10 CFR 20 and further regulates power plant radiation protection programs via the plant Technical Specifications. NRC performs regular inspections and has the power to issue citations, monetary fines, press criminal actions and require a plant to cease operation. INPO is a post TMI (Three Mile Island) industry organization formed to preclude another such accident by promulgating voluntary consensus standards and by inspecting against those standards. While INPO does not have the same federal regulatory authority as NRC to issue an order to shut down, INPO deals directly with each utility Chief Executive Officer and has the power of the entire industry behind it. INPO has sufficient “peer power” to cause a power reactor to decide to shut down.

American Nuclear Insurers (ANI) is a third organization which has some radiation protection program guidelines and which influences each power plant via the insurance rates it charges. ANI performs periodic inspections. Insurance rates are proportional to the risk assessment which flows from these inspections. The major focus is nuclear plant safety and the utility’s capability to support a litigation with necessary documents.

The Nuclear Energy Institute (NEI) is the industry organization that represents the nuclear energy industry policy to the Administration, Congress, regulatory agencies, the media and others. NEI is the lead organization for radiation protection interaction with the regulatory agencies, legislators, public affairs and for workplace issues. NEI members number about 300 and include all nuclear power utilities, nuclear steam system suppliers (NSSS), architect engineers (A&E), fuel suppliers and universities. NEI is the lead organization for RP 2020, the effort to plan how radiation protection will function in the year 2020 when the industry is expected to be in a strong growth and renewal period. See www.nei.org for more information about NEI.

The Electric Power Research Institute (EPRI) includes all U.S. and some foreign utilities as members. EPRI performs research and development for the industry in all areas of power generation and electrical transmission. EPRI performs substantial work in radiation protection, ALARA, radwaste and Chemistry for nuclear power plants. EPRI's Chemistry guidelines are used by nearly all U.S. nuclear power plants. EPRI has a major role in RP 2020. See www.epri.com for more information.

The Information System On Exposure (ISOE) is a Paris based world-wide organization that collects and analyses radiation exposure information. ISOE has four technical centers; North American, European, Asian, and IAEA. The North American Technical Center, NATC, is based at the University of Illinois and provides radiation protection information and data to and about all U.S. nuclear power plants. NATC promotes world-wide experience sharing by hosting international technical meetings and by conducting benchmarking visits between U.S. and foreign plants. See www.natcisoe.org.

Leadership and Management

Administrative

Administration of the radiation protection program at a nuclear power plant is typically by the Radiation Protection Manager (RPM). This person must meet the educational and experience requirements of NRC Regulatory Guide 1.8 and should also meet INPO guides for the position. Typically, the RPM will have a Bachelors degree in radiation protection or a closely related technical field and over five years of responsible power plant experience. Some RPMs are Board Certified Health Physicists and some have advanced degrees in business because power plants are now being operated as competitive businesses rather than traditional regulated monopolies. The major administrative functions of the RPM and his or her staff are listed in Figure 27.

Staffing

Staffing is driven by a balance between plant operation requirements, the need to operate at a profit in the deregulated and competitive business environment, and regulatory requirements and commitments. As deregulation of the electrical utility industry progresses, and consolidation of nuclear power plant operators continues, there is increasing downward pressure on the size and diversity of staff. Utilities are focusing on core requirements to operate a plant which is in good operating condition. Staff needs for outage support and specialized technical areas (e.g. turbine engineers, TLD lab operation, radwaste disposal) are being outsourced to reduce operating costs. Fuel cycles have been increased and outage durations have shrunk dramatically. Some plants now have fuel cycles near 600 days. The median refueling outage duration in 2001 was about 33 days and decreasing. The shortest refueling outages in the U.S. are under 20 days.

Average staff size for a large two unit PWR site in 2002 was about 1150 people.

Establishing high standards for radiation protection and assuring that the standards are communicated to all site personnel

Developing and promulgating radiation protection standards, policies and practices via written procedures (required by Technical Specifications and by INPO)

Establishing radiation protection program goals (e.g. individual and collective exposure, radwaste volume, personnel contaminations, contaminated area accessible to personnel)

Monitoring and reporting radiation protection performance to station management and site workers

Holding workers accountable for their radiological performance

Assuring that radiation protection activities and occurrences are documented and the documents are preserved

Assuring appropriate radiation protection equipment and facilities are available

Demonstrating leadership

Striving to continuously improve the radiation protection program through observations, critical self-assessment, and investigation into and correction of off-normal occurrences

Fig. 27 - Major administrative functions of the Radiation Protection Manager and staff

A single unit site had about 850 persons. These numbers continue to decline in all areas except security. Radiation protection functions included in these numbers are ALARA, operational HP, technical support, decontamination, and radwaste. Total radiation protection staff ranged from about 50 at the single unit plant to about 85 at the large two unit plants. Technical Specifications influence these staffing levels by the commitments a utility may make to such areas as back-shift and weekend HP coverage and on-shift emergency response personnel.

Training

Instruction of radiation workers is required by 10 CFR 19.12 to be commensurate with the potential radiation hazards a worker may face while working in a restricted area. Topics for instruction include health problems associated with radiation exposure, the use and purpose of protective equipment, methods to minimize radiation exposure, and response to hazard warnings. The depth of information provided varies with the potential hazards and the duties of the worker.

Training programs at nuclear power stations are accredited by INPO through the National Academy of Nuclear Training. INPO provides extensive guidance on the

content of General Employee (GET) and on Radiation Protection Personnel training. Workers at a nuclear power plant receive initial and continuing GET.

Initial GET in radiation protection includes information on all of the topics listed in Figure 28. GET also requires demonstration by the student of practical tasks such as how to use an RWP, how to use the plant RCA entry/exit process, where to wear dosimetry, how to don and doff protective clothing, how to use whole body frisking equipment and how to use a manual frisker to check hands and feet. Some plants integrate all these practical aspects of training by setting up a simulated RCA where students enter and perform work under simulated radiological conditions. An instructor observes, grades and assists the student as necessary. GET is good for one year and must be refreshed so that workers maintain skills and are updated on program changes which may have occurred.

Radiation health risks	**Protective clothing use**	**Whole body counting**
Natural background	**Respiratory protection**	**Entry/exit procedures for accessing the radiologically controlled area**
Naturally radioactive materials	**Radioactive material control**	
Contamination control techniques	**Release of items from the RCA**	**ALARA techniques**
		Radioactive waste reduction

Fig. 28 - Topics included in initial GET training

GET also includes training on safety, emergency plans and responses, quality assurance, fitness for duty, noise control and protection, confined space access, industrial chemicals, heat stress, first aid, and plant specific information which management wants employees to know.

Training for Radiation Protection Technicians who work at a power reactor is required by the Technical Specifications of the plant and is prescribed by INPO. The minimum training required by the Technical Specifications is that specified by ANSI 18.1-1971. This standard requires a high school diploma plus two years of related work experience which includes one year of technical training. Some Technical Specifications require RP technicians to meet ANSI 18.1-1976 which requires three years of work experience as an RP technician.

The RP technician is one of those positions for which a power plant must have an INPO accredited training program. The objective is to have RP technicians who are trained and qualified with the skills and knowledge needed to implement a radiation protection program in support of safe and reliable plant operation.

Topics that are included in the initial training, in addition to all the material covered in GET, are listed in Figure 29. RP technicians are qualified and signed off in

RP discipline related information about plant layout, plant systems, systems operations, records requirements and management, procedure and document issue and control, plant communications, procurement of materials, industrial safety, first aid

Theory and practice of radiation protection and the application of ALARA philosophy and techniques

Plant components, plant systems and associated radiological hazards, relation and interaction of plant systems which handle radioactive materials

Radiological surveys, expected radiological conditions during routine operation, refueling outages and emergency conditions, analytical methods and techniques, selection and use of survey equipment, survey documentation, posting of radiological survey results and establishment of appropriate warnings, barriers and controls

Fig. 29 - Topics covered in initial radiation protection technician training

a Qualification Manual for each task they are permitted to do independently. An "unqualified" RP technician may do any radiation protection task so long as there is direct supervision by a person already qualified to perform the task independently. Working with an already qualified RP technician, supervisor or trainer is part of the on-the-job training program used to qualify personnel for new tasks. Each power plant also has an ongoing training program which regularly:

- reviews material from the initial training,
- reviews plant specific and industry-wide incidents and lessons learned,
- reviews changes to the RP program,
- reviews infrequently used and difficult skills, and
- develops supervisory and leadership skills.

Contractor and temporary RP personnel must have and meet the same qualification standards as regular plant employees. Nuclear power plants typically encourage radiation protection technicians to seek NRRPT® registration. Some plants pay regular RP technicians a bonus for NRRPT® registration and nearly all plants pay an additional $1 to $2 per hour for contract RP technicians who are NRRPT® registered.

Performance Indicators

Regulation of power reactor safety was relatively prescriptive in the post-TMI era. For example, equipment which was important to reactor safety often had specific tests and test frequencies called out in regulations or Technical Specifications. Experience, accumulated test data, and probabilistic risk analyses have now demonstrated that some of the testing was unnecessary or even counterproductive to nuclear safety.

An example in the health physics field is the post TMI view that any internal exposure to radioactive material should be avoided at essentially any cost. This position drove the industry to apply massive respiratory protection programs to avoid radioactive material intakes and uptakes. It made ALARA and physical health considerations subservient to the goal of preventing internal exposures at the cost of receiving additional total exposure and suffering such problems as heat exhaustion. A movement toward risk-based regulations corrected this situation when 10 CFR 20 was revised and became effective at the beginning of 1994. The new regulation emphasized total effective dose equivalent (TEDE) control and implemented ICRP 60 recommendations. The result was a balancing between potential internal exposure and the possible additional external exposure which would result from using respiratory protection to avoid the internal exposure. There was a tremendous reduction in respiratory protective equipment use because most TEDE ALARA evaluations favored accepting small internal exposures in exchange for avoiding a larger external exposure. For example, one large three unit reactor site reduced respirator use from 6000-7000 per year to fewer than 10 per year. Because particulate contamination at power reactors is typically large (> 10-20 microns) pieces of oxides, nearly all intakes are by ingestion. Elimination is via the feces, occurs in a few days, and typically produces minimal doses.

The Nuclear Regulatory Commission has been moving away from a prescriptive based system and toward a risk-based system of regulations over the past few years. This was driven by long years of plant operating experience, by the industry's Nuclear Energy Institute (NEI), by academic studies, by the former head of NRC, Dr. Shirley Jackson, and by budgetary restraints. In effect, the NRC has been required to provide the same or improved public protection at a lower cost.

A transition to risk-based regulations for all aspects of commercial nuclear power occurred in 2000. Reactor Safety is divided into seven "cornerstones" (shown in Figure 30) and each cornerstone has some relevant "indicators." Each indicator is graded as green, white, yellow or red and is based on plant performance. The color of the indicator will be determined by applying the Significance Determination Process (SDP) to inspection findings. A green SDP rating is fully acceptable, a white rating will lead to increased regulatory attention, a yellow rating has a required regulatory response, and a red classification is unacceptable. Performance Indicators (PIs) are submitted to NRC each quarter by each power station. Subsequent inspections by the NRC confirm that PI information is complete and correct.

The occupational radiation safety cornerstone includes (1) violations of Technical Specification high rad area controls, (2) very high radiation area occurrences, and (3) unintended exposure occurrences.

NRC performs regular "baseline inspections" at each power plant. These inspections focus on (1) areas where performance indicators don't fully cover a cornerstone, (2) verification of the licensee performance indicator program, and (3) the licensee's problem identification and resolution program. More details about the NRC

CORNERSTONE	INDICATOR
Initiating events	Unplanned scrams Scrams with loss of normal heat removal Unplanned power changes
Mitigating systems	Safety systems not available: Specific emergency core cooling Emergency electric power
Integrity of barriers to release of radioactivity	Fuel cladding (measured by radioactivity in reactor cooling system) Reactor cooling system leak rate
Emergency preparedness	Emergency response organization drill performance Readiness of emergency response organization Availability of notification system for area residents
Occupational radiation safety	Compliance with regulations for controlling access to radiation areas in plant Uncontrolled radiation exposure to workers greater than 10 percent of regulatory limit
Public radiation safety	Effluent releases requiring reporting under NRC regulations and license conditions
Physical protection	[*Not available in the public domain*]

Fig. 30 - NRC performance cornerstones and indicators

Performance Indicators and the quarterly status of each plant is available at **www.nrc.gov/NRR/OVERSIGHT/ASSESS/.**

Self-Assessment

Self-assessment has a critical role in the operation of a nuclear power plant because it is now an integral part of the regulatory process which has been transferred from the NRC to the licensee. Self-assessment is the engine which drives continuous improvement. Each licensee must use the self-assessment process to document site performance versus the Performance Indicators. NRC inspections now focus substantial attention on how effectively the licensee implements self-assessment and how effective corrective actions are. Ineffective corrective action is a basis for NRC citations and fines.

In response, some nuclear power plants dedicate specific personnel to the self-assessment process. For example, one plant has assigned a senior HP engineer to

lead the self-assessment effort in HP. Information from audits, self check programs, and job and facility observations are combined to assess the "health" of the program and the status relative to the Performance Indicators. All supervisory personnel are required to perform and document monthly observations of work in radiologically controlled areas. A first line supervisor and some HP technicians assist with a quarterly analysis of the data and comparison with the PIs.

Control of Radiation Dose to Personnel

Plant Design

Plant layout and design are important aspects of each nuclear power plant ALARA program. Early plants such as those constructed in the 1960s tended to be relatively small and to have many systems and pieces of equipment (e.g. pumps, valves, controls, instrumentation) together in a single room or an open area. Once "contaminated" with radioactive material from normal operation, routine servicing of any component in the area was adversely influenced by the radiation field from each other component in the area.

U.S. NRC Regulatory Guide 8.8 and other ALARA documents provided guidance on how to design nuclear power plants with the incorporation of ALARA principles. ALARA must be designed into a plant; it can't be treated as an "add on" after construction. The most basic aspects of ALARA design include (1) permanent shielding for the radiation types and energies produced during normal and emergency operating conditions, (2) compartmentalization of systems and equipment, (3) provision for lifetime maintenance, (4) selection of materials and alloys to minimize radiation fields, and (5) control of airborne radioactivity.

Permanent shielding considerations for a BWR are different than for a PWR because the BWR experiences much higher levels of N-16, a radionuclide with a 7 second half-life and a gamma energy of about 7 Mev, and because the BWR secondary plant is always radioactive. The presence of N-16 in the secondary (turbine) plant of a BWR requires that the turbines also be heavily shielded. The secondary plant of a PWR requires little or no radiation shielding.

Modern plant design places radioactive piping in heavily shielded tunnels and puts individual plant components in separate, shielded compartments. For example, a pump and the associated piping, valve(s) and controls would likely be in adjoining, individually shielded compartments. Instrumentation is often located in the hallway outside the equipment cubicles. This permits the instrumentation to be calibrated in a low radiation, uncontaminated area and supports servicing of the other components under low exposure conditions.

Plant areas expected to have airborne particulates or radioactive gases should be operated at reduced pressure so as to direct the radioactivity to filters, charcoal beds, delay lines or the plant vent stack. Plant design and ventilation system balance

should move air from the less to the more contaminated areas to reduce spreading of contamination.

Access Control

Administrative controls are a second major ALARA tool. Administrative controls are required by plant Technical Specifications and are detailed in written and approved procedures. The radiation protection program at a power reactor typically includes 25-100 written and approved procedures which detail every aspect of the program including organization, administration and implementation. For instance, a separate procedure often details the operation, calibration and maintenance of each survey instrument brand/model. One or more procedures control access to the restricted area (RA), to the radiologically controlled (RCA) area, and to radiation, high radiation and very high radiation areas. A photo badge is typically required for access to the owner controlled and restricted areas of the facility. A photo badge, a security background check, a TLD or electronic dosimeter, and a radiation work permit (RWP) are often all required for accessing the RCA.

Every power reactor uses a radiation work permit system to authorize and control work inside the RCA. An RWP is a radiological prescription; it specifies what work may be done, where the work will be done, what dosimetry is required, what protective measures (clothing, respiratory protection, engineering controls, dose and dose rate limits) are imposed, the radiological conditions, any special training, and the requirement for a pre-job brief. A sample of an RWP was included in Chapter 11. To use an RWP, a worker typically requires (1) general employee training, (2) a whole body count, (3) personnel dosimetry such as a TLD plus a pocket dosimeter or electronic dosimeter, (4) a briefing on the expected radiological conditions and the limits of the RWP, and (5) special training for certain jobs. The worker must sign onto the RWP, acknowledge having read and understood the imposed controls, and agree to comply with the requirements.

Many power reactors use automated, computer controlled systems to verify workers' qualifications as they sign onto an RWP. Access to the RCA is also typically assisted by a computer system which records the RWP and electronic dosimeter used, sets the dose and dose rate limits and alarms prescribed by the RWP, and reads and records the electronic dosimeter on exit. These entry records are maintained and can be searched for information about the specific entry or the job. ALARA reports can be developed from the individual worker's dose and entry records to show cumulative dose and time for a job.

Surveys

Federal regulations, Technical Specifications, and written procedures collectively specify when and how radiation, surface contamination and airborne contamination surveys will be performed. Survey results must be disseminated so that workers and HP technicians have current radiological conditions information. Some

plants have developed intranet systems for the rapid, site-wide distribution of survey information. Computers are placed at strategic locations during a major plant outage to make current radiological information immediately available. Digital photos of equipment and areas are also distributed over the intranet. Electronic scanning of survey and other records is also growing in use as a means of rapid dissemination.

Emergency plans specify required on-site and off-site survey and monitoring capabilities and equipment inventories necessary for response to a plant emergency. Most power plants have strategically placed and separate emergency response kits with survey meters.

The typical nuclear power station has portable radiation survey equipment for measuring environmental (microrem/hour) to accident level (~20,000 R/hour) gamma radiation; neutron radiation to 10-20 rem/hour; airborne particulate, iodine and noble gas; and surface alpha and beta contamination. Count rooms have gamma spectrometers and laboratory alpha and beta counters. Nuclear power plants also have programs and equipment for the calibration and servicing of their survey and lab equipment. Many facilities lack neutron calibration equipment and use the services of facilities such as the University of Arkansas.

Among the more common gamma survey instruments are the Bicron Micro-Rem (see Chapter 12, Figures 14 and 15) for environmental level measurements. This instrument is popular because it uses a plastic scintillation detector with a tissue equivalent response rather that a sodium iodide detector which is strongly energy dependent. Mid-range instruments include both Geiger-Mueller and ion chamber detectors. The Eberline RO-20 (and it's predecessors, RO-2/2A) and the Bicron RSO 50/500 are popular ion chamber instruments. Smart instruments such as the Eberline 600 are gaining popularity because they accept many detectors and store the calibration parameters with the detector. High range gamma instruments are dominated by the Teletector (Figure 31), an extendable GM instrument with two detectors and ranging to 1000 R/hour, and the Eberline RO-7 with a high range detector good to 20,000 R/hour. A requirement for a hand-held instrument such as the RO-7 emerged in the post TMI era, but few folks ever plan to use one in a field of 1000s of R/hour. The instrument has a waterproof housing and finds considerable use for underwater surveys in reactor vessels.

Removable surface contamination is typically assessed by wiping a 100 square centimeter area with a cloth or paper disc and subsequently counting with a count rate instrument and a pancake GM detector or a sample changer system. The pancake frisker probe is usually assumed to have a 10% counting efficiency and the swipe is often assumed to have a 10% collection efficiency. Ludlum, Bicron and Eberline all make similar count rate instruments used for frisking. These are the "work horse" instruments for contamination measurement and control. When heavily contaminated surfaces are assessed, the swipe may be measured using an ion chamber (open window) and the result expressed in "mrad or rad smearable." Internal surfaces in the primary system and reactor components can easily produce swipes which measure 1-10 rad. Large areas (e.g. floors, walls, large components) may be wiped with Masslin or a similar brand of slightly oiled dust cloth and frisked for contamination.

Courtesy, Eberline Instrument Co.

Fig. 31 - The Eberline Teletector high range gamma survey instrument

This technique is effective for monitoring floors for low levels of contamination and for hot particles.

Airborne particulate and iodine contamination is collected by passing a large volume (e.g. 40-200 cubic feet) of air through a 47 mm filter plus charcoal cartridge combination. These grab samples are used to assess an area or a certain phase of a job (e.g. opening a primary system component) The particulate filter and iodine cartridge may be field checked using a frisker for a “go-no go” assessment. Quantitative results are obtained by counting the filter for gross beta-gamma or by doing a gamma

spectroscopy analysis using a GeLi or HpGe detector system. The iodine cartridge is best assessed on a gamma spectrometer system. Iodine is collected near the inlet surface. Therefore, it is important to keep track of the inlet side of the charcoal cartridge for proper orientation on the spectrometer. Both RaDeCo and Hi-Q make air samplers used in the industry.

Continuous air monitors (CAMs) are used where long-term monitoring is desired with an alarm function for abnormal conditions. Thermo Eberline AMS-4 air monitors offer features including readout in uCi/cm^3 or DAC. These instruments have a radio transmitter and will transmit data to a remote location for tracking, trending and alarm. See Figure 32.

It is always important to consider whether an explosive (hydrogen gas) atmosphere could be present before starting a standard air sampler or air monitor. Hydrogen in gas radwaste systems and in freshly breached primary systems has been the cause of explosions when an air sampler was started. Explosion proof air samplers eliminate this concern.

Sampling for noble gas is accomplished by emptying a water filled container or by opening an evacuated container in the area to be sampled. Either of these techniques is simpler and quicker than setting up an air pump and drawing an air volume through a container.

Fig. 32 - A popular continuous air monitor, the AMS-4

Courtesy, Eberline Instruments

Dose Measurement and Assignment

The primary personnel dosimeter in U.S. nuclear power plants is a multi-element TLD. About half of the plants/utilities operate a NVLAP accredited TLD lab and the remainder of the plants buy this service from a vendor. As discussed in Chapter 8, federal regulations [10 CFR 20.1501(c)] require that anyone who processes personnel dosimetry must be NVLAP accredited.

The large majority of radiation exposure received by workers at nuclear power plants is deep dose delivered by gamma radiation. Dose from neutron exposure

occurs infrequently and represents less than one percent of the total deep dose received by plant personnel. Dose to the skin, eye lens and extremities are normally a very small fraction of personnel dose. Loose contamination and immersion in a noble gas atmosphere are typical sources of shallow dose. Exposure to the hands during refueling outage work is the most common extremity exposure.

Most U.S. power plants use electronic dosimeters as the secondary dosimetry system and to measure exposure for each RCA entry. This provides "minute-by-minute" control of exposures and the ability to allocate dose among many jobs and RWPs. Automated recording of the entry-exit dose readings simplifies gathering this data for later analysis. Most electronic dosimeters used in U.S. plants are either Thermo-Eberline (formerly Siemens) or MGP Instruments. As discussed in Chapter 8, electronic dosimeters are not presently accepted as a primary dosimeter of record in U.S. plants. Some plants in England use the Thermo-Eberline dosimeter for dose of record.

Pocket ion chambers (PICs) still find some use in U.S. nuclear power plants, but they tend to be relegated to emergency response kits and other backup uses. Few U.S. plants still use PICs for day-to-day dose records and controls.

Time and dose equivalent rate are often used to assign personnel doses which occur, for example, during PWR containment entries at power or while manipulating a neutron source. This approach is still used by some plants for control of dose under high exposure rate conditions such as exist inside a steam generator channel head (e.g. 5-25 R/hour).

10CFR20.1201(a) requires that occupational doses from radiation shall not exceed 5 rem (0.05 Sv) total effective dose equivalent (TEDE). Recall that TEDE is the sum of EDE and CDE and that EDE may be conservatively approximated by DDE. Most NRC licensees presently measure deep dose equivalent (DDE) with one or more external dosimeters (TLD or film) and report the highest value as the effective dose equivalent (EDE). Both nuclear power plant and medical licensees have begun to use methods to estimate EDE as a more realistic measure of personnel dose than the DDE is. EPRI developed a method for estimating EDE by using two dosimeters and an algorithm to derive EDE. NRC approved the methodology for use by one power plant licensee where steep radiation gradients exist, but limited use to no closer than 12 inches. Another power plant licensee has applied to use a different algorithm and combination of dosimeters to assess EDE. The use of EDE will lower the total dose assigned to some workers because the more conservative use of DDE will be eliminated.

The issue of skin dose from hot particle exposure was fully resolved when NCRP issued Report 130 plus NCRP Statement 9 and recommended that dose from a hot particle or other contamination be averaged over 10 cm^2. NRC acted on this technical recommendation and incorporated it into 10 CFR 20. The computer code Varskin 3 is available for purchase from Oak Ridge National Laboratory and is used at all power plants for calculation of skin dose resulting from hot particles or distributed contamination.

Operational ALARA Practices

Every nuclear power plant has an ALARA Committee which is typically composed of the managers of each work division (e.g. Health Physics, Maintenance, Operations, Engineering, Chemistry). This Committee sets the ALARA policy and goals for the station and meets on a regular frequency to review individual and collective radiation dose. ALARA considerations are an integral part of the way a plant is operated. For example, ALARA influences the work content and sequence during a refueling outage and the selection of new and replacement plant equipment. ALARA performance is rated by INPO, ranked by quartiles, and then published for the industry to see. NRC also reviews ALARA practices and results on a frequent basis.

The most important and effective control over radioactivity in the reactor and associated systems is chemistry. Both BWR and PWR plants have "optimum" reactor chemistry operating parameters intended to minimize system corrosion, minimize transport of materials to/from the reactor core, and promote fuel cladding integrity.

A BWR uses high purity water as the moderator and the heat transfer medium. The reactor water cleanup system continuously filters and ion exchanges a fraction of the total coolant to maintain water quality specifications. Objectives of the chemistry controls are to (1) minimize system corrosion, (2) prevent stress corrosion cracking, (3) minimize fuel cladding failures, and (4) control corrosion product deposition, activation, and resultant radiation fields.

Hydrogen gas may be added to the BWR coolant system to suppress the radiolytic production of oxygen and hydrogen peroxide, both of which promote system corrosion. Zinc additions are also being used to reduce the deposition of activated corrosion products throughout the reactor system.

PWR reactor chemistry is quite different from BWR chemistry because PWRs use boron (as boric acid) dissolved in the reactor coolant to control reactivity. The boron chemical shim reacts with neutrons to remove them from the reactor. It permits ongoing reactivity control without needing frequent control rod movement. Boron concentration typically begins a fuel cycle at 1500-2000 ppm and is decreased by about 3 ppm per day to offset the core burnup. This boron reduction is accomplished by a feed-and-bleed (dilution) process.

Work Planning

Work planning in a nuclear power plant includes the use of temporary shielding, engineering controls, pre and post-job briefings, training, mockups, review of prior lessons learned, and documentation for the next time the job is done. Whatever ALARA measures are recommended by the work planning process, a TEDE ALARA evaluation is performed to judge whether to implement the practice. One of the most common TEDE ALARA evaluations is a determination of whether to apply respiratory protection. Engineering controls are employed for a number of purposes as indicated in Figure 33.

Reduce the radioactivity present

Reduce the radiation level

Reduce the time to do a job

Increase the distance between the source and the worker

Fig. 33 - Some uses of engineering controls

Removable external surface contamination is controlled by normal decontamination methods or by fixing in place. Internal contamination (e.g. inside a pump, valve, pipe) may be controlled either by isolation or removal. Isolation might involve placing caps or seals on/in an item to prevent the removable contamination from coming in contact with the worker. Removal may employ physical or chemical techniques.

Airborne radioactivity controls may involve surface decontamination, HEPA filtration to reduce particulate and/or iodine, purging the space, or working the job wet. Any of these approaches is generally preferable to using respirators.

Ambient radiation levels are frequently reduced by installing temporary shielding such as lead blankets. Temporary shielding normally requires a static and seismic engineering evaluation prior to installation to prevent collapse onto a safety related component or structure. Some components (e.g. steam generators, tanks) can be filled with water to increase the shielding and reduce the radiation. Working a job under water is common and applies well to BWR control rod drive servicing and PWR reactor pump seal rebuilds.

Reducing the collective time to do a job is an important ALARA tool. Remote monitoring (discussed below), job specific training and practice, special tools and equipment, scheduling and sequencing to avoid interference with other work, and robotics are some of the means used to shorten job duration. Nuclear power plants use many special tools to perform difficult and exposure intensive work. For example, machines can remove or replace multiple reactor head studs at a time and require minimal human physical work. Equipment can be set up by a person to cut a pipe, prepare the end for welding and complete the weld. All but the setup is done remotely. Wall-walking equipment decontaminates reactor cavity walls and is controlled from above where contamination and exposure are minimal. Underwater robots inspect, test, video and document work inside a reactor vessel, a tank, a pipe or other highly radioactive component.

Planning, pre-job briefing and post-job reviews are important administrative tools for performing work with radiation exposure ALARA. Planning is the phase in which a job is carefully thought through, equipment and techniques are selected, exposure estimates are made, the numbers and types of workers are determined, and training is provided. Pre-job briefings bring the workers and supervisors together to

review exactly who will do what and when, what conditions or situations are cause for stopping the job, what contingencies are in place if a problem arises, what individual and collective exposure is anticipated, what radiological conditions are present, and what radiological controls are in place. The post-job review is a time for workers and planners to compare the plan with the performance and to identify and document what went well and opportunities for improvement. The records generated from the job history should be available for use the next time the job is to be done. Repetitive application of this process leads to improved job and ALARA performance over time.

Remote Control and Monitoring of Radiological Work

Most nuclear power plants use some remote monitoring and control of radiological work. This varies from a few video cameras and some hand-held radios to complex computer systems which integrate video (color, pan-tilt-zoom), video recorders, voice communications (radios, cell phones), teledosimetry, remote radiation and airborne radioactivity measurements. Figure 34 shows the central control area for such a system installed at San Onofre Nuclear Generating Station. Bartlett Nuclear Services provides the RMS software package for integrating the systems, displaying live

Courtesy, Southern California Edison Co.

Fig. 34 - The central control area for the San Onofre Nuclear Generating Station remote monitor system

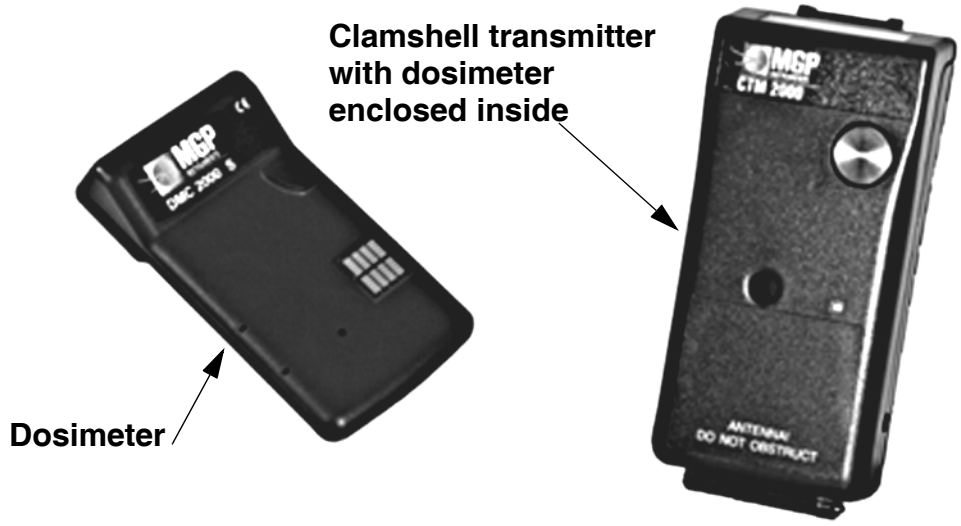


Fig. 35 - The MGP Instruments teledosimeter system

Courtesy, MGP instruments

personnel exposure and exposure rate information, and alarming when an administrative limit is being approached. Use of these remote radiological control systems can save personnel exposure and reduce the number of HP technicians required to monitor and control radiological work. Thermo-Eberline and MGP Instruments manufacture the common teledosimetry systems presently used in the industry (Figure 35).

The SAIC PDE-4 teledosimeter with one detector in the dosimeter body and four additional detectors is still used by a few plants. Such equipment is used for high dose rate jobs (e.g. steam generator channel head work) with variable radiation fields. The multiple detectors can be placed to simultaneously monitor different whole body and extremity locations and transmit the data to a remote location for monitoring and control.

Control of Radioactive Material

The primary focus of the radioactive material control program has been to prevent the inadvertent release of radioactive material from the site. Loss of control of even tiny quantities of licensed radioactive material have brought regulatory action from the NRC, concern and assistance from INPO, and have occasionally led to a public outcry against the plant. Most of the lawsuits brought against nuclear power plants have involved the alleged or actual loss of control of radioactive materials.

Radioactive material control is a cornerstone of Public Radiation Safety. Included in this category is unintended release of radioactive material, the effluent release program, the environmental monitoring program, transportation of radioactive

material, breach of a radioactive material package, noncompliance with a certificate of compliance, 10 CFR 61 waste classifications, and refusal of a burial facility to accept a shipment.

Administrative

Just as with every other aspect of operating a nuclear power plant, the release of material from the RCA and the RA of a power plant is controlled by a written procedure on which all plant personnel are trained. The fully qualified RP technicians are the workers who implement the procedure and perform the surveys which authorize removal of items. This is a major responsibility!

It is common for NPPs to have a single release point for materials leaving the RCA. This location is staffed with radiation protection personnel who can survey and release items. Having a single entry point greatly simplifies control of radioactive material. An occasional plant uses a two-stage monitoring program where items are checked as they leave the RCA and again as they leave the RA.

Davis-Besse - A Near Disaster

Davis-Besse entered a refueling outage in February 2002 and began inspection of Inconel 600 reactor head nozzles. The reactor head had large amounts of boric acid on the external surface and this material had been present for a number of years. A number of nozzle cracks were detected and a football-sized cavity in the 8 inch thick carbon steel reactor head was discovered. Wet boric acid promotes rapid corrosion of carbon steel. This cavity was adjacent to a CRDM nozzle and extended through the carbon steel head to the primary system stainless steel liner. The approximately one-quarter inch thick stainless steel liner was acting as the primary system boundary for a 600° F/2200 psig system. The liner was bulged outward and was not designed to withstand the primary system pressure. This situation was poised to potentially cause the most serious reactor accident in the U.S. since the 1978 TMI accident. The industry, the NRC, and INPO were stunned that such a failure could occur without early detection.

Root cause analyses indicated that the Inconel “600” nozzle(s) probably cracked in 1990 ± 3 years, that the crack propagated through-wall in the 1994-1996 period, that the leak was not identified during 1998 and 2000 inspections, and that discovery occurred in 2002. It was determined that the plant culture put electrical production and minimization of operating costs above nuclear safety and that a questioning attitude was absent. Davis-Besse was out of service for over two years while the reactor head was replaced and many repairs and upgrades were performed. Essentially all management and supervisory personnel were replaced. Go to www.nrc.org and search on “davis-besse reactor head” for extensive reading on the subject.

Future Directions

RP 2020; A Plan for the Future

The U.S. commercial nuclear power industry envisions adding 60,000 megawatts of capacity between now (2005) and 2020. Fifty thousand of these megawatts will come from new plant construction and 10,000 will come from increased efficiency and performance of the current 103 operating plants. This vision is called Vision 2020 and was developed shortly after the Bush administration released a comprehensive national energy policy that includes recognition of the need for and benefits of nuclear power. This vision is supported by the DOE's Nuclear Power 2010 program that proposes construction of new nuclear plants before the end of the decade. This DOE program offers financial incentives and has induced three consortia to file the first documents with NRC that indicate an interest in constructing new nuclear plants. One group is led by Dominion, another by TVA and the third is a collection of eleven energy companies. If constructed, the new plants would be built at existing nuclear power plant sites (possibly TVA's Bellefonte, Dominion's North Ana, and Entergy's Grand Gulf) to avoid the time and costs associated with "qualifying" additional sites for a nuclear power plant. Adding to existing sites is economical and is practiced in other countries, especially South Korea, where up to 12 plants are planned for existing sites (e.g. Ulchin).

A sub-set of Vision 2020 is RP 2020 that deals with the radiation protection issues that must be addressed before 2020. Industry professionals met during 2004 and enunciated the RP 2020 Mission as "Reshape radiological protection at nuclear power plants to achieve significant improvements in safety performance and cost-effectiveness." RP 2020 strategies and the organizations responsible to accomplish this are (1) to improve execution of RP fundamentals (INPO), (2) to improve RP technologies utilization (EPRI), (3) to reduce radiation dose fields (EPRI), (4) to assure future RP work force needs are met (NEI), (5) to establish a stable, predictable safety-focused regulatory environment (NEI), and (6) to standardize RP practices (INPO).

One of the knottiest industry problems is staffing for the future. Many of the U.S. nuclear power plants began operation during the mid 1980s. Workers that began a career during that period will be approaching retirement within another ten years. College and university programs in radiation protection had been eliminated or combined with other disciplines (e.g. environmental health and safety, engineering) because there were insufficient students to justify the programs. However, in view of the new plant license applications before the NRC, professional and technician programs are being rebuilt and students have been attracted. The future of nuclear power is optimistic at this time. NEI has been working successfully on this issue.

Deregulation and Consolidation

Deregulation of the electric utility industry is bringing major changes to the nuclear component of the electrical generation business. Some nuclear plants have operated because they were part of a regulated utility rather than because they were

economically viable in a competitive market. Under the old model, utilities earned money based on their capital expenditures and the utility commission practice of paying a reasonable rate of return on the investment. Clearly, the incentive for a company was to invest heavily and work hard to secure a high rate of return on the investments. Deregulation changes that formula, establishes an open market for electricity (just like other commodities), and places the emphasis on generating power at a competitive rate. All generation forms (nuclear, gas, oil, coal, hydro, renewables) must now compete in the open marketplace. This change from the original model for plant cost recovery yields large "stranded resources" and a financial dilemma for utility management. Public utility commissions have recognized the problem and made provision for some utilities to recover long ago committed capital via "transition fees" or write-offs.

Companies such as Exelon and Entergy have purchased a number of NPPs to add to their already large and successful nuclear fleets. Purchases and consolidations will likely continue over the next few years until operating NPPs are concentrated under a few large electrical generation companies. Among the critical issues when a NPP changes hands are (1) continued focus on plant operation and nuclear safety, (2) retention and management of essential personnel resources, and (3) careful control over the change in management processes.

Decommissioning

Humbolt Bay 3, Millstone 1 and San Onofre 1 are in the decommissioning process at this time. Dresden 1, Hanford 1, and Zion 1 and 2 are shut down and await decommissioning.

The 103 operating NPPs hold licenses which expire from 2006 to 2035. Figure 36 shows when licenses for existing NPPs will expire. These dates assume that each plant will apply for and receive construction recapture. When a plant is initially licensed, it is for 40 years of operation. But some of that time is consumed during construction. It is necessary to apply to NRC to "recapture" that lost part of the license. Plants which were under construction just after the TMI accident had their completion delayed and can recapture up to 10 years of operating time. There is also provision for a license extension of up to 20 years beyond the original 40 years. Calvert Cliffs and Oconee were the first plants to apply for the 20 year license extension and did so in 1998. As of mid-2003, 14 plants had received license extensions and more than 40 other plants were in the process. All operating U.S. NPPs can be expected to apply for license extensions.

Projections of a few years ago that up to one third of the operating power reactors in the United States would be shut down for decommissioning in the early 2000s are not materializing because the industry has steadily and substantially improved its economic and safety performance. Of possible future value to NPPs is the fact that they do not contribute to the problems of acid rain (sulfur dioxide emission) and greenhouse gases (carbon dioxide).

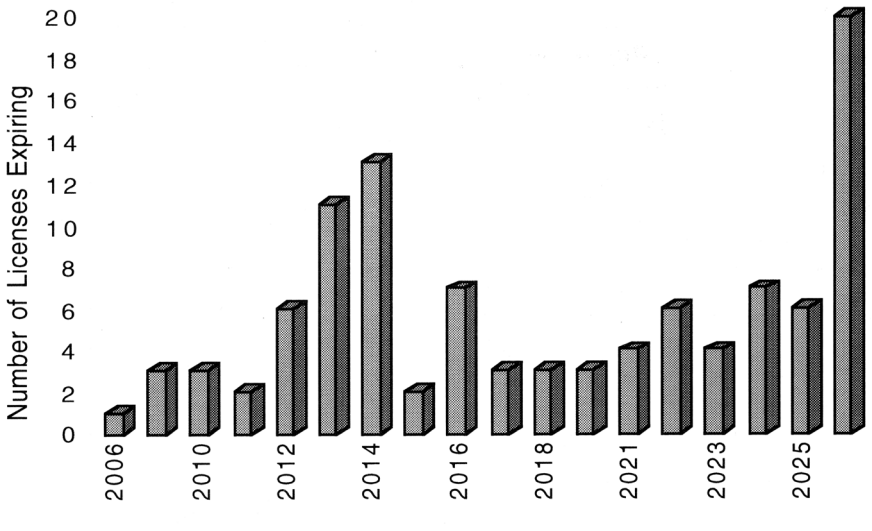


Fig. 36 - U.S. power reactor license expirations by date

Radwaste Disposal

The Low-Level Radioactive Waste Policy Act of 1980 which required formation of compacts among states and which envisioned that each compact would develop a radioactive waste disposal facility has to be considered an abject failure. Compacts were formed, but no new waste disposal facility has been or presently appears likely to be built by the compacts. Power reactors are limited to sending radioactive waste to Barnwell, S.C. and to Energy Solutions of Utah. South Carolina politics closed Barnwell to all but Atlantic Compact states in 2008. Energy Solutions is licensed to accept Class A waste and presently receives most of the Class A radwaste from nuclear power plants.

Availability of Trained Personnel

The number of trained and experienced personnel available to the nuclear industry has been steadily declining and retirements are increasing. Reasons include decreased availability of Nuclear Navy trained personnel because the military program has shrunk, present employees are aging and retiring or, in some cases are leaving (especially HP technicians) for other work. A large fraction of students in graduate HP programs are foreign nationals who intend to return to their home country after graduation.

As of 2010, the word about the nuclear power recovery is getting out there. Enrollment in nuclear engineering and health physics programs has begun rising.

Two-year Nuclear Technology programs have entered a growth phase with approximately 20 new programs now functioning.

Also on the positive side, the Oak Ridge Institute for Science and Education (ORISE) salary survey for 2001 shows health physicist salaries increasing 6-8% over 2000. Nuclear engineers experienced similar salary movement. These increases are greater than for chemical and mechanical engineers during the same period. Increasing starting salaries and job openings will assist in attracting students to the nuclear industry.

Health Physics at Research Reactors

Introduction

As of 2010, there were 33 nuclear reactors designed for research, testing, and teaching licensed to operate in the United States. They were distributed among 22 different states. The largest operates at 20 MW thermal. The U.S. NRC classes these reactors as Research and Test Reactors or RTRs. Internationally they are collectively known as Research Reactors. The number continues to decline. In 1991 there were 46 in the U.S. That number fell to 39 in 1997. Since 1958, 82 RTRs have been removed from service and decommissioned. The locations of these non-power reactors is shown in Figure 37. At present (2010), twelve research reactors are being decommissioned in the U.S. There are about 100 NRC licensed operators for the nation's RTRs. These licensees must renew at six year intervals.

The U.S. trend toward fewer non-power reactors reflects the world situation. In 1975, worldwide there were 373 units. In 1990 the number fell to 323. As of 2011, the number has been further eroded to 230.

The most popular design type is the pool reactor discussed in Chapter 6. Worldwide, this design accounts for 28% of the RTRs. TRIGA reactors and tank reactors account for an additional 28%. The TRIGA (for training, research, isotopes General Atomics) uses a uranium zirconium hydride fuel that allows the reactor to operate both in steady-state mode and pulsed mode. A TRIGA pulse can exceed 3,000 MW of power, comparable to power reactors but only for a small fraction of a second. U.S. TRIGA reactors use low enrichment fuel and run at less than 2 MW steady state. Even though the size, complexity and power level are many times smaller in a research reactor compared to a power reactor, it still has the ability to tax the limits of a radiation protection technologist in terms of health physics problems. Many of the currently licensed facilities are operated for experimental purposes so the configuration may change daily. Multiple penetrations are present, some with a direct line of sight to the reactor core.

An administrative Reactor Safety Committee is responsible for overall safety at the installation. Committee membership should include experts on reactor engineering, radiation protection, chemistry and reactor physics. At some research reactors, safety responsibilities are further subdivided into reactor operations and experimental

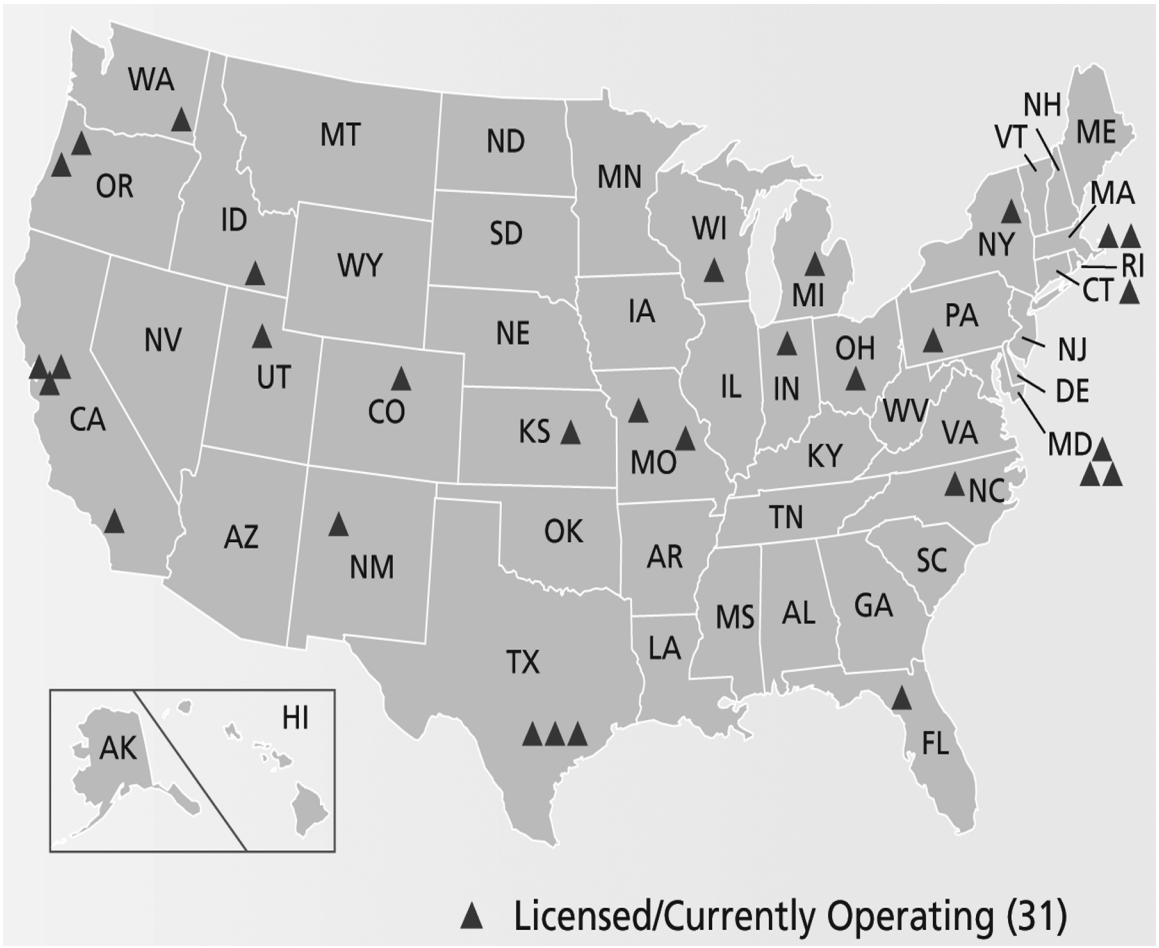


Fig. 37 - Locations of non-power reactors in the USA

operations. This latter category is concerned with the effects of each experiment conducted on the reactor. A sample or piece of apparatus with high reactivity or neutron absorption properties that is rapidly removed from the core region will cause a transient power spike which must be taken into account by the reactor operator. Core penetrations must be properly shielded before reactor start-up. Experiment rooms may have to be interlocked to prevent exposure of personnel to high radiation areas. Proper handling apparatus must be available for high activity samples that are removed following neutron activation.

RTR Health Physics Operations

Primary Shielding Integrity: The primary biological shield in a swimming pool reactor is the waterpool and associated thick concrete walls. Changes in water level

must be immediately corrected. The main pool is usually fairly watertight, but the external circulating and cooling pumps and associated plumbing for water cleanup (using an ion exchange resin bed) may spring a leak, leading to a drop in pool level. Water level alarms and area radiation monitors would alert operators in such a case.

Beam Tubes and Associated Shielding: The beam tubes consist of pipes pointing at the core and penetrating the main shield. They usually terminate at a shielded experimental room and are typically between 4" and 12" in diameter. Different configurations are used to maximize the desired radiation field emerging from the tube. A pipe, aimed toward the core but not pointed at any fuel elements, will have a high thermal neutron flux. Conversely, one directed at fuel elements will have a high proportion of fast neutrons and gamma rays.

The tubes usually have a remotely operated shielding shutter and in addition are equipped with an external shield plug when not in use. These two features are designed to prevent radiation streaming along the tube. Local shielding is used at the exterior end of the beam tube. Various experimental equipment is positioned here. Health physics monitoring is essential as changes of configuration of the local shielding can produce hot spots in the shielded fields. Proper choice of local shielding materials is important. As discussed in Chapter 3, thermal neutron interactions can produce capture gamma rays. This adds a gamma ray component to the radiation field. Gammas are reduced by use of thermal neutron shielding containing boron or lithium. The capture gamma rays for these shields are relatively easier to attenuate.

At the core end of the beam tube, the gamma dose rate may reach 10^8 R/hr and the neutron flux is typically in the range of 10^{13} n/cm^2-sec. At the experimental room end, the neutron flux might be 10^7 n/cm^2-sec. If personnel are prevented from entering the experimental area during reactor operation, a further savings in time and shielding can be effected.

The final potential problem of note related to the beam tube has to do with the removal of beam plugs. If the plugs have been in place for an extended period of reactor operation, then, the corrosion around the outside of the plugs will have become activated by neutron bombardment. This can create an airborne hazard when they are opened.

Pool Gas Releases: In light water pool reactors the two major contributors to radioactive gas production are argon-41 and nitrogen-16. The Ar-41 results from neutron capture by stable Ar-40 gas present in trace amounts in our atmosphere. Unfortunately, the research reactor contains numerous air pockets in the form of beam tubes, pneumatic transfer system piping, experimental cavities, etc. Similarly, capture of a neutron by an oxygen-16 atom in the pool water produces N-16. An adequate room air ventilation system is one way to handle the problem. A better way is to engineer features into the reactor design to minimize production and/or release of these gases. Fortunately, the half-lives are short for these two gases - 108 minutes for Ar-41 and 7 seconds for N-16. Pneumatic transfer systems for sample capsules can be powered by compressed nitrogen or carbon dioxide rather than compressed room air. If there is an air gap designed between the reactor vessel and the bioshield, it can

be purged with an inert gas. Irradiated samples containing air can be held for decay of the associated Ar-41 before opening them. In the case of the nitrogen-16, its extremely short half-life means that it is possible to reduce pool top releases by slowing down or diverting pool convection currents that would normally flow directly up from the core to the pool surface. Radioactive gas monitors near the pool surface and in release stacks give warning if the levels become excessive.

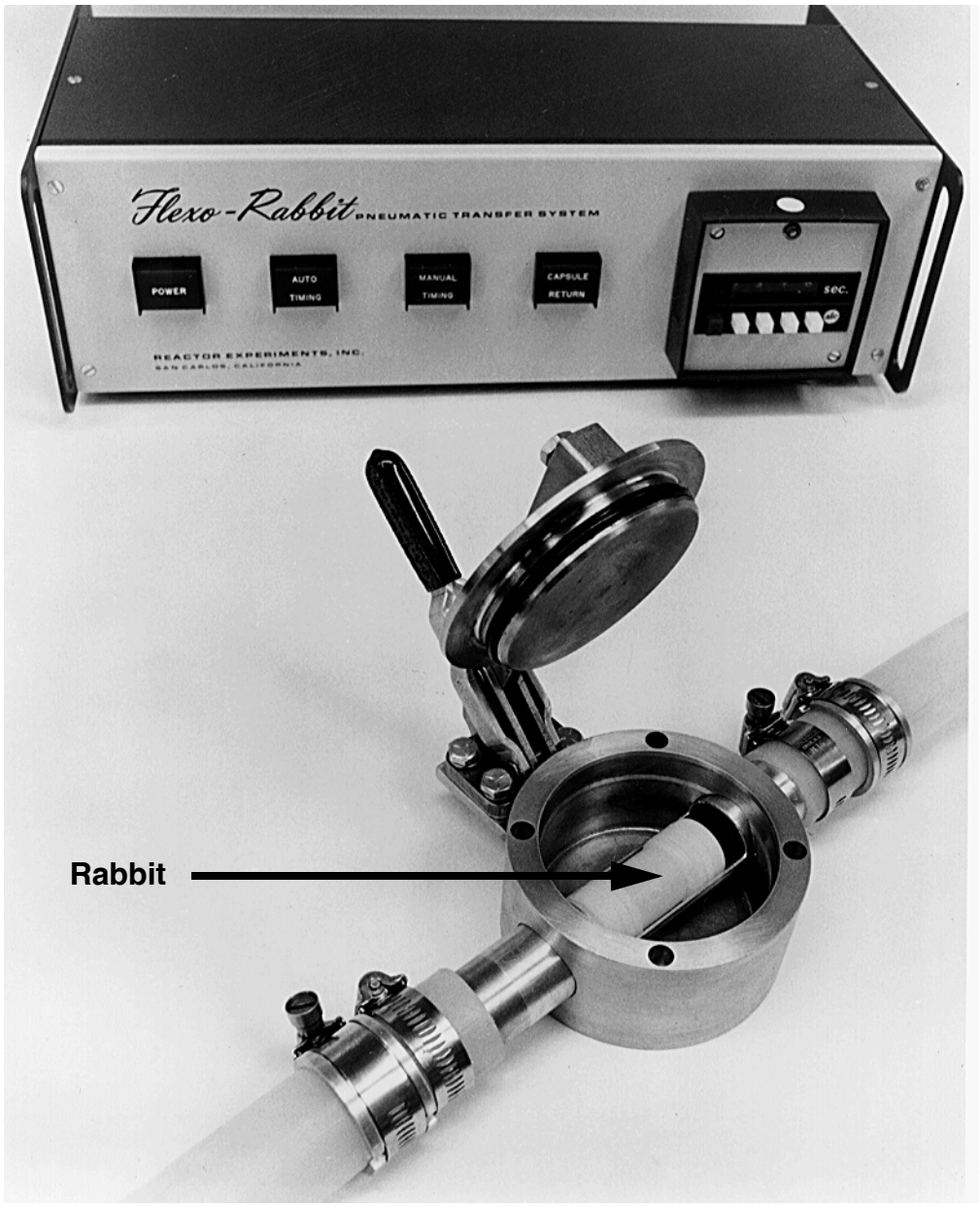
Rabbit Systems: Virtually all research reactors are equipped with a rabbit system - a pneumatic sample transfer system. A plastic or aluminum sample capsule, the rabbit, is moved in and out of the reactor core by compressed gas through a tubing system (see Figure 38). The tube is installed with bends in the line to reduce radiation streaming up the hollow pipe.

Often, the activated sample has high induced gamma activity upon extraction from the core. It is necessary to have a radiation monitor installed at the receiver end of the tubing system. The monitor can be connected so that if the sample exceeds a preset value on the monitor, the monitor causes the rabbit to be sent back into the reactor while additional provisions are made to handle it upon its return, such as installation of more shielding bricks around the receiver. Some installations have the option of several different receivers, each with a different amount of shielding. A switcher unit in the rabbit system is used to route the capsule to the desired receiver. Contact dose rates can easily exceed 100 R/hr immediately after the arrival of the rabbit.

The chemical and physical form of the sample plays an important role in the overall experimental safety analysis. Many materials, including solid metal chunks in some cases, will have their internal structure altered by the intense radiation fields in the reactor core. Such a sample might disintegrate under the high forces associated with the arrival of the rabbit at the receiver. This leads to the release of large amounts of airborne particulate activity, usually of short half-life. Many non-power reactor facilities use a fume hood or sealed glove box around at least one receiver station to minimize this problem. The glove box is also convenient if any chemistry needs to be performed on the sample.

Another way to minimize the release of degraded samples is to seal the original material in a quartz ampule. Such samples must be properly annealed or the quartz can break under irradiation. Another related problem is gas pressure buildup of the sealed sample during irradiation. These samples have been known to explode upon arrival at the rabbit receiver. On occasion, the activity can escape from the rabbit and contaminate the entire rabbit tube. Most tubing systems are designed to allow cleaning following such a disaster.

Handling A Large Sample: It is not uncommon for an experimenter to want to expose a physically large sample to neutrons or gamma rays. In the case of pool reactors, these samples can usually be accommodated by lowering them on a cord down through the pool water into the core. When the required dose has been delivered, based on reactor power level, sample location and exposure time, the sample is retrieved by hoisting it up on the cord. A health physics survey is usually conducted



Courtesy, Intertech Worldwide Corp., Woodside, CA

Fig. 38 - A research reactor rabbit system

at poolside. If the sample activity is excessive, it can be lowered back into the pool (away from the core) to allow time to decay down to an acceptable dose rate. A Teletector type instrument, with the probe extended, is a good way to keep technician doses ALARA.

If the sample is heavy, a pipe can be used instead of the cord to suspend it over the core region for irradiation. Make sure that the pipe has holes drilled along its length to allow it to flood with pool water. Otherwise a gamma and neutron channel

has been created up the hollow center of the pipe which can cause high doses to handling personnel in the vicinity.

In reactor designs without an open pool, large samples can be accommodated by placing them in line with a large area beam tube and then adding local shielding. Some reactor facilities have completely shielded rooms such as the MIT medical treatment reactor facility shown in Chapter 6.

Before samples are placed in the reactor, it is necessary to calculate the reactivity of the sample. Some materials have very high neutron cross sections (high chance of catching a neutron), and so when the sample is inserted, the operator must be ready to withdraw a control rod to compensate. When the sample is withdrawn, the process must be reversed or else the reactor power will undergo a surge which might exceed design specifications.

Reactor Fuel Management: Low power research reactors may have "lifetime cores" where the low burnup allows use of the same fuel for decades. (Recall that 1 gram of U-235 produces a megawatt-day of power so a 1 kilowatt research reactor operating 8 hours a day would take about 10 years to burn 1 gram ^{235}U). Since these fuel elements are not handled very often, it is all the more important to periodically check for corrosion or leaks. High power RTRs often shuffle fuel similar to the way fuel is handled in power reactors. Historically, RTR fuel has been exceptionally reliable. If an element suffers cladding failure, it is usually detected quite rapidly in pool reactors from air monitor readings. The moderator/cooling water is also monitored but has been demonstrated to be less sensitive to leak detection than the air monitoring. Once leakage has been detected, it is common practice to locate the leaker by sipping. Here, a detector is used to monitor water flow past each fuel element, one at a time.

No U.S. RTR has the security necessary to safeguard large masses of fresh reactor fuel so they keep only minimal amounts on hand. High power RTRs may receive a fresh fuel shipment every six to eight years. These facilities also maintain spent fuel pools for storing used elements. Health physics efforts are thus directed at detection of fuel cladding leaks and in maintaining water quality regarding contamination of the spent fuel pool.

Ion Exchange Resin Beds: The primary cooling and moderating water in an RTR is continuously cleaned up by passing it through an ion exchange resin bed. Typical radioisotopes found in the resin bed include C-14, Mn-54, Fe-55, Co-60, Tc-99 and Cs-137. The activity concentrations are much lower than in power reactors. Of course, this is due to the much lower power rating. It also reflects the more extensive use of aluminum components (rather than stainless steel in the power reactor). So there are a lot less iron, manganese and cobalt atoms subject to neutron activation. The aluminum is the source of the Na-24 found in the primary water. The resin bed may last for years before replacement is necessary. If the RTR design uses heavy water, efforts are usually made to recover the heavy water trapped in the resin bed at the time of its replacement.

Other Health Physics Duties: The radiation protection technologists at a research reactor have many other duties which are similar to those of power reactor HP technicians. These duties have been described earlier in this Chapter, and include ALARA attitude promotion, radiation surveys, personnel dosimetry, environmental sampling, training of personnel, maintaining radiation zones, survey meter calibration, rad waste handling and refueling operations. Since the size of a research reactor is smaller in terms of personnel, budgets and physical facilities, the research reactor health physics technician is involved in a much wider variety of tasks compared to their power reactor counterpart. The ability to get along with others is especially important since the experimenters are frequently senior scientific personnel much more interested in getting their work done than in radiation safety concerns. Where the power reactor will have separate organizations, each with several people, to handle tasks such as environmental monitoring, meter calibration or personnel dosimetry, the research reactor may only have a total of one or two HP technicians. Thus, each technician has to have the skills to perform many tasks at the research facility.

Other Resources

1. "Information Relevant to Insuring that Occupational Radiation Exposures at Nuclear Power Stations will be ALARA," USNRC Regulatory Guide 8.8, Washington, D.C.

2. "Radiochemistry in Nuclear Power Reactors," Chien C. Lin; Nuclear Science Series NAS-NS- 3119; National Academy Press; Washington, D.C. 1996.

3. "Occupational Exposures at Nuclear Power Plants," Eighth Annual Report; OECD Nuclear Energy Agency; International Atomic Energy Agency, Paris, 1998.

4. "Work Management in the Nuclear Power Industry," OECD Nuclear Energy Agency, Paris, 1997.

5. "Radiation Protection at Nuclear Reactors," Constantine J. Maletskos, editor, Medical Physics Publishing, Madison WI, 1995.

6. "Vision 2020," published by Nuclear Energy Institute in 2004 and available on the NEI web site, www.nei.org.

7. "Biological Effects and Exposure Limits for 'Hot Particles'," NCRP Report 130, 1999.

8. "Radiological Control at Research Reactor Facilities," Amer. National Std., ANSI 15.11, New York, NY.

9. “Information Digest,” NUREG-1350. The U.S. Nuclear Regulatory Commission, Washington, DC, issues updated volumes annually. The latest volume can be downloaded at www.nrc.gov/reading-rm/doc-collections/nuregs and enter “1350” in the Search field.

10. The “scorecard” for radiation safety performance for each licensed power reactor in the U.S. is updated quarterly and listed, IN COLOR, for a whole variety of different Performance Indicators at the following website maintained by the US Nuclear Regulatory Commission: www.nrc.gov/NRR/OVERSIGHT/ASSESS/.

S-2

Working Safely with Radioisotopes

Outline of this Chapter

Chapter Summary

The chemical and physical properties, metabolism and radiobiological behavior, methods of hazard measurement, toxicity and current protection standards are explored for three common, representative radioisotopes.

In the case of uranium, the chemical valence state determines whether compounds are soluble or insoluble. These two forms of uranium are treated differently when taken into the body. The route of entry – either through the nose or through the mouth – determines the ultimate fate of the deposited material. More than 90% of the insoluble forms of uranium taken in through both entry routes is cleared through the bowels by fecal elimination. For the soluble form, about a fourth of it is cleared through the urine.

The amount of activity of uranium deposited in the body can be measured in two ways: 1) the lung counter, placing radiation detectors directly over the lungs, and 2) the urine sample, where the content is determined by chemical analysis. For natural and depleted uranium and uranium with a small percentage of enrichment, the chemical hazard far outweighs the radiological hazard. A number of federal and state agencies have established standards for the working environment to protect workers.

I-125 is a short half-life, low energy electron and x-ray emitter often used in medical applications. Under ICRP models, 30% of iodine in blood is taken up by the thyroid gland. I-125 clears the thyroid with an effective half-life of 40 days. The thyroid burden is usually measured with a thin NaI(Tl) scintillation counter. The process involves making a correction for absorption of emitted radiations by the tissues overlying the gland. Under U.S. NRC guidelines, some form of containment is required if workers handle more than 1 millicurie of volatile I-125 during a three-month period.

Krypton-85 is a radioactive noble gas which does not occur naturally. The half-life is about 11 years. The decay process produces two betas and a gamma ray. Most of the hazard is due to the 687 keV beta. Even filtered by room air, it carries sufficient energy to penetrate a worker's skin. As a noble gas, the element is not readily taken up internally, and lung dose from inhaled Kr-85 is much smaller than the corresponding skin dose.

Kr-85 is somewhat tricky to properly measure. Clearly, gas dispersed throughout room air does not constitute the desirable "point source" situation. The low to medium energy beta rays do not easily penetrate many survey meters, so special calibrations must be performed. Similarly, conventional film and TLD personnel badges are usually unable to record the dose due to krypton-85 exposures.

Introduction

This supplemental chapter contains information useful for handling three representative radioisotopes – Uranium-238/235, Iodine-125 and Krypton-85. Each of these nuclides has commercial and research applications. Much of the information will also apply to radioisotopes which are radiologically similar to those listed. The information presented for each is organized under the following headings:

- Chemical and Physical Properties
- Radiobiology and Isotope Metabolism
- Radiation Protection Measurements
- Toxicity and Protection Standards

Working Safely with Uranium

Chemical And Physical Properties

In the chemically pure state, uranium appears as a shiny, silvery metal. Actually, the atoms which make up uranium metal consist of three different chemical isotopes, having the atomic mass numbers 238, 235 and 234. More than 99% of the uranium that occurs in nature is the isotope U-238. A small fraction of uranium atoms are of the 234 isotope and the 235 isotope (which fissions and is used in the nuclear fuel cycle). Uranium which has had the 235 isotope removed is referred to as "depleted uranium." It is the form commonly encountered when uranium is used industrially in non-nuclear reactor applications. On the other hand, uranium in which the 235 isotope concentration has been artificially increased above the natural 0.7% level is called "enriched uranium."

In the chemical compounds which are formed by uranium, the valence state has been observed to range from plus 2 to plus 6. However, by far the most commonly occurring valence states are the plus 4 and plus 6 compounds.

The density of uranium, the mass per unit volume, is relatively high at 18.7 g/cm^3. In fact, uranium is 65% more dense than lead, which is normally thought of as being a very dense metal. In the entire periodic table, only eight metals have a density greater than uranium.

Uranium is fairly widely distributed around the world. For example, the crust of the earth contains a concentration on the average of about 4 parts per million. In order to be considered usable, or economically viable as uranium ore, the concentration has to be in the vicinity of 30 ppm or higher.

Common foods also contain uranium to some limited extent. Foods have an average concentration of uranium somewhere between 0.1 and 1 ppm. In addition, drinking water contains small amounts. In the United States the average for domestic water supplies is about 2 millionths of 1 microcurie for each liter.

Uranium, along with a few other radioactive materials, is unusual in that it is part of a radioactive decay chain. Physicists talk about the uranium series. What this means is that when an atom of uranium-238 decays it converts into a different radioisotope. This next generation, or "daughter product," turns out, in the case of decay chain radionuclides, to also be radioactive. The daughter decays and causes a third nuclide to be formed. The third element also turns out to be radioactive and, in fact,

this process of successive decays from parent to daughter goes on for seventeen steps. This means that, for seventeen consecutive generations of decays, starting with uranium 238, the decay daughter products which are formed are also radioactive.

As found in nature, uranium ore actually contains uranium radioisotopes from two different radioactive decay chains. Depending on the steps taken during processing of uranium ore, the resulting material will have widely varying ratios of the three isotopes U-234, U-235 and U-238. Also, it is necessary to distinguish between the mass of a particular isotope and the activity of that isotope in a given sample. To attempt to reduce some of the confusion, the following discussion introduces some concepts and terms used to describe "uranium," and concludes with a Table (see Figure 1) that summarizes the results.

When dug out of the ground, uranium ore contains, by weight, about 0.006% U-234, 0.7% U-235 and 99.3% U-238. The uranium occurring in such material is variously called natural uranium, normal uranium or unprocessed uranium. Then the processing begins! Once natural uranium is chemically extracted from the ore, the process isolates the uranium (all three radioisotopes) from all the other chemical elements (e.g., Th, Ra, Po, etc.) found in the ore. As mentioned above, if steps are taken to remove most of the U-235, the result is called depleted uranium or DU and conversely, increasing the amount of U-235 produces enriched uranium. If the enrichment reaches 20% or more of U-235 then the material is referred to as highly enriched uranium, HEU. (HEU is used for nuclear weapons and as naval reactor fuel.)

Recall from Chapter 2, Equation 3 that there is a relationship between the activity of a source and the mass of source material, i.e.,

$$\textbf{Activity} = A = \lambda N \text{ or } A = (\ln 2 \times N)/T_{1/2}.$$

So the activity (measured in Ci for example) is proportional to the mass (N) and inversely proportional to the half-life. Finally, recall from Chapter 5 that the Specific Activity is the radioactivity per unit mass, e.g., Ci/gram. From the equation just above, it should make sense then that the Specific Activity is also inversely proportional to the half-life. Therefore, a long half-life means low activity per gram and short half-life means high activity per gram. The three uranium radioisotopes being discussed here have widely varying half-lives – of the order of 10^5, 10^8 and 10^9 years. (Note that the 10^9 year $T_{1/2}$ of U-238 is comparable to the age of the universe!) Thus, the specific activities are going to be widely different and that means the weight % of uranium radioisotopes will be markedly different than the activity %.

Finally, due to the physics of the uranium decay series, depleted uranium contains only four isotopes – U-238 and the first three daughters of the seventeen that were just mentioned. Uranium 238 will decay into an isotope of thorium, Th-234, which decays into protactinium- 234. It is again radioactive and decays into uranium-234. Because of its long half-life, U-234 will not decay to produce significant daughter radioactivity for several thousand years. So depleted uranium will contain only these four individual isotopes.

The table in Figure 1 attempts to put all of this information in perspective. At the top of the table, isotopic properties are listed. The lower part of the table lists the bulk properties. Note that while natural U is about 50-50 U-238 and U-234 by activity, it is over 99% uranium-238 by weight.

In the process of uranium-238 decay through the first four generations, there are three different types of radiation given off. First, there are a total of seven gamma

Uranium Isotopic Properties

Radioisotope	U-238	U-235	U-234
Atomic Mass Number	238	235	234
Half-life in years	4.47E+09	7.04E+08	2.34E+05
Specific Activity microCi/gm	0.3362	2.162	6218

Uranium Bulk Properties

Type of Uranium	Natural Uranium	Depleted Uranium	Highly Enriched U
% U-238 by weight	99.274	99.75	79.994
% U-235 by weight	0.72	0.25	20
% U-234 by weight	0.0057	0.0005	0.006
% U-238 by activity	48.43	91.7	25.0
% U-235 by activity	2.26	1.48	40.3
% U-234 by activity	51.68	8.54	34.7
grams U-238 per gram total U	0.9927	0.9975	0.7999
µCi U-238 per gram total U	0.334	0.3356	0.2690
grams U-235 per gram total U	0.0072	0.0025	0.20
µCi U-235 per gram total U	0.0156	0.0054	0.4324
grams U-234 per gram total U	0.0001	0.000005	0.00006
µCi U-234 per gram total U	0.3565	0.0313	0.3731
Total U µCi per gram	0.6897	0.3660	1.074
Total U dpm per milligram	1531	812.5	2385

Fig. 1 - Properties of the element uranium

rays, all of which are either very low in energy, or occur only rarely in terms of the overall number of disintegrations taking place. There are two major alpha rays that are given off with about 4 MeV of energy each. And finally there are five different beta particles given off. The lowest has an energy of about a tenth of an MeV. The highest, dominant beta carries an E_{max} energy of 2.3 MeV.

Because of the long half-life of U-238, the specific activity is unusually low. One gram of pure uranium metal has a radioactivity of only 0.33 microcuries! Another way of looking at this figure is to see that it takes over 3 tons of uranium to equal one curie of activity.

From the point of view of radiation protection, the beta component of the radiation presents the most significant hazard to personnel. The highest energy (2.3 MeV) beta is able to travel about 25 feet in air. A person could be working on the far side of a large room and still receive beta exposure to their skin from uranium on the near side of the room. This high energy beta particle, when it interacts with the human body, is capable of penetrating to a depth of about one-half an inch in soft tissue. This means that, for purposes of radiation protection, the eye becomes a critical organ. With adequate eye protection, the next major organ at risk is the skin. In 1977, the International Commission on Radiological Protection (ICRP) published a report which documents the depths at which the radiosensitive growing layer of cells lies in humans. In the case of the eye, the most radiosensitive tissue is the lens, located inside the eyeball under a tissue called the cornea. The ICRP's recommended average human corneal thickness, for radiation protection purposes, is 3 mm. In the case of skin, the sensitive layer lies at an average depth of 0.07 mm. Thus, in many practical cases, both the eye and the skin of a worker are at risk when handling uranium.

For example, if you were to place your hand on a piece of a depleted uranium metal slab, or if you were to grasp a depleted uranium metal rod, your skin would receive a dose rate of about 230 millirem/hour while in direct contact. Thus, if you left your hand on the piece of uranium for 1 hour, your skin would receive about 230 millirem of radiation dose. If work only involves very small pieces of uranium metal, the inverse square law for point radiation sources applies and the dose rate will drop off fairly rapidly. However, in the real workplace, it is common to have multiple sources, such as a rack full of rod stock. In the case of physically large masses of metal, the dose rate falls off much more slowly with increasing distance, so care must be taken when estimating dose rates at long distances from such sources.

Radiobiology and Isotope Metabolism

The average person has a body content of uranium of about 90 micrograms – much, much smaller than the head of a pin. The average human intake has been measured at about 2 micrograms per day, and is due primarily to food ingestion. The radioactivity contained in that food is only a millionth of a μCi – a very, very small amount.

In general, to determine the hazard resulting from uranium which becomes metabolized by the body, we have to determine the relative isotopic composition of the material. In the case of natural unenriched uranium, or pure U-238, i.e., depleted uranium, we find it chiefly presents a chemical hazard to the body. This chemical toxicity far outweighs the hazard from the radiation being emitted by the deposited uranium. Only when the amount of U-235 exceeds 8%, which is more than 10 times its natural isotopic abundance, does the uranium pose a more significant radiological hazard than the hazard associated with its chemical toxicity.

As indicated above, there are two primary valence states, the +4 and the +6. Uranium in the +6 valence state tends to form soluble compounds which will dissolve in body fluids. Therefore, uranium in this state can get taken up into the bloodstream and circulated throughout many body organs. On the other hand, when uranium is in

the +4 state, the compounds formed are usually insoluble - i.e., they do not dissolve in body fluids. When taken into the body, the compounds initially stay bound together. However, once deposited in some body tissue, they will slowly convert from the +4 to the +6 state. The uranium is then soluble and can spread throughout the body. Uranium isotopes in soluble form simulate, or behave chemically like, calcium ions. Physiologically, the biggest task for the body is the extraction of the uranium from the bloodstream in the kidneys. Uranium that does make it into the blood is cleared rather rapidly through kidney action. In fact, in the +6 soluble state, about 60% of the uranium is cleared in a 24-hour period, though only about 20% of the insoluble +4 valence state is cleared by way of the urinary excretion pathway over a 1 day period.

Consider this clearance of deposited uranium from the body in more detail. The clearance half-time refers to the amount of time it takes the body to excrete half of the uranium which has been deposited in some preceding event. The clearance half-times have been measured for a number of body organs. In bone, it takes about 20 days for half the uranium deposited there to be removed by normal body metabolism. In the kidneys, the process is more rapid, taking on the average only 6 days. In the lungs, the clearance half-time varies dramatically, depending on whether the uranium is in soluble or insoluble form. Where the uranium is soluble, it is removed very rapidly, with a clearance half-time of only half a day. Where it is in insoluble form, it takes about 500 days – longer than a year – for half of the deposited uranium to clear out of the lung tissue.

The overall behavior of uranium which is taken into the body depends primarily upon the route of entrance into the body. For instance, uranium which is inhaled through the nose behaves quite differently from that which is ingested through the mouth. In 1979 the International Commission on Radiological Protection published a comprehensive report describing the fate of inhaled and ingested uranium. Figure 2 is a summary of the results of this study. Listed are the percentages of uranium which clear through excretion in the urine. There are four combinations of variables: the results depend on whether the entrance route is by ingestion – through eating the material, or by inhalation – by breathing the material, and on whether the uranium is in soluble or insoluble form in body fluids.

Consider first the two cases of ingestion. In soluble form, 3% of the uranium is excreted in the urine vs. only a tenth of one percent if the uranium is in insoluble form. For the cases of inhalation, 24% of uranium in soluble form will clear through the urine, vs. only 1% if it is insoluble. The remainder of the deposited uranium eventually clears through the gastrointestinal tract. More than 90% of the uranium which enters the body in three of these four categories is eliminated through fecal excretion.

Intake by:	Soluble	Insoluble
Mouth	3%	0.1%
Nose	24%	1%

Fig. 2 - Urinary clearance of uranium

The exception is the 57% clearance via the feces for soluble uranium which is inhaled.

It is probably useful to make a comment here about breathing by mouth. Some workers have this habit in the workplace. This practice greatly increases the likelihood of radioactive contamination entering the body. The hairs that line the nasal passage are extremely effective in removing small particles of uranium in the form of dust and suspended particles in the air. The information just discussed about deposition of inhaled uranium assumes intake through the nose, not the mouth. Workers should be encouraged to try to avoid breathing through their mouths in the workplace so that they can take advantage of their body's own effective mechanism for preventing internal deposition of particulates.

Radiation Protection Measurements

There are two major practical methods that radiation safety personnel can use to assess the amount of internally deposited uranium in workers. (This internal radioactivity is called the "body burden" when total body content is meant. The term "lung burden" would refer to the amount of activity deposited in a worker's lungs.) The measurement methods for these are: 1) use of a lung counter, an electronic detector which measures the amount of radiation coming out of the lung from internally deposited radioactivity, and 2) measurement of uranium concentration in the urine and subsequent calculation of the corresponding amount of activity that is deposited in the body. A number of lung counters are commercially available. The photographs in Chapter 9 illustrate one example of a commercial service which provides mobile lung counting facilities.

The second practical measurement technique available – urinalysis – involves the collection, on a periodic basis, of urine samples from the worker. The urine is sent to a laboratory where the concentration of uranium in the urine is measured by chemical analysis. As indicated previously, some fraction of all uranium that enters the body will eventually end up in the urine, and so routine analysis of urine samples is a convenient way to keep track, on a month-to-month basis, of the amount of uranium intake by workers. It is particularly useful as a simple technique that can be done between lung counts, which are only occasionally taken. This allows the radiation safety officer to keep track of occupational exposures and, if necessary, to change work habits or job locations temporarily to assure that workers stay well within the legal dose limits which have been established for their safety.

Toxicity and Standards

The primary problem in terms of the health effects of depleted or natural uranium intake results from the chemical toxicity. The chief biological effect of overexposure is nephritis – inflammation or infection of the kidney. Anatomically, the nephron is a tiny structure inside the kidney itself. Blood passes through a filter-like structure and the foreign material, in this case the uranium, is extracted into a compartment called the glomerulus. From there it is transferred to a connecting tubule, which eventually leads the collected fluid to the bladder. A typical kidney contains about 1

million of these tiny units, the nephrons, where this filtration takes place. Nephritis begins to take place in humans and animals after a single acute intake of about 7 milligrams of uranium.

For intakes, either by breathing or by mouth, of less than 7 mg., there has been no damage observed in humans or animals. Based on results of a Russian study, the National Academy of Sciences in the U.S.A. has established a working level for urine concentrations of 210 micrograms of uranium per liter following a 7-day exposure. They consider there to be no significant adverse effects on humans at lower concentrations. The allowed level that the National Academy has established for a single short-term exposure is about 3500 micrograms/liter of urine. A U.S. Army study conducted years ago showed damage beginning at about 2500 micrograms/liter in the urine. At the present time there still have been no reported studies on humans demonstrating effects on the kidney of long-term, chronic, year-after-year exposure to uranium. In addition to nephritis, other observed bioeffects of uranium exposure include lung disease and interference with normal bone formation.

It would be nice if there were some magic potion that could offer protection to a person with a significant uranium uptake. Unfortunately, to date, nothing has been approved for use. A number of chelating agents, including EDTA and DTPA, have been examined over the years for a protective effect. A 1994 study reports 100% survival in rats exposed to toxic levels of uranium when protected by a single dose of EHBP (ethane hydroxy biphosphonate). This agent is used in human treatment of Paget's bone disease with no known health risks.

There are a number of standards which have been established in the uranium industry. Some of these have been formalized, while others are more like rules-of-thumb. The National Council on Radiation Protection and Measurements has established maximum permissible amounts of uranium in various organs of the body. For natural uranium, they recommend 0.005 microcuries in the kidneys as the maximum permissible level. In bone, they recommend 0.03 microcuries, and in the total body, 0.2 microcurie is the maximum allowed activity. By inference, the maximum permissible lung burden becomes 0.02 microcuries, which would be equivalent to about 50 milligrams in the lungs.

In 1981, the Environmental Protection Agency in the United States proposed a standard for drinking water of ten picocuries -10^{-5} microcuries – of uranium per liter of water. At that time more than 97% of all community water supplies in this country would have met the standard. 1 or 2% of community water supplies in areas with high uranium concentration in the ground would require water treatment to meet the 1981 proposed EPA standard. In July, 1991, in response to pressure from the U.S. Congress, EPA reissued a new proposed set of standards for radionuclides in drinking water. That version recommended a Maximum Contaminant Level of 20 μgm per liter. It was expected that a final rule would be issued on drinking water in 1993 or 1994. Finally the standard was issued with an effective date of December 2003. It ended up setting the uranium MCL at 30 μgm per liter of drinking water, i.e., about 20 pCi/liter.

The International Commission on Radiological Protection has also established some standards for working with uranium. Their standard is based on the air concentration, i.e., how much mass of uranium is allowed in a unit volume of air. They established a working level of 1.6 milligrams of uranium/cubic meter of air for long-term exposure to uranium in insoluble form.

The NRC regulations, as contained in 10 CFR 20, established an ingestion annual limit on intake, ALI, of 30 grams of natural or depleted uranium. This ridiculous figure was arrived at without consideration of the chemical toxicity. As indicated above, radiologically, uranium is not particularly hazardous. However, in recognition of the special case of uranium, 10 CFR 20.1201(e) imposes a 10 mg per week limit on soluble uranium intake, "in consideration of chemical toxicity." This would correspond to an ALI of only 0.5 grams of uranium per year. The Derived Air Concentration, DAC, for natural or depleted uranium under the 10 CFR 20 regulations is similar to the ICRP value – 1.8 mg/m^3.

Finally, the ACGIH, the American Conference of Government and Industrial Hygienists has made a pronouncement on uranium. They propose a long-term average limit for air that is breathed in the workplace of two tenths of a milligram per cubic meter, with a short-term maximum allowed of 0.6 mg/m^3.

Workers are often interested in the radiation doses which result from taking into the body certain amounts of uranium. The ICRP model that we referred to earlier made a detailed calculation of the doses to various body organs as a result of intake of uranium in the two forms, soluble and insoluble. The results of that study are summarized here as Figure 3.

Organ	Organ Dose (mrem per milligram U)	
	Soluble	Insoluble
Lung	0.3	300
Bone	1	0.04
Kidney	0.5	0.02

Fig. 3 - Internal doses from uranium-238

The three major organs which receive most of the dose are shown. For each organ the doses are expressed as the number of millirem of radiation dose received for each milligram of deposited U-238. The two forms, soluble and insoluble, are indicated for each organ. For example, lungs containing soluble uranium would receive only three-tenths of a millirem for each milligram deposited, whereas, if it were in insoluble form, they would receive 300 millirem/mg. The reason for the large difference, a factor of one thousand, between these, is that uranium in insoluble form takes more than a year to clear, whereas the soluble form clears in only a few days. Note that the radiation dose to the kidneys is actually quite small, emphasizing once again that the chemical toxicity of uranium is the overriding risk to the kidney.

Working Safely with Iodine-125

Introduction

Several iodine isotopes are frequently used in diagnostic (and occasionally therapeutic) medical procedures and in many biological research applications. These primarily include the ^{131}I, ^{123}I and ^{125}I radioisotopes. While the specifics of the decay

energies and particles differ, along with the half-life, the metabolic behavior of the mentioned isotopes is identical, and handling procedures are similar. The remainder of this section is devoted exclusively to the I-125 isotope.

Chemical and Physical Properties

The decay energies and radiations for ^{125}I are summarized in the table of Figure 4. I-125 decays 100% of the time by a process called electron capture. In this type of radioactive decay, the nucleus captures an orbital electron, almost always from the K shell. The conversion electrons mentioned in Figure 4 refer to orbital atomic electrons that are sometimes ejected from the atom in place of the gamma ray that would normally be emitted. This process of internal conversion is often thought of as a type of "internal photoelectric effect" in which the gamma ray is emitted by the nucleus but hits an orbital electron on its way out of the atom. The electron is ejected and is then called a conversion electron to distinguish it from beta particles which might also be emitted. The chief difference between a conversion electron and a beta particle is the fact that internal conversion electrons are monoenergetic, whereas beta particles exhibit a spectrum of energies as discussed in Chapter 2.

The Auger (pronounced "oh-zhéy") electrons also listed are very similar to the conversion electrons. Instead of a characteristic x-ray that would normally be produced when an inner shell electron is captured by the nucleus, the atom ejects one of its inner shell orbital electrons. Thus, the Auger Effect (discovered by the physicist Auger in 1925) is analogous to an "x-ray photoelectric effect." As in the case of internal conversion, the Auger electrons are monoenergetic.

The physical half-life of ^{125}I is 60.14 days. This is long enough to be useful yet short enough that contaminated items can sometimes be held for decay below levels of regulatory concern.

Using the 1979 ICRP internal dosimetry model discussed in Chapter 9, the dose to a person's thyroid gland can be easily calculated for a unit uptake of I-125. The calculation gives a result of 0.0126 Sv to the thyroid per μCi of oral intake. Using the revised 10 CFR 20 models, the ingestion Annual Limit on Intake, ALI, is 40 μCi of

RADIATION	ENERGY(KEV)	DECAY%
Gamma	35.5	6.7
K_α X-ray	27.4	114
K_β X-ray	31	25.6
L X-ray	3.9	12
K Conversion Electron	3.7	80
L Conversion Electron	31	11.8
M_+ Conversion Elect.	35	2.5
K Auger Electrons	23	20
L Auger Electrons	3-4	160

Fig. 4 - The major radiations from I-125

I-125 per year. The inhalation ALI value is 60 μCi per year. The specific exposure rate constant, Γ of Chapter 5, has the value 14 mrem/hr at 10 cm per millicurie of ^{125}I. As shown in Chapter 5, this number can be used to calculate dose rates at other distances and for other source activities using the inverse square law.

Chemically, the diatomic form, I_2, is very volatile (evaporates readily at room temperature). Elemental iodine (I) readily converts to the I_2 form. If the iodine is in solution, the iodide ion (I^-) can be easily oxidized to I_2 gas through a reaction with oxygen in the atmosphere. This reaction is catalyzed by light so it is usual to store iodine solutions in brown colored bottles. The oxidation reaction is inhibited if the solution is kept basic (pH>7.8). Contrary to popular belief, storing iodine solutions in a freezer makes matters worse. Freezing leads to volatilization.

Radiobiology and Isotope Metabolism

For radiation protection purposes, the ICRP uses a thyroid gland mass of 20 grams in adults. The fractional uptake of iodine by the thyroid gland of iodine circulating in blood is assumed to be 30%. Iodine which is ingested is assumed to be absorbed rapidly through the wall of the intestinal tract whereupon virtually 100% ends up in the bloodstream. If the iodine is inhaled rather than ingested, 100% still transfers to blood from the lung. This process has a clearance half-time of about 12 hours.

Some workers are not aware of the rapid percutaneous absorption of iodine compounds. The material is readily and rapidly absorbed into the bloodstream following surface deposition on intact human skin. For this reason, it is common practice to require double gloves when handling single quantities of ^{125}I in excess of 1 mCi. As a result of the ability of iodine to permeate glove materials, the outer pair of gloves should be changed at 10 minute intervals.

The 30% fraction of an iodine uptake that binds to the thyroid gland will clear from the gland with a biological half-life of 120 days according to ICRP models. This produces an effective half-life (see Chapter 9) of 40 days for ^{125}I in the thyroid. The iodine released by the gland is in the organic form. The 70% fraction of iodine left circulating in the blood clears through kidney action with a biological half-life of 12 days (T_{eff} = 10 days).

Radiation Protection Measurements

Over the years, many attempts have been made to measure iodine body or thyroid burden by urinalysis. In principle, the method should work. In practice, results are usually subject to large uncertainties having to do with sample collection time and volume. Thus, the only reliable way to determine thyroid burdens is by use of an *in vivo* counter. The detector of choice is a 1 mm thick NaI(Tl) scintillation counter. A photo of such an instrument in use is shown in Chapter 9. The 1 mm thick crystal is thick enough to have almost 100% detection efficiency for the low energy iodine gamma and x-rays that interact with the detector. On the other hand, it is so thin that higher energy photons in the background spectrum are mostly rejected. (The detection efficiency is very low due to the short travel distance of the photons in the thin crystal.)

As mentioned in Chapter 9, the chief problem in thyroid burden measurements is the necessary correction for gland depth. In adults, the thyroid gland is located anywhere from 1 to 4 cm beneath the surface of the neck. Since the photon emissions of I-125 are so low in energy, between 4 keV and 36 keV, substantial attenuation of the emissions occurs and must be corrected for. To do this, the variability in gland depth from person to person must be taken into account. This is usually done by use of a "two count method" in which the x-ray detector is positioned at two different locations for measurements. For the first measurement, the usual system employs counts taken with the detector directly over the thyroid gland, in contact with the neck surface. The second count is then taken with the detector repositioned to the side of the neck, 90° with respect to the first reading. The ratio of counts for the two readings is used to effectively determine gland depth and the correction is then applied to the 0° reading to obtain the thyroid burden. The correction factors needed to perform this calculation depend on the specific detector and procedure used. Thus, the necessary factors for a given setup are determined experimentally using an iodine source in a neck phantom with provisions for adjusting the "gland" depth over the normal anatomical range of 1 to 4 cm. See Chapter 9 for an example of these calculations.

Toxicity and Standards

The radiotoxicity of I-125 greatly exceeds any chemical toxicity. The critical organ at risk is the thyroid gland. As previously discussed, for radiation protection purposes it is assumed that 30% of an iodine intake binds to the gland while 70% continues to circulate in the blood until filtered out by kidney action. At first glance, it might seem that more consideration should be given to the larger fraction, the 70%. There are two reasons why this is not so. Remember that radiation dose is the quotient of energy deposited and deposition mass. The 30% fraction is deposited in only 20 grams of tissue while the remaining 70% is spread throughout 70,000 grams in Reference Person. In addition, the 30% fraction has an effective half-life of 40 days in the thyroid versus only 10 days for the 70% fraction in body fluids.

The Nuclear Regulatory Commission has established guidelines for bioassay programs for workers handling ^{125}I. The reference document is NRC Regulatory Guide 8.20, **Applications of Bioassay for I-125 and I-131**. Routine bioassays are required if a worker handles in excess of the Figure 5 quantities of I-125 in volatile form during a three-month period:

An initial "baseline" thyroid count is performed and then additional counts are taken each two weeks for the first three months of work. If the average thyroid burden is less than 0.12 μCi, the bioassay frequency can be changed to quarterly. If any measured burden exceeds 0.12 μCi, corrective action should be taken and then the

Open room . . . . . 1 mCi

Fume hood . . . . . 10 mCi

Glove box . . . . . 100 mCi

Fig. 5 - I-125 limits for bioassay programs

worker is re-assayed in 2 weeks. If any burden measures above 0.5 μCi, medical consultation on a blocking agent such as potassium iodide should be undertaken at once The worker should be re-assayed weekly until the burden falls below 0.12 μCi.

Working Safely with Krypton-85

Introduction

Kr-85 is a noble gas used in a sensitive leak detection method employed by integrated circuit manufacturers and as an internal check source in survey meters. It is an artificially produced radioisotope, primarily through nuclear fission of U-235.

Chemical and Physical Properties

As one of the noble gases, krypton is colorless, generally chemically inert and without taste. The earliest report of a chemical reaction involving the element was published in 1963. A reaction producing KrF_4 was described. In addition, krypton can form compounds called clathrates involving certain organic chemicals. Other than these very unusual reactions, krypton is rather inert.

Krypton gas is quite insoluble in water. However, it is slightly soluble in some waxes, oils and fat. Thus, when inhaled, krypton shows low solubility in blood and soft tissues but relatively high solubility in lipids (natural fatty acids).

Krypton is a naturally occurring element with the following stable isotopes and percent abundances: 78(0.4%), 80(2.3%), 82(11.6%), 83(11.6%), 84(56.9%) and 86(17.4%). The radioisotopes have mass numbers ranging from a low of 72 to a high of 97. The naturally occurring radioisotopes are present only in insignificant trace amounts, having been formed as a result of cosmic ray reactions or uranium spontaneous fissioning. Krypton-85 gas which is used industrially has been produced as a fission product in a nuclear reactor. On the average, about 300 curies of Kr-85 are produced each year in a 1,000 MWe power reactor. Subsequent fuel reprocessing (described in Lesson 13) releases the Kr-85 gas.

Krypton has the following physical constants:

Atomic Number = 36
Chemical Atomic Weight = 83.80
Melting Point = -157 degrees C.
Density, at STP = 3.7 grams/liter

Krypton-85 is a radioactive gas. It has a physical half-life of 10.72 ± 0.01 years. The only radiations emitted in the decay are simply two beta particles and a gamma ray. The energies of these radiations and the percentages of the decays which result in them are shown in Figure 6.

A number of different investigators have measured the fission yield of Kr-85, i.e., the fraction of the fissioning atoms which produces a Kr-85 nucleus. For uranium-235 being bombarded with thermal neutrons, the values measured range from 0.27% to 0.34%.

The beta spectrum for a Kr-85 source in a vacuum (i.e., not being attenuated

DECAY OF KRYPTON-85		
Beta #1	0.44%	173 keV
Beta #2	99.56%	687 keV
Gamma	0.44%	514 keV

Fig. 6 - Radiations emitted in the decay of krypton-85

by interactions with the room air) is shown in Figure 7. The presence of room air actually makes the energy fall off much more rapidly than illustrated. Collisions with the air quickly remove the higher energies from the spectrum. Based on published depth dose curves for Kr-85 betas in water, it can be shown that the air attenuated beta spectrum from Kr-85 has an effective (or average) energy of only 310 keV.

The 687 keV beta particle emitted in almost all (99.6%) of the disintegrations of Kr-85 has a range, measured in density thickness units (see Chapter 3), of 240 mg/cm^2. This means that the maximum beta emitted could travel 185 cm or about 6 feet in air. In water or human soft tissue, it would travel 2.4 mm or about 1/10 inch. The thickness of a shield for these beta rays is found by dividing the range by the shield material density. To cite an example, acrylic plastic (Plexiglas or Lucite) has a density of 1.180 mg/cm^3. Thus, the thickness needed to stop Kr-85 betas is 240/1180 (cm) = 0.2 cm or about 3/32 of an inch.

Since few beta particles are ever emitted with the maximum energy, it may be more realistic to consider the average beta energy, 251 keV. This beta particle has a range of 60 mg/cm^2. It would travel 46 cm or about 18 inches in air and 0.6 mm in water or soft tissue.

The Half-Value Layer, HVL, for the 514 keV Kr-85 gamma rays is about 0.4 cm in lead. Thus, a factor of 10 reduction in the gamma ray dose rate from a gas cylinder of krypton-85 would be obtained with about 3 1/3 HVL or 1.3 cm of lead (a bit over

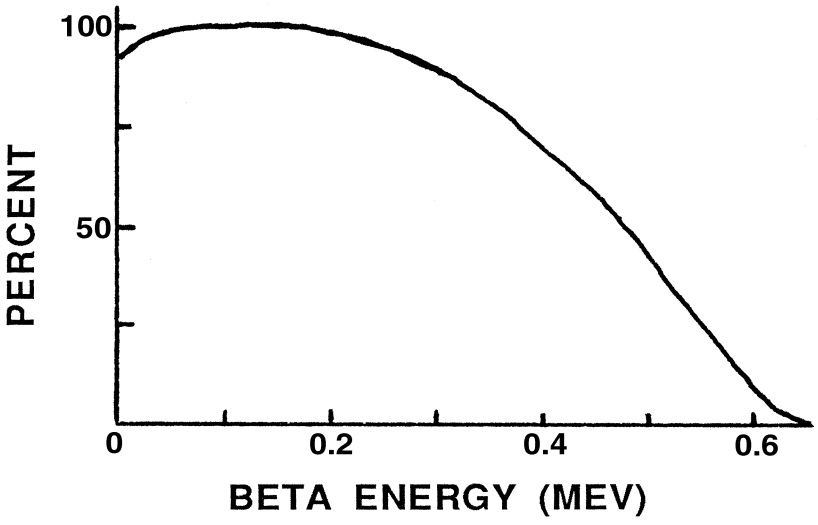


Fig. 7 - Kr-85 beta spectrum in a vacuum

half an inch). Note that the beta ray dose rate from a cylinder of the gas is zero since the cylinder wall is thick enough to stop every last beta particle emitted.

Radiobiology and Isotope Metabolism

The beta component dominates in almost every conceivable exposure situation. This means that there are potentially two organs at risk – the skin and the lenses of the eyes. The International Commission on Radiological Protection (ICRP) has established values for skin and eye tissue sensitive depth for radiation protection purposes. According to their Report 26, "In adults, the equatorial portion of the anterior epithelium of the lens is the anatomical region generally considered to be the part of the lens most susceptible to the induction of lens opacities (cataracts). For the purposes of radiation protection the equator of the lens can be considered to lie 3 mm behind the surface of the eye." Similarly, they specify a depth of 0.07 mm "as a reasonable mean value for practical dose assessment" for skin cells at risk.

As just specified in the physical data above, the maximum krypton-85 beta range in soft tissue is 2.4 mm and the average beta range is 0.6 mm. Thus, it should be clear that NONE of the Kr-85 betas can ever reach the lens of the eye since it is protected by a cornea thicker than the maximum beta range. On the other hand, the sensitive growing layers of the skin are at risk both for the maximum range beta as well as the average range beta (and even those carrying significantly less energy than the average).

The most definitive study of human doses from exposure to Kr-85 gas was published in 1973 by Snyder, Dillman, Ford and Poston of Oak Ridge National Laboratory. They performed complex computer calculations to determine the doses to the various tissues of a person standing in a vast uniform cloud of krypton-85 gas and concluded that there were six components to consider (Figure 8).

1) Photons and betas emitted in air
2) Bremsstrahlung in air from the betas
3) Bremsstrahlung as betas enter the skin
4) Photons and betas from gas absorbed in tissues in the body
5) Bremsstrahlung from internal betas
6) Photons and betas from gas in air passages of the lungs

Fig. 8 - The dose components due to a Kr-85 gas cloud

After carefully considering each of these sources of the dose, results were presented summarizing the total annual dose to various organs for a person immersed in a continuous concentration of one microcurie per m^3. Some of the data is shown in Figure 9. Again, it is clear that only the skin of the worker is at risk under normal working conditions involving use of Kr-85 gas. It can also be seen that the doses to internal organs for a worker exposed, even under conditions of time long enough for the gas to reach equilibrium in body tissues, are about 100 times less than the corresponding skin dose. Thus, under normal working conditions, there are no significant doses to internal organs if the workplace ventilation is sufficient to keep skin exposures within allowed limits.

ORGAN	DOSE mrem/year
Skin	1,800
Lungs	31
Marrow	18
Ovaries	6
Testes	16
"Total Body"	15

Fig. 9 - Annual doses from one μCi per cubic meter of Kr-85

Radiation Protection Measurements

Virtually all commercially available geiger counter survey meters in the USA are calibrated to read exposure rates (mR/hr) produced by high energy gamma rays. Recall from Chapter 5 that the roentgen is not even defined for beta rays!! Thus, at best, conventional geiger counters can only detect the presence of beta rays. In the standard sidewall tube, beta detection is done by opening the "beta window" through rotation of the probe cover. BUT, for the counter to record it, a beta particle must first penetrate the wall of the GM tube. With a conventional sidewall tube of 300 mg/cm^2, no betas from ^{85}Kr can penetrate the wall. With a "thin wall tube" with 30 to 40 mg/cm^2 density thickness in the wall, the beta transmission is only about 7% of the total.

To verify the preceding argument experimentally, comparative measurements of the Kr-85 beta response of several geiger counter instruments were made by Pacific Radiation Corp. The table in Figure 10 shows the results of exposure of the instruments to a "semi-infinite cloud of Kr-85."

INSTRUMENT	TYPE	READING (% Of Actual)	UNDERRESPONSE
Eberline E-120	GM, win.open	6	16 times
Eberline E-120	GM, win. closed	0.3	333 times
Eberline E-500	GM, 1B85 Open	23	4 times
Eberline E-500	GM, 1B85 Closed	0.3	333 times
Radector II	Ion Cham., Open	0.7	142 times

Fig. 10 - Response of various detectors to a Kr-85 gas cloud

As can be seen, considerable error can be made in reading the dose rate to workers exposed to krypton-85 gas. The 1B85 geiger tube mentioned is of the "thin wall" type. It only reads 23% of the true dose rate. Clearly, the answer is to reduce the wall thickness to a point where a significant fraction of the Kr-85 betas can penetrate. This can be done through use of a geiger tube with a thin mica window (about 1.5 mg/cm^2) such as a pancake GM tube.

Mere substitution of a mica window still doesn't solve all the problems, however. Commercial mica window tubes are commonly available in 1", 1 1/2" and 2"

GEIGER TUBE	WINDOW	READING(% actual)	RESPONSE
Nuc. Chicago D-35	1.0" mica	2,410	24X over
Atom. Acc. EWH 108	1.5" mica	435	4.35X over
Eberline HP-260	2.0" mica	64	36% under

Fig. 11 - Response of thin window GM detectors to a Kr-85 gas cloud

diameters. The next table, Figure 11, summarizes some additional measurements by Pacific Radiation Corp. on mica window tubes calibrated to accurately read high energy gamma rays and then exposed to the semi-infinite cloud of Kr-85. The reason for the huge over or underresponse with these tubes is the fact that the beta response depends on window area but the gamma response depends on tube volume, wall thickness and atomic number.

Commercial film badges are frequently used for beta ray dosimetry. While this practice is acceptable for high energy betas from nuclides such as depleted uranium or Sr-90/Y-90 with which the film badge is calibrated, detection of krypton-85 medium energy betas is a completely different story. The efficiency for detection by film is strongly dependent on both the angle of incidence into the badge and upon the beta energy.

The chief problem with film is the absorption of the beta rays by the paper wrapping around the film emulsion before the betas reach the sensitive layer. Published response curves show the film emulsion can detect only beta rays above 0.2 to 0.3 MeV in energy. Unfortunately, the average energies of the two Kr-85 betas are 0.04 and 0.25 MeV and the air-scattered and absorbed spectrum from Kr-85 has an effective energy of 0.31 MeV. With a typical paper wrapper thickness, about 92% of the betas are excluded.

Published angular response curves for film badges show that betas incident at 45 degrees rather than perpendicular to the badge will be underreported by 58%. In an air-scattered spectrum of Kr-85 gas, betas would be entering the badge from all directions up to 180° equally.

Some of these difficulties are recognized by commercial suppliers of film badges. Examination of the "fine print" on the report forms from two of the larger U.S. suppliers gives the following information. Supplier A reports beta doses only "for beta energies above 1 MeV." Supplier B reports only "beta: over 1.5 MeV." Based on these stated specifications, none of the beta rays from krypton-85, with maximum energies of 0.69 MeV, are being reported! (Incidentally, both of these companies routinely supply film badges for use with Kr-85 leak testing equipment.)

The performance of most commercially available TLD badges also leaves much to be desired. Under U.S. Nuclear Regulatory Commission performance standards, these badges are calibrated with a 2.27 MeV beta source held perpendicular to the badge. Angular response curves show a 49% underresponse at 45 degrees incidence in TLD badges. In addition, common TLD materials have a sensitivity that is dependent on the beta energy. Measurements by Pacific Radiation Corporation, in cooperation with the NRC personnel dosimetry testing laboratory, show that LiF TLD crystals

indicate only 28% of the correct response based on the 2.27 MeV standard. Thus, specially designed and calibrated TLD badges are needed to accurately measure skin doses to workers from Kr-85 betas.

Toxicity and Standards

The major point made in the text about placing more emphasis on control of distance than time to keep radiation doses ALARA doesn't apply to krypton-85 gas in room air because the gas is dispersed uniformly and thus does not constitute a point source. Under these conditions, worker doses can only be controlled by limiting the exposure time. Use of a room air monitor will warn of unsafe levels. Note, however, that Kr-85 gas confined in a pressurized gas cylinder does constitute a "point source" as defined in the text.

In the event of an accidental release of krypton-85 gas, the emergency procedures are relatively simple. Basically they can be summarized as

EVACUATE AND VENTILATE!

Warning of a gas leak might come in several different ways. If there is a physical rupture of tubing, gas cylinder or other component containing krypton, obviously there will be a release. Small leaks that discharge substantial amounts of Kr-85 can be warned of only through use of radiation detection instrumentation. The chief hazard in Kr-85 emergencies is to the skin of the workers. Even if the gas is directly inhaled into the lungs, the lung dose will be less than 2% of the skin dose. For a short-term exposure, virtually no uptake will occur in internal body organs. Finally, since the radioisotope is a gas, there is no possibility of radioactive contamination in the form of loose or airborne particulates that can be tracked about the facility.

Taking all of the above into account, EVACUATE, i.e., notify all persons to vacate the area where the accidental release is occurring. If it can be done in a short amount of time, make an attempt to stop the leak – tighten the pipefitting, close the cylinder hand valve, turn off the electrically operated valve, loosen the sticking valve or whatever. Then, leave the area. Turn on all possible ventilation equipment and portable fans. To prevent further exposures of personnel, entry to the area should be restricted to emergency personnel only. Rooms should be locked or other steps taken to exclude bystanders.

The next step to be taken by the operator of the equipment is to notify the appropriate emergency personnel. The particular persons to be contacted will depend on company policies but might include the radiation safety officer, an industrial hygienist or security personnel. These persons can assess the hazard and make radiation surveys to determine what further steps need to be taken.

If the release involves a large quantity of Kr-85, then federal and state radiation control regulations require notification of the appropriate licensing authority. Current telephone numbers for the NRC Emergency Operations Center is listed in the text in Chapter 14. To determine whether notification is necessary, a determination of personnel skin doses and air concentrations of Kr-85 at the release point is needed. The table in Figure 12 gives the skin dose reporting requirements. Readings from a

room air monitor that integrates the DAC hours would be particularly useful in getting a first estimate of potential personnel skin doses.

There are several different numerical standards for ^{85}Kr that are legally binding on licensees in the USA. In the case of a radioactive Kr-85 gas, the most relevant standards are the air concentration limits and the dose limits to the skin of the worker. The new 10 CFR 20 limits are set at 50 rem per year to the skin. In the case of air concentrations, the new 10 CFR 20 quantity Derived Air Concentration, DAC, replaces the MPC_{air}. It is set at 100 $\mu Ci/m^3$, a ten fold increase over the old value! The annual average allowed concentration at the boundary of a nuclear facility releasing Kr-85 gas to the environment is 0.7 $\mu Ci/m^3$ under 10 CFR 20 rules.

ESTIMATED DOSE		**NOTIFY WITHIN:**
(rem)	**(Sv)**	
>50	**>0.5**	**24 hours**
>250	**>2.5**	**Immediate!**

Fig. 12 - Kr-85 skin dose reporting requirements

S-3

The MARSSIM Decommissioning Process

Outline of this Chapter

Chapter Summary

This supplemental chapter introduces the technician to the intricacies of conducting a facility decontamination and decommissioning project using the MARSSIM process endorsed by the U.S. EPA, DOE, and DOD and required by law by the U.S. NRC. After some background on the evolution of the MARSSIM, the chapter describes the overall process of conducting surveys at a facility to release a radioactive site. The MARSSIM is simplified to a twelve-step process, each of which is described, followed by a Checklist to assure all aspects have been covered in a decontamination and decommissioning project.

Introduction to D and D

In July, 1997, The Code of Federal Regulations was amended to add a new "Subpart E" to Title 10, Part 20, entitled, **Radiological Criteria for License Termination**. The published date for full implementation was August 20, 1998. Agreement States were also required to implement these new provisions which apply to decontamination and/or decommissioning (D and D) of facilities with a State Radioactive Materials License. A license can now be released for either "unrestricted use" or "under restricted conditions." This latter category is applied in certain cases to major licensees where ALARA conditions can be met. Also, long-term legally enforceable institutional controls must be instituted at the site, and a substantial financial surety bond posted. The former category, unrestricted use, is clearly the desired choice for termination and is the objective sought in most projects. This Supplementary Chapter will cover only issues related to "unrestricted use."

Introduction to MARSSIM

The Draft NRC Regulatory Guide DG-4006 **Demonstrating Compliance with the Radiological Criteria for License Termination**, issued in July 1998, set the parameters of D and D projects. The Guide "endorses the final status survey method described in NUREG-1575, **Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM**), published December 1997. The Guide also specified values for certain MARSSIM parameters that the NRC deemed acceptable for NRC licensed sites undergoing decommissioning.

In 2003, the NRC issued NUREG-1757, **Consolidated NMSS Decommissioning Guidance**. Volume 2 of this NUREG replaced DG-4006 and is now considered the final word in interpreting MARSSIM.

> **It is worth pointing out that there are plans to publish an updated MARSSIM. Comments from the public regarding improvements were formally solicited in February 2011. Watch for a revision issued in late 2011.**

The **Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)** was developed jointly by the four Federal agencies controlling radioactive materials, The Department of Defense, The Department of Energy, The Environmental Protection Agency and the Nuclear Regulatory Commission. It provides detailed guidance for planning, implementing, and evaluating environmental and facility radiological surveys conducted to demonstrate compliance with a dose- or risk-based regulation. MARSSIM guidance focuses on the demonstration of compliance during the final status survey following scoping, characterization, and any necessary remedial actions.

The process of planning the survey, implementing the survey plan, and assessing the survey results prior to making a decision is called the Data Life Cycle. MARSSIM Chapter 2 and Appendix D provide detailed guidance on developing appropriate survey designs using the Data Quality Objectives (DQO) Process to ensure that the survey results are statistically of sufficient quality and quantity to support the final decision. The survey design process is described in MARSSIM Chapters 3, 4, and 5. Guidance on selecting appropriate measurement methods (i.e., scan surveys, direct measurements, samples) and measurement systems (i.e., detectors, instruments,

analytical methods) is provided in MARSSIM Chapters 6 and 7 and Appendix H. Data Quality Assessment (DQA) is the process of assessing the survey results, determining that the quality of the data satisfies the objectives of the survey, and interpreting the survey results as they apply to the decision being made. The DQA process is described in MARSSIM Chapter 2 and Appendix E and is applied in MARSSIM Chapter 8. Quality Assurance and Quality Control (QA/QC) procedures are developed and recorded in survey planning documents, such as a Quality Assurance Project Plan (QAPP) which is described in MARSSIM Chapter 9.

Finally, compliance demonstration is simply a decision as to whether or not a survey unit meets the release criterion. For most sites, this decision is supported by statistical tests based on the results of one or more surveys. The initial assumption used in MARSSIM is that each survey unit is contaminated above the release criterion until proven otherwise. The surveys are designed to provide the information needed to reject this initial assumption. MARSSIM recommends using the Data Life Cycle as a framework for planning, implementing, and evaluating survey results prior to making a decision.

Basically, what this says is that MARSSIM procedures call for identifying a contaminated area. Through surveys and historical research, the likely contaminant(s) is/are identified. Then meters with sufficient sensitivity are scanned with sufficiently slow speed in enough locations that we can finally say there is, for example, a 95% chance that the facility meets guideline levels.

The basic steps begin with the identification of possible radionuclide contaminants. For each contaminant, guideline values for residual activity on building surfaces and in contaminated soil are established. In MARSSIM terminology, these guideline values are "Derived Concentration Guideline Levels," or DCGLs. These numerical standards may be obtained using site specific computer modeling, through use of the generic NRC code DandD or established by negotiation with the applicable regulatory agency. (The current version of the DandD code is Version 2, a major upgrade from Version 1 in that it "allows full probabilistic treatment of dose assessments.") Usually release guidelines include surface contamination levels (in dpm/100 cm^2), external or direct radiation field levels (in μrem/hr at 1 meter), and soil or building material levels (in pCi/gram). Release guidelines invariably are expressed as values above the natural background level.

MARSSIM is a risk-based approach to decommissioning. In the "old days" the regulators used a short list of surface and soil contamination levels that were acceptable. The same numerical limits applied to thousands of radionuclides. In the risk-based approach, we decontaminate and clean down so that the residual radioactivity will not produce an unacceptable public dose rate. This rate is established in 10 CFR 20 Subpart E as follows. "A site will be considered acceptable for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a TEDE to an average member of the critical group that does not exceed 25 mrem per year." This target dose rate has been accepted in 49 of the 50 states. Litigation by an environmental coalition in California led to the State Superior Court throwing out the 25 millirem and replacing it with zero! This created a log jam in the state radioactive materials licensing organization as licensees tried vainly to terminate licenses or remove

certain rooms or buildings from their radioactive material license. As of 2011, procedures were in place to consider actions on a case-by-case basis.

Once the DCGLs are set, the physical areas of the site and its structures can be classified into one of three "classes" according to the potential for contamination. Figure 1 shows this classification for building structures, which depends heavily on a "Historical Site Assessment," HSA, and a "Scoping Survey." The HSA process is begun with a questionnaire that examines, among other things, the former and present radioactive material licensed activities at the site, NORM activities, waste disposal practices, etc. After the site is classified into contamination classes, the area is divided up into separate working units called "Survey Units." A grid reference system is set up separately in each survey unit and survey locations for scanning measurements, fixed contamination measurements and removable contamination surveys are superimposed on the grid.

Parameter	Class 1	Class 2	Class 3
Definition	Area with a reasonable possibility of residual radioactivity >DCGL, prior to remediation	Area not likely to have residual radioactivity >DCGL, prior to remediation	Area that has a low probability of containing residual radioactivity
Suggested Survey Unit Size	Approximately 100 sq m	100-1000 sq m	No limit

Fig. 1 - Area classification parameters for buildings

Note the difference between Area Classes and Survey Units. Area Classes are not limited in size. They just specify the likelihood of contamination. The Survey Units, on the other hand, are size limited. This is because the MARSSIM specifies the number of survey locations needed in the Survey Unit and if the Survey Unit is too large, there will be too high a chance that "hot spots" may slip through the survey process. Also, note that MARSSIM allows ceilings and upper walls to fall into a different Survey Unit <u>and</u> Class than floors and lower walls (up to a two-meters height).

An important part of the MARSSIM process is to assist the remediation team in choosing the proper radiation survey instruments and survey procedures. The meters need to have sufficient efficiency to be able to detect the contamination at low enough levels to assure the passing of the statistical testing at the conclusion of the survey. MARSSIM spells out detailed specifications for the survey equipment and for the speed and pattern in which the instruments are scanned. A reference grid system is standard practice in facility or site decontamination. This involves setting up a coordinate system for each contaminated room and for the remainder of the site. It allows precise locating of each measurement point. Typically, a one meter grid is used for

contaminated building floors and lower walls to 2 meters height. Contaminated exterior areas are usually gridded on a 10 meter layout.

Another confusing aspect is gridding. MARSSIM calls for the above mentioned Reference Grid to establish a location within the site. In the "old days" the interior 1 meter grid was the locator for fixed and removable survey points. Not so in MARSSIM. The Sample Grid has a spacing that is calculated based on the variability in the contamination levels. This is discussed further below.

The survey design is developed and documented using the Data Quality Objectives (DQO) Process. The DQOs for the project are established and preliminary surveys (e.g., scoping, characterization) are performed to provide information necessary to design the final status survey for compliance demonstration. The DQOs for the project are reevaluated for each of the preliminary surveys.

MARSSIM provides for an "iterative" process. We initially guess at a Class (likelihood of contamination or the lack thereof) for all areas in the project. Then, the Scoping Survey tells how good we guessed - if there is more contamination than suspected, the Class number is lowered (Remember Class 1 has highest potential contamination and Class 3 the lowest). If less contamination is found, reclassify that area to a higher number. Then, the Characterization Survey gives us more information, and again areas, or portions of them, are re-classified. Ultimately, the Final Status Survey is performed. If some areas fail, they are still again reclassified and the process goes on (hopefully not endlessly!) until each Survey Unit passes.

Finally, the requisite survey data, specific to each Class of survey unit, is obtained and statistically tested against the Data Quality Objectives. If the site doesn't pass, the data is examined for "Areas of Elevated Activity." Additional statistical testing or remediation is required to then finally release the license.

The MARSSIM Twelve-Step Process

The MARSSIM document contains 658 pages of instructions, equations, explanations and examples. NUREG-1757 Volume 2, the "instruction manual" for MARSSIM contains 524 pages. Together they were designed to answer all of the situations that might occur in the decommissioning of a huge nuclear site. But, for purposes of this Chapter, we want to condense this down to a few essentials that would be appropriate for a small decontamination project that might be directed by a radiation protection technologist, e.g., removing a small former radiation storage room from a radioactive materials license. In this case, there are twelve basic steps to the process.

Step 1 - Historical Site Assessment

Identify the property - physical address, type of facility, environmental setting.

Provide the client with a copy of the Historical Site Assessment Questionnaire (from MARSSIM Section 3.4) for completion.

Review past & current radioactive materials licenses, state and NRC to determine radioisotopes, quantity, form and dates of use of radionuclides.

Review documentation of any spills and/or accidents at the site.

Pacific Radiation has not necessarily found this step to be useful in the small MARSSIM projects we have completed. In most cases it was obvious what radioisotopes were used and where they were used or else the present occupants of the facility had no information on the previous licensee and, for reduced risk of litigation, the previous licensee refused to answer any inquiries of this nature.

Step 2 - Identify Contaminants

Based on Step 1, identify which radioactive materials, and in what ratios, are the likely contaminants on building surface structures.

Based on Step 1, identify which radioactive materials, and in what ratios, are the likely contaminants in on-site surface soils.

Step 3 - Establish DCGLs

Use the DandD computer model to derive Design Concentration Guideline Levels (DCGLs) for building surfaces and soils. Apply the "Unity Rule" in case of radionuclide mixtures.

Under the Unity Rule, the total fractions of each nuclide compared to the DCGL must be less than 100%, i.e., if the concentrations of 3 nuclides are C_1, C_2 and C_3, then $C_1/DCGL_1 + C_2/DCGL_2 + C_3/DCGL$ must be < or = to 1 (100%). If there are many radionuclides, the individual concentrations may get rather small, increasing the difficulty in detecting those particular radionuclides.

Consult with the applicable Regulatory Agency to set the Design Concentration Guideline Level for building surfaces.

Consult with the applicable Regulatory Agency to set the Design Concentration Guideline Level for contaminated soils.

The NUREG Guide suggests obtaining approval of the proposed DCGLs before beginning the remediation. If the NRC's DandD computer code is used, a copy of the computer generated report must be submitted to the regulatory agency along with a statement "that no conditions are reasonably expected to exist at the site, outside of those incorporated in the default scenarios and modeling assumptions, that would cause a significant increase in the calculated dose." In other words, pick a DandD scenario that fits the projected future use of the site you are decommissioning.

Step 4 - Classify Areas by Contamination Potential

Class 1 areas are expected to exceed the DCGL before remediation.

Class 2 areas are not likely to exceed the DCGL before remediation.

Class 3 areas have low probability of containing residual radioactivity.

Using site and building maps and floor plans, assign a classification to the impacted areas.

Select a background reference area if the identified radiocontaminants are present in background.

With the possible exception of a hot cell, it is reasonable to assume that floors are contaminated at higher levels than ceilings. Pacific Radiation usually assigns area classifications to all floor areas first. Then, the ceilings are designated one class higher (higher class number means less contamination potential). Also, remember floors include walls up to 2 meters high. Ceilings include walls above 2 meters.

Step 5 - Define Survey Units

Review "Area Classification Parameters - Buildings" chart in Figure 1 of this Supplemental Chapter to help establish the criteria for classification.

Class 1 areas are up to 100 sq m for building surfaces and up to 2000 sq m for land. Define Class 1 Survey Units in the facility.

Class 2 areas are up to 1,000 sq m for building surfaces and up to 10,000 sq m for land. Define Class 2 Survey Units in the facility.

Class 3 areas have no size limit. Define Class 3 Survey Units in the facility.

Pacific Radiation has found it convenient to post floor and ceiling maps at the job site with Survey Units drawn in and color coded to designate the MARSSIM Class.

Step 6 - Choose Survey Instrumentation and Procedures

Choose an appropriate instrument (= detector probe + electronics package) for each procedure type - scanning survey, direct measurement or wipe test.

Complete an "Instrument Qualification Form" for each individual meter to be used for surveys.

Use MARSSIM Chapter 6 as a guide and develop a "Qualification Form." Pacific Radiation's form includes data on the manufacturer, model and serial numbers of the electronics package and probe. It then addresses the MDC capabilities of the instrument (for scans, direct measurements

and/or wipe test evaluation) and compares them to the project DCGLs. Finally, it states that the instrument meets a whole list of qualifications that basically mean the meter is being used as intended by the manufacturer, within design limits (e.g., temperature & humidity) and has been calibrated and maintained properly.

Design an appropriate technique for each instrument, including distance from surface, scan speed (if applicable), acceptable background levels, etc.

Measure the Minimum Detectable Concentration, in dpm/100 square centimeters, for each instrument

$$MDC_{scan} = \frac{1512 \times \sqrt{R_B}}{\sqrt{i} \times \varepsilon \times A}$$ (For beta-gamma emitters)

$$MDC_{static} = \frac{(3 + 4.65\sqrt{B}) \times 100}{\varepsilon \times A \times t}$$ (For alpha and for beta-gamma emitters)

where R_B = probe background rate in cpm
i = interval of time, in seconds, that the moving probe is over any given point = probe size (cm) along scan direction ÷ scan speed in cm/second
ε = total efficiency of probe, in counts/disintegration
A = probe's sensitive area in square cm
B = background counts in a time t
t = counting time for static measurement, in minutes.

The above formulas will not be found in the MARSSIM! These formulas have been derived by inserting the NRC accepted values into the MARSSIM formulas to simplify them.

At the present time, neither the NRC or MARSSIM has a scanning MDC formula for alpha emitters. Alpha probes have such a low background rate that the analysis that produced the beta-gamma scan MDC equation breaks down. The reason that the ratemeter time constant does not appear in the scan MDC equation is that the detection of contamination is assumed to be by audio (in contrast to visual, [meter needle]) means. The technologist hears the click rate increase. As discussed in the Instrumentation Unit, the audio output of a survey meter is instantaneous!

The "i" in the scan MDC formula establishes the scanning speed for the technician using the probe. It can be used two ways. The surveyor can decide on a speed for moving the probe. This can then be quickly measured, and "i" can be calculated and put in the equation to find the scan MDC. If the resulting scan MDC is too high, compared to the DCGL, then an "Elevated Measurement Comparison" is done for Class 1 areas. This process is complicated and is partially described in the MARSSIM but not covered here. Alternatively, the scan MDC formula can be used to find the scan speed that corresponds to the DCGL. (Set Scan MDC = DCGL and solve for "i.") Then, scan speed = probe width divided by "i" seconds. If this speed is unacceptably slow, you may be forced to increase the number of sample locations as specified, again, by the "Elevated Measurement Comparison."

In the static MDC case, the "A" has two different interpretations. If a probe is placed at a survey location and a fixed contamination count taken, A = the sensitive probe area. However, if the counter is used to count a wipe test sample that has been wiped over a 100 sq cm area, then the "A" is 100 sq cm, the effective wipe test area.

NUREG-1757 Volume 2 specifies that the Static MDC should be less than ½ the DCGL. This should be verified and recorded on your Instrument Qualification Form for the project.

Calibrate each meter and establish criteria for response checks and twice daily Quality Assurance tests.

The frequency of response checks ("field calibrations") is dependent on the type and use of each instrument. Scanning meters are subject to a lot more wear & tear than a bench instrument for counting wipe tests. Scanning pancake GM probes fail more obviously than alpha probes in the sense that the background click rate of about 1 per second going suddenly silent is so easy to catch. In an alpha scintillator probe, a small light leak can convincingly mimic low level alpha contamination. In transuranic decon jobs, Pacific Radiation personnel check alpha probes at 30 minute intervals for light leaks. Twice daily testing uses a calibrated standard to verify efficiency, on one scale, for each meter in use that day and to verify that the background rate is still within acceptable limits.

Step 7 - Plan & Conduct Scoping Surveys

The Scoping Survey is used primarily to verify that the area classifications chosen in Step 4 are appropriate. They may be conducted as a Final Status Survey for Class 3 areas if designed as such (See Step10).

In order to save time (and the client's money) try to plan your scoping and characterization surveys so the results can become part of the Final Status Survey. If you have chosen your Class 3 Areas carefully, the initial surveys can easily become the Final surveys for those areas. Basically, jump to Step 11 to calculate the number of Direct Measurement locations needed. Then use a random number generator (laptop computer with a spreadsheet program) to choose the correct number and location of Direct Measurement points. These measurements will satisfy the Final Status Survey criteria.

Step 8 - Plan & Conduct Characterization Surveys

The Characterization Survey is used to positively identify portions of the site (building surfaces and outside soils) that are contaminated. It also provides data on the variations in the contaminant distribution which affects the number of data points needed for the Final Status Survey. A minimum of about 20 data points are needed.

Verify, by wipe testing, that the ratio of the removable to total contamination,

F_r, is < or = 10%. If F_r > 10%, then reduce a DandD derived DCGL by a factor of 10.

This is the place where MARSSIM probably differs more from past practice than anywhere else. In the "old days" wipe testing played a crucial role in D and D projects. It was not uncommon to lay out a one meter grid over the entire floor area of a building and then make one minute direct measurement counts with a probe and perform a wipe test at every grid intersection. Pacific Radiation's experience with MARSSIM surveys is that a great deal less time is spent on wipe testing than under the old rules. (Unfortunately, this is offset by more time being committed to paperwork before, during and after the survey phase.) The DCGL is expressed as TOTAL residual contamination, and is not further characterized as "fixed" or "removable." MARSSIM doesn't even discuss removable contamination and wipe testing! On the other hand, wipe testing still does have a role. The NRC's DandD computer program that a technologist will probably use to set the DCGL assumes in the modeled scenarios that removable contamination is 10% or less of the total contamination. So, the bottom line is that you take some wipe test samples during the Scoping or Characterization phase to verify that the "removable fraction" is < 10%. If this is not the case, NRC licensees can elect to assume that the removable fraction is 100% and then raise the DCGL computed by the DandD code by a factor of 10 to account for this change in the scenarios modeled.

Step 9 - Establish DQOs to Evaluate Final Survey Results

Choose values for the statistical parameters α and β (Type I and Type II decision errors). The U.S. Nuclear Regulatory Commission will accept a maximum value for α of 0.05 when the Relative Shift, defined in Step 10, is 3 or larger. Any value for β is acceptable to the Commission.

Here is where the statistical testing that underlies MARSSIM comes to the fore. A decision has to be reached on the level of "false positives" that will be allowed. This is the case where the contractor mistakenly states that the site meets the release guidelines when in fact it doesn't, i.e., a future resident might receive more than 25 mrem/year. The parameter α is the key here. The quantity (1 - α) x 100% is the confidence you can have, statistically, in the project results. In other words, if α = 0.05, (1 - α) x 100% = 95% confidence. MARSSIM includes tables covering values of α from 0.01 up to 0.25 (confidence from 99% down to 75%). As mentioned, the NRC will only entertain values of α up to 0.05. But a small α costs money! The higher the statistical confidence in the testing, the more samples have to be taken per Survey Unit, the familiar tradeoff of higher confidence for higher cost.

What about β? This parameter controls the false negatives, the case of mistakenly saying that the site is not clean enough yet to pass when in fact it is. Clearly, the regulators don't care how much extra effort you put into cleaning beyond what is needed to pass, so they allow you any choice of β.

Step 10 - Plan and Conduct Final Scanning Survey

In Class 1 areas, scan coverage should be 100%. Use the instruments and procedures selected and "MARSSIM qualified" in Step 6.

In Class 2 areas, scan coverage should be 10 to 100% for floors and lower walls (to 2 meters) and should be 10 to 50% for upper walls and ceilings. Use the instruments and procedures selected and "MARSSIM qualified" in Step 6.

In Class 3 areas, scan coverage should be judgemental. Use the instruments and procedures selected and "MARSSIM qualified" in Step 6.

Here is where the time is spent. If you were too cautious in assigning Area Classifications, you will pay now (or more properly, your employer will!). Having to physically scan 100% of floors, walls and ceilings in Class 1 Areas can be bankrupting. Particularly if you choose a scan speed such that the Scan MDC is near the DCGL. Pacific Radiation's experience is that it sometimes pays to jump to Step 11 and do the Direct Measurement surveys first. The hope is that the data may support changing the classification to a Class 2 or Class 3 Area. Then, the effort devoted to scanning is reduced dramatically. Remember, you can always revert back to the Class 1 designation if you begin finding elevated contamination levels.

Step 11 - Plan and Conduct Final Direct Measurement Survey

For Direct Measurement surveys, the number of samples and the sample locations are determined to provide the statistical confidence chosen in the DQO development in Step 9.

The number of samples, N, for each Survey Unit depends first on the Relative Shift, which is the ratio of the concentration being measured to the variability in the concentration, i.e.

$$\text{Relative Shift} = \frac{\Delta}{\sigma_s} = \frac{DCGL - LBGR}{\sigma_s}$$

where σ_S is the standard deviation of the contamination concentration in the Survey Unit
and LBGR is the "Lower Bound of the Gray Region" (see explanation below).

The "Relative Shift" is one of the major new terms in MARSSIM and maybe one of the most confusing. In a nutshell, it merely tells us how variable the residual contamination is. The standard deviation of the contamination level can easily be calculated from data in the Scoping Survey. The "Lower Bound of the Gray Region" is a statistical term that once again is intimately tied in with project costs. Practically speaking, it merely means the contamination level that you are committing to clean down to. It is the concentration limit to which the Survey Unit must be

remediated in order for the statistical testing to have a reasonable chance of passing as "releaseable to unrestricted use."

The U.S. NRC recommends that the Lower Bound of the Gray Region, LBGR, be set to half the DCGL for this calculation. If the relative shift exceeds 3, then the LBGR may be increased until the Relative Shift = 3.

The "MARSSIM professionals" at the Oak Ridge Associated Universities suggest that the LBGR be set to the average value of the residual contamination that was measured in the Characterization Surveys.

The number of sample locations per Survey Unit also depends on the statistical decision error parameters, α and β. Finally, the number N is dependent on whether or not the contamination contains a radionuclide that also occurs in the background at the site. If the contaminant is not in the background, the Sign Test is applicable and the value of N is read from Table 5.5 of MARSSIM. If the contaminant is present in background, the value of N is read from Table 5.3. In this case, the indicated number of samples must be taken in BOTH the Survey Unit and the Reference Background area.

As an example, a portion of MARSSIM Table 5.5 is included here as Figure 2. The values of N for a 5% false positive rate of incorrectly releasing the Survey Unit (SU) while still contaminated (α = 0.05) are shown for various values of the false negative decision errors (incorrectly failing to release a Survey Unit). If the site owner agrees to accept 5% false positives and false negatives, then, α and β are set to 0.05. Let us say that the average contamination found in a Characterization Survey in a SU is 250 ± 310 (1σ) dpm/100 sq cm and the DCGL is 500 dpm/100 sq cm. In this example, the relative shift is (500-250)/310 = 0.8 and number N then becomes 40 (See highlighted value in Figure 2).

Next, the sample locations must be determined. MARSSIM recommends establishing a reference coordinate system on the floor, walls and ceiling of the Survey Unit, SU. Then, the locations are determined by the Classification of the SU. For Class 1 and Class 2 SUs, the sample location spacing, on a square grid, is given by:

$$L = \sqrt{\frac{A}{N}}$$

Here A is the surface area of the SU and N the number of sample locations determined above. A random number generator is used to establish the X and Y coordinates for the starting point of the square Sample Grid.

For Class 3 SUs, N samples are taken randomly from the surface area. For each sample, generate an X and Y coordinate with a random number generator.

As discussed earlier in this Chapter, MARSSIM uses two different grids for buildings and land areas. They suggest a reference 1 meter grid (10 m for land) and then a sample grid with the spacing L calculated above.

$\alpha = 0.05$

	β =				
Δ/σ	0.01	0.025	0.05	0.1	0.25
0.1	2984	2459	2048	1620	1018
0.2	754	622	518	410	258
0.3	341	281	234	185	117
0.4	197	162	136	107	68
0.5	130	107	89	71	45
0.6	94	77	65	52	33
0.7	72	59	50	40	26
0.8	58	48	40	32	21
0.9	48	40	34	27	17
1.0	41	34	29	23	15
1.1	36	30	26	21	14
1.2	33	27	23	18	12
1.3	30	24	21	17	11
1.4	28	23	20	16	10
1.5	27	22	18	15	10
1.6	24	21	17	14	9
1.7	24	20	17	14	9
1.8	23	20	16	12	9
1.9	22	18	16	12	9
2.0	22	18	15	12	8
2.5	21	17	15	11	8
3.0	20	17	14	11	8

Fig. 2 - Value of N for a given Relative Shift for α = 0.05

Pacific Radiation's experience has been that two grids are confusing to the survey technologists. We use only the sample grid. We have also discovered that a vertical laser (common in the construction industry) easily allows transfer of a floor sample grid to the corresponding ceiling grid in case the same sample spacing can be used.

Complete a Survey Unit Radiation Survey Plan for each SU, then complete a Final Status Survey Checklist.

- **Were at least "N" samples taken?**
- **Were reference background samples taken?**
- **Was enough surface area scanned?**
- **Were all sample locations documented?**
- **Were all meters & technicians identified on field data logs?**
- **Did the removable fraction fall below 10%?**

Step 12 - Determine Compliance

Complete a Data Interpretation Checklist to verify that all is in order.

Convert the data to "Standard Units," i.e., dpm per 100 sq cm for surface measurements and pCi/g for soils, evaluate hot spots ("Elevated

Measurements") to see if "N" must be increased, check raw data for suspicious trends or unusual results, conduct the Sign Statistical Test (contaminant not in background) or WRS Statistical Test (contaminant in background) for each survey unit (SU) for each type of contaminant found, calculate average concentrations in each SU, and compare results to the DCGLs. The statistical test details are described in MARSSIM sections 8.3 and 8.4.

Make a decision whether the Survey Unit meets the radiological criteria for license termination. The NUREG-1757 Volume 2 contains Tables that enable this decision. The statistical testing is different depending whether the contaminant is present in the site background radiation field or it is not. The Tables are reproduced here as Tables 1 and 2.

The "Measurement Results" referred to in these Tables are the Direct Measurement numbers obtained in the Survey Unit.

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Table 1: Final Decision When Contaminant NOT Present in Background

Measurement Result	Conclusion
All concentrations < DCGL	Survey Unit meets release criterion
Average concentration > DCGL	Survey Unit fails
Any concentration > DCGL and average concentration < DCGL	Conduct Sign Test and elevated measurement comparison

.

Table 2: Final Decision When Contaminant Present in Background

Measurement Result	Conclusion
Difference between max survey unit concentration and min reference area concentration is < DCGL	Survey Unit meets release criterion
Difference between survey unit ave. concentration and reference area ave. concentration is > DCGL	Survey Unit fails
Difference between any survey unit concentration and any reference area concentration is > DCGL and the difference of survey unit ave. concentration and reference area ave. concentration is < DCGL	Conduct WRS test and elevated measurement comparison

Make a decision whether the entire site meets the radiological criteria for

license termination, i.e., all Survey Units are releaseable.

Perform an ALARA Analysis to demonstrate that the residual radioactivity has been reduced to a level that is ALARA per 10 CFR 20.1402, 1403(a & e) and 1404(a)(3).

Just when you thought you were done, the NRC throws this in. The point is that just decontaminating a site to the guideline level (25 mrem/ year) is not enough. The remediation efforts, which may have been costly and lengthy, still do not address the basic foundation of modern radiation protection practice - IS IT AS LOW AS REASONABLY ACHIEVABLE? If we spent more time and money, could we clean the site to even lower contaminant levels and cause a significant decrease in the annual dose rate to the critical population that moves into our site in the future? This is clearly an ALARA problem as was discussed in Chapter 11. The same solution is employed - a cost-benefit study is performed. Appendix "N" of NUREG-1757 Volume 2 is a great resource for this step. It includes some very specific formulas for calculating the benefit (reduced Collective Dose in person-rem) and the costs. The Guide uses a figure of $2,000 for each person-rem avoided. It also describes circumstances when "the results of an ALARA analysis are known on a generic basis and an analysis is not necessary." It also defines what constitutes a "net public or environmental harm," it demonstrates when ALARA is "not technically achievable" and describes how to make a claim of "prohibitively expensive." All of these ideas may be used if applicable when making the ALARA analysis for the site.

The last step is to issue a Final Report to the Client.

NOW you are finished!

Starting on the next page is a Checklist which will hopefully make it easier to thread your way through the multitude of actions which make up a MARSSIM D and D project.

MARSSIM - Twelve-Step Checklist

Step 1 - Historical Site Assessment

- ❑ Identify property
- ❑ Have client complete the HSA Questionnaire
- ❑ License reviewed
- ❑ Accidents/spills reviewed

Step 2 - Identify Contaminants

- ❑ Likely radioactivity on surfaces
- ❑ Likely radioactivity in soils

Step 3 - Establish DCGLs

- ❑ Run each radionuclide on the DandD code.
- ❑ Apply the Unity Rule to the DandD results
- ❑ Consult & set DCGL for surfaces
- ❑ Consult & set DCGL for soils

Step 4 - Classify Areas by Contamination Potential

- ❑ Assign a preliminary class to each area of the site.
- ❑ Select Background Reference Area as needed.

Step 5 - Define Survey Units

- ❑ Class 1 Units
- ❑ Class 2 Units
- ❑ Class 3 Units

Step 6 - Choose Survey Instrumentation and Procedures

- ❑ Scanning survey meters/procedures
- ❑ Direct measurement survey meters/procedures
- ❑ Wipe test meters/procedures as necessary
- ❑ Information Qualification form for each meter
- ❑ Calibrate each meter

Step 7 - Plan & Conduct Scoping Surveys

- ❑ Select about 20 sample locations
- ❑ Make & evaluate measurements

Step 8 - Plan & Conduct Characterization Surveys

- ❑ Select sample locations within each Survey Unit

- ❑ Make & evaluate measurements
- ❑ Calculate the standard deviation for each Survey Unit
- ❑ Calculate the removable fraction, F_r, for each Survey Unit

Step 9 - Establish DQOs to Evaluate Final Survey Results

- ❑ Choose a value for α.
- ❑ Choose a value for β.
- ❑ If $F_r > 0.1$, then reduce a DandD derived DCGL by a factor of 10

Step 10 - Plan and Conduct Final Scanning Surveys

- ❑ Scan 100% of Class 1 areas
- ❑ Scan appropriate fraction of Class 2 areas.
- ❑ Scan appropriate fraction of Class 3 areas.

Step 11 - Plan and Conduct Final Direct Measurement Surveys

- ❑ Calculate N for each Survey Unit
- ❑ Calculate sample grid spacing
- ❑ Calculate random start coordinates for Class 1 and Class 2 square sample grids.
- ❑ Calculate random sample locations for Class 3 areas
- ❑ Complete a SU Survey Plan for each SU
- ❑ Complete a Final Status Survey Checklist

Step 12 - Determine Compliance

- ❑ Complete Data Interpretation Checklist
- ❑ Site meets radiological criteria for termination
- ❑ Perform ALARA Analysis
- ❑ Issue Final Report
- ❑ Submit invoice for payment!

A-1

Radionuclide Decay Information

This appendix contains a listing of the major radiations, and their energies and intensities for radionuclides which may be encountered by a practicing radiation protection technologist. Most of the numerical data is courtesy of L. Dillman and the Society of Nuclear Medicine.

The format for the listings is to first give the chemical symbol and the isotope number. This is followed by radiation type (A = alpha particle, B = negative beta particle, P = positron, G = gamma ray, and SF = spontaneous fission), and the energy of the ray in MeV, where the maximum (E_{max}) of the spectrum is listed for both beta and positron sources. Next, the percentage of the disintegrations which produce that ray is listed in parentheses. Finally, the physical half-life is tabulated.

If you are interested in the most up-to-date information or in more detailed information than is provided for in the summary, go online to

http://www.nndc.bnl.gov/mird/index.html.

If you are really serious about nuclear data, go online to

http://www.nndc.bnl.gov/nudat2/.

Nuclide	Major Radiations, Energies and (%)	Half-Life
n-1	B,0.78(100)	12m
H-3	B,0.0186(100)	12.3y
C-11	P,0.96(99.7),G,0.511(199.5)	20.38m
C-14	B,0.1561(100)	5730y
N-13	P,1.20(99.8),G,0.511(199.6)	9.96m
N-16	G,7.11(5),G,6.13(69),B,4.3(68)	7.13s
O-15	P,1.700(100), G,0.511(200)	124s
F-18	P,0.633(97)	110m
Na-22	G,1.27(100), P,.546(90.6), G,.511(181)	2.60y
Na-24	B,1.392(100), G,1.37(100), G,2.75(100)	15h
Al-28	B,2.86(100), G,1.78(100)	2.24m
Si-32	B,0.214(100)	104y
P-32	B,1.71(100)	14.3d
S-35	B,0.167(100)	87.0d
Cl-36	B.0.71(99)	3.01E5y
Ar-41	B,1.98(99.2), G,1.293(99.2)	1.83h
K-40	B,1.30(89.5), G,1.46(0.103)	1.27 E9y
Cr-51	G,0.32(9.8)	27.7d
Mn-54	G,0.835(99.9), G,0.005(14)	312d

Fe-55	G,0.006(23), B,0.005(52)	2.70y
Co-57	G,0.122(86), G,0.136(10.4)	270d
Co-58	G,.811(99.5), P,.474(15), G,.511(31)	71.3d
Co-60	B,.313(99.8), G,1.17(99.8) G,1.33(100)	5.26y
Cu-62	P,2.93(97.6)	9.74m
Zn-65	G,1.115(50.6) P,.325(1.5) G,.511(3)	243d
Kr-85	B,.672(99.6), G,0.514(0.4)	10.7y
Sr-89	B,1.463(100), G,0.91(0.01)	52d
Sr-90	B,0.546(100) + Y-90 daughter	28.1y
Y-88	G,0.898(93), G,1.84(99.4)	107d
Y-90	B,2.273(100)	64.0h
Mo-99	B,1.23(80), G,.74(14)	66.7h
Tc-99	B,0.292(100)	2.1 E5y
Tc-99m	G,0.141(88)	6.03h
Cd-109	G,0.022(56)	453d
In-115m	G,0.335(39)	4.50h
In-116m	G,1.27(75) G,1.085(54) B,1.0(51)	54.0m
I-123	G 0.159(84), G,0.027(71)	13.0h
I-125	G,0.035(6.7), G,0.027(115)	60.2d
I-129	B,0.15(100), G,0.039(8)	1.57 E7y
I-131	B,.606(90) G,.364(82) G,.637(6.5)	8.06d
Xe-133	B,.346(98) G,.081(36) G,0.031(39)	5.31d

Cs-137	B,0.514(95) + Ba-137m Daughter	30.0y
Ba-133	G,0.081(33), G,0.303(18), G,0.356(60)	10.5y
Ba-137m	G,0.662(90)	2.55m
Pm-147	B,0.224(100)	2.62y
Ir-192	B,0.67(49), G, 0.308(28), G, 0.468(46)	74.2d
Au-198	G, 0.412(96), B, 0.961(99)	2.69d
Hg-203	B,0.212(100), G, 0.279(82)	46.5d
Pb-210	B,0.016(80), B,0.063(20), G,0.046(4)	22.3y
Pb-214	B,.672(48),.729(42),G,.295(19),G.352(37)	26.8m
Bi-214	B,1.51(18),1.54(18),3.27(17),G,.609(46)	19.9m
Po-210	A,5.305(100) + daughters	138d
Po-218	A, 6.00(100)	3.05m
Rn-222	A,5.49(100) + daughters	3.825d
Ra-226	A,4.78(95),G,0.186(4) + daughters	1602y
Th-230	A,4.62(23.4), A,4.69(76.3)	7.7E4y
U-235	A, 4.40(56), G, 0.186(54) + daughters	7.04 E8y
U-238	A, 4.2(75), A, 4.15(25) + daughters	4.5 E9y
Pu-238	A, 5.50(72), A, 5.46(28)	87.75y
Pu-239	A, 5.16(74), A, 5.14(15)	2.4 E4y
Am-241	A, 5.48(85), G, 0.0595(38)	433y
Cf-252	SF(3.1), A,6.075(15), A,6.12(81.6)	2.64y

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Dose Rate Factors for Skin Exposure

The tables in this Appendix have been reproduced, with permission, from Health Physics 53, pages 138 - 141, Kocher and Eckerman, 1987, Pergamon Journals, Limited. The proper use of the information is detailed in Chapter 5, along with a sample calculation. Information on choosing the best tissue depth is also provided in Chapter 5. Briefly, the following table can be used to approximate the thickness of the skin at the indicated body locations:

Body Location	Approximate skin thickness
Head, trunk, upper arm, upper leg	4 mg/sq cm
Lower arm, wrist, back of hand, lower leg, top of foot	8 mg/sq cm
Palm of hand, sole of foot	40 mg/sq cm
U.S. NRC regulatory value	7 mg/sq cm

"The Dose Rate Factors" in this compilation are in units of Sv/yr per Bq/cm^2. If desired, the Dose Rate Factor" can be calculated in units of rem/hr per $\mu Ci/cm^2$ by multiplying the table value by 422.

Nuclide	Ha1f-life	4 mg/sq cm	8 mg/sq cm	40 mg/sq cm	7 mg/sq cm
H-3	12.28 y	0.00	0.00	0.00	0.00
Be-10	l.6 x 1O+6 y	2.2 x 10-2	1.6 x 10-2	3.2 x 10-3	1.7 x 10-2
C-1 1	20.48 m	2.4 x 10-2	1.9 x 10-2	7.9 x 10-3	2.0 x 10-2
C-14	5.73 x 10+3 y	7.9 x 10-3	2.1 x 10-3	0.00	2.9 x 10-3
F-l8	109.74 m	2.3 x 10-2	1.7 x 10-2	4.6 x 10-3	1.8 x 10-2
Na-22	2.602 y	2.1 x 10-2	1.5 x 10-2	3.3 x 10-3	1.7 x 10-2
P-32	14.29d	2.4 x 10-2	2.0 x 10-2	1.1 x 10-2	2.1 x 10-2
P-33	25.4d	l.4 x 10-2	6.5 x 10-3	2.5 x 10-5	7.6 x 10-3
S-35	87.44d	7.9 x 10-3	2.3 x 10-3	0.00	3.1 x 10-3
Cl-36	3.01 x 10+5 y	2.2 x 10-2	1.6 x 10-2	4.7 x 10-3	1.7 x 10-2
K-40	1.277 x 10+9 y	2.1 x 10-2	1.7 x 10-2	8.3 x 10-3	1.8 x 10-2
Ca-45	162.7 d	1.4 x 10-2	6.5 x 10-3	3.2 x 10-5	7.7 x 10-3
Sc-46	83.80 d	1.7 x 10-2	1.0 x 10-2	5.5 x l0-4	1.2 x 10-2
Cr-51	27.704 d	0.00	0.00	0.00	0.00
Mn-54	312.7 d	0.00	0.00	0.00	0.00
Fe-55	2.7 y	0.00	0.00	0.00	0.00
Fe-59	44.63 d	1.7 x 10-2	1.0 x 10-2	9.4 x 10-4	1.1 x 10-2
Co-57	270.9d	9.6 x 10-4	6.1 x 10-4	0.00	6.9 x 10-4
Co-58	70.80 d	3.6 x l0-3	2.6 x 10-3	4.4 x 10-4	2.8 x 10-3
Co-60	5.271 y	1.6x 10-2	8.7 x 10-3	2.5 x 10-4	9.9 x 10-3
Ni-59	7.5 x 10+4 y	0.00	0.00	0.00	0.00
Ni-63	100.1 y	1.6 x 10-4	0.00	0.00	0.00
Zn-65	244.4d	3.3 x 10-4	2.1 x 10-4	l.0 x 10-5	2.3 x 10-4
Ga-67	3.261 d	7.6 x 10-3	1.4 x 10-3	0.00	2.6 x 10-3
Ga-68	68.0 m	2.1 x 10-2	1.8 x 10-2	1.1 x 10-2	1.9 x 10-2
Se-75	1 19.78 d	1.5 x 10-3	6.5 x 10-4	2.9 x 10-5	8.4 x 10-4
Se-79	6.5 x 10+4 y	l.0 x 10-2	2.8 x 10-3	0.00	3.8 x 10-3
Rb-86	18.66d	2.3 x 10-2	1.9 x 10-2	1.0 x 10-2	2.0 x 10-2
Rb-88	17.8m	2.6 x 10-2	2.3 x 10-2	1.5 x 10-2	2.3 x 10-2
Sr-85	64.84 d	1.5 x 10-4	1.3 x 10-4	7.3 x 10-5	1.3 x 10-4
Sr-89	50.55 d	2.3 x 10-2	1.9 x 10-2	9.8 x 10-3	2.0 x 10-2
Sr-90	28.6 y	2.1 x 10-2	1.5 x 10-2	3.4 x 10-3	1.6 x 10-2
Sr-91	9.5 h	2.3 x 10-2	1.9 x 10-2	9.5 x 10-3	2.0 x 10-2
Y-90	64.1 h	2.4 x 10-2	2.0x 10-2	1.2 x 10-2	2.1 x 10-2
Y-91	58.51 d	2.3 x 10-2	1.9x 10-2	9.9 x 10-3	2.0 x 10-2
Zr-93	1.53 x 10+6 y	2.1 x 10-4	0.00	0.00	0.00
Zr-95	64.02d	1.7 x 10-2	1.0 x 10-2	7.4 x 10-4	1.2 x 10-2
Zr-97	16.90 h	2.3 x 10-2	1.9 x 10-2	1.0 x 10-2	2.0 x 10-2
Nb-93m	14.6 y	0.00	0.00	0.00	0.00
Nb-94	2.03 x 10+4 y	1.9 x 10-2	1.2 x 10-2	1.6 x 10-3	1.4 x 10-2
Nb-95	35.06d	6.4 x 10-3	l.7 x 10-3	1.8 x 10-5	2.3 x 10-3

Nuclide	Ha1f-life	4 mg/sq cm	8 mg/sq cm	40 mg/sq cm	7 mg/sq cm
Nb-95m	86.6 h	1.9 x 10-2	1.5 x 10-2	1.6 x 10-3	1.6 x 10-2
Nb-97	72.1 m	2.3 x 10-2	1.9 x 10-2	8.6 x 10-3	1.9 x 10-2
Nb-97m	60 s	4.6 x 10-4	4.0 x 10-4	2.5 x 10-4	4.1 x 10-4
Mo-93	3.5 x 10+3 y	0.00	0.00	0.00	0.00
Mo-99	66.02 h	2.3 x 10-2	1.8 x 10-2	7.1 x 10-3	1.9 x 10-2
Tc-99	2.13 x 10+5y	1.4 x 10-2	7.4 x 10-3	1.2 x 10-4	8.6 x 10-3
Tc-99m	6.02 h	2.9 x 10-3	1.8 x 10-3	0.00	2.1 x 10-3
Ru-103	39.35 d	1.1 x 10-2	4.8 x 10-3	2.1 x 10-4	5.8 x 10-3
Ru-l05	4.44 h	2.3 x 10-2	1.8 x 10-2	7.6 x 10-3	1.9 x 10-2
Ru-106	368.2 d	0.00	0.00	0.00	0.00
Rh-103m	56.119m	0.00	0.00	0.00	0.00
Rh-l05	35.36 h	1.8 x 10-2	1.2 x 10-2	2.1 x 10-3	l.3 x 10-2
Rh-105m	45 s	2.3 x 10-2	1.3 x 10-2	0.00	1.6 x 10-2
Rh-106	29.92 s	2.5 x 10-2	2.1 x 10-2	1.4 x 10-2	2.2 x 10-2
Pd-103	16.961 d	0.00	0.00	0.00	0.00
Pd-107	6.5 x 10+6 y	0.00	0.00	0.00	0.00
Ag-108	2.37 m	2.3 x 10-2	1.9 x10-2	9.9 x 10-3	2.0 x 10-2
Ag-108m	127 y	3.7 x 10-4	2.3 x10-4	1.3 x 10-4	2.5 x 10-4
Ag-109m	39.6 s	1.8 x 10-2	2.3 x 10-3	0	4.5 x 10-3
Ag-110	24.57 S	2.4 x 10-2	2.1 x10-2	1.3 x 10-2	2.2 x 10-2
Ag-110m	249.85 d	6.9 x10-3	4.3 x 10-3	7.2 x 10-4	4.7 x 10-3
Cd-109	464 d	0.00	0.00	0.00	0.00
Cd-113m	13.7 y	2.0 x 10-2	1.4 x10-2	2.9 x 10-3	1.5 x 10-2
Cd-115m	44.6 d	2.3 x 10-2	1.9 x10-2	9.9 x 10-3	2.0 x 10-2
In-111	2.83 d	4.2 x 10-3	3.1 x10-3	1.1 x 10-4	3.3 x 10-3
In-113m	1.658 h	8.4 x 10-3	7.1 x10-3	3.5 x 10-3	7.4 x 10-3
Sn-113	115.l d	0.00	0.00	0.00	0.00
Sn-119m	293.0 d	7.5 x 10-3	0.00	0.00	0.00
Sn-126	1.0 x l0+5 y	1.6 x 10-2	5.9 x 10-3	1.8 x 10-5	7.3 x 10-3
Sb-124	60.20 d	2.1 x 10-2	1.5 x 10-2	5.5 x 10-3	1.6 x 10-2
Sb-125	2.77 y	1.2 x 10-2	6.5 x 10-3	7.5 x 10-4	7.4 x 10-3
Sb-126	12.4 d	2.0 x 10-2	1.5 x 10-2	4.6 x 10-3	1.6 x 10-2
Sb-126m	19.0 m	2.1 x 10-2	1.7 x 10-2	9.2 x 10-3	1.8 x 10-2
Sb-127	3.85 d	2.2 x 10-2	1.7 x 10-2	5.9 x 10-3	1.8 x 10-2
Sb-129	4.40 h	2.2 x 10-2	1.6 x 10-2	5.7 x 10-3	1.7 x 10-2
Te-123m	119.7 d	2.0 x10-2	5.2 x 10-3	0.00	7.8 x 10-3
Te-125m	58 d	2.6 x10-2	7.0 x 10-3	0.00	1.0 x 10-2
Te-127	9.35 h	2.1 x10-2	1.5 x 10-2	4.0 x 10-3	1.6 x 10-2
Te-127m	109 d	1.6 x10-2	2.5 x 10-3	9.4 x 10-5	4.7 x 10-3
Te-129	69.6 m	2.3 x10-2	1.9 x 10-2	9.1 x 10-3	2.0 x 10-2
Te-129m	33.6 d	2.3 x 10-2	1.1 x 10-2	3.5 x 10-3	1.3 x 10-2

Nuclide	Ha1f-life	4 mg/sq cm	8 mg/sq cm	40 mg/sq cm	7 mg/sq cm
Te-131	25.0 m	2.8 x 10-2	2.2 x 10-2	1.0 x 10-2	2.3 x 10-2
Te-131m	30 h	2.2 x10-2	1.4 x 10-2	1.7 x 10-3	1.5 x 10-2
Te-132	78.2 h	1.3 x10-2	5.9 x 10-3	4.7 x 10-5	7.0 x 10-3
I-123	13.13 h	4.3 x10-3	2.9 x 10-3	0.00	3.2 x 10-3
I-125	60.14 d	0.00	0.00	0.00	0.00
I-129	1.57x 10+7 y	5.7 x10-3	1.3 x 10-3	0.00	1.9 x 10-3
I-131	8.040 d	2.1 x10-2	1.4 x 10-2	3.0 x 10-3	1.5 x 10-2
I-132	2.30 h	2.3 x 10-2	1.8 x 10-2	8.2 x 10-3	1.9 x 10-2
I-133	20.8 h	2.3 x 10-2	1.8 x 10-2	7.6 x 10-3	1.9 x 10-2
I-134	52.60 m	2.4 x 10-2	1.9 x 10-2	9.7 x 10-3	2.0 x 10-2
I-135	6.61 h	2.2 x 10-2	1.7 x 10-2	6.5 x 10-3	1.8 x 10-2
Cs-134	2.062 y	1.6 x 10-2	1.l x l0-2	2.7 x 10-3	1.2 x 10-2
Cs-135	2.3 x 10-6 y	9.6 x 10-3	3.6 x 10-3	5.0 x 10-7	4.5 x 10-3
Cs-136	13.16 d	2.0 x 10-2	1.2 x 10-2	5.9 x 10-4	1.3 x 10-2
Cs-137	30.17 y	2.0 x 10-2	1.3 x 10-2	2.3 x 10-3	1.4 x 10-2
Ba-137m	2.552 m	2.4 x 10-3	2.0 x 10-3	1.2 x 10-3	2.1 x 10-3
Ba-140	12.789 d	2.2 x 10-2	1.6 x 10-2	5.0 x 10-3	1.7 x 10-2
La-140	40.22 h	2.4 x 10-2	1.9 x 10-2	9.2 x 10-3	2.0 x 10-2
Ce-141	32.50 d	2.5 x 10-2	1.5 x 10-2	1.6 x 10-3	1.7 x 10-2
Ce-143	33.0 h	2.4 x 10-2	1.8 x 10-2	7.7 x 10-3	1.9 x 10-2
Ce-144	284.3 d	1.5 x 10-2	7.6 x 10-3	1.7 x 10-4	8.9 x 10-3
Pr-143	13.56 d	2.2 x 10-2	1.7 x 10-2	6.2 x 10-3	1.8 x 10-2
Pr-144	17.28 m	2.4 x 10-2	2.1 x 10-2	1.3 x 10-2	2.2 x 10-2
Pr-144m	7.2 m	1.9 x 10-3	0.00	0.00	0.00
Nd-147	10.98 d	2.3 x 10-2	1.5 x 10-2	4.2 x 10-3	1.7 x 10-2
Pm-147	2.6234 y	1.1 x 10-2	4.4 x 10-3	3.6 x 10-6	5.4 x 10-3
Sm-151	90 y	2.5 x 10-4	2.2 x 10-5	0.00	5.2 x 10-5
Eu-152	13.6 y	1.4 x 10-2	6.9 x 10-3	1.5 x 10-3	7.9 x 10-3
Eu-154	8.8 y	3.0 x 10-2	1.6 x 10-2	3.3 x 10-3	1.8 x 10-2
Eu-155	4.96 y	7.6 x 10-3	2.2 x 10-3	2.8 x 10-6	2.9 x 10-3
Gd-153	241.6 d	3.5 x 10-3	7.4 x 10-4	0.00	1.1 x 10-3
Tb-160	72.3 d	3.0 x 10-2	1.6 x 10-2	3.7 x 10-3	1.8 x 10-2
Ho-166m	1.2 x 10-3 y	2.2 x 10-2	6.8 x 10-3	5.9 x 10-4	8.3 x 10-3
Yb-169	31.97 d	1.4 x 10-2	7.7 x 10-3	3.9 x 10-5	8.8 x 10-3
Os-185	93.6 d	5.3 x 10-4	2.8 x 10-4	1.1 x 10-4	3.2 x 10-4
Os-191	15.4 d	1.0 x 10-2	3.5 x 10-3	0.00	4.3 x 10-3
Ir-192	74.02 d	2.3 x 10-2	1.6 x 10-2	3.4 x 10-3	1.7 x 10-2
Au-198	2.696 d	2.3 x 10-2	1.8 x 10-2	6.4 x 10-3	1.9 x 10-2
Hg-203	46.60 d	1.6 x 10-2	8.5 x 10-3	3.7 x 10-4	9.6 x 10-3
TI-201	73.06 h	5.3 x 10-3	1.6 x 10-3	0.00	2.3 x 10-3
Tl-204	3.779 y	2.1 x 10-2	1.6 x 10-2	5.0 x 10-3	1.7 x 10-2
Tl-207	4.77 m	2.3 x 10-2	1.9 x 10-2	8.7 x 10-3	1.9 x 10-2
Tl-208	3.053 m	2.5 x 10-2	2.0 x 10-2	9.8 x 10-3	2.1 x 10-2
Tl-209	2.20 m	2.5 x 10-2	2.1 x 10-2	1.1 x 10-2	2.2 x 10-2

Nuclide	Ha1f-life	4 mg/sq cm	8 mg/sq cm	40 mg/sq cm	7 mg/sq cm
Pb-209	3.253 h	2.0 x 10-2	1.4 x 10-2	3.3 x 10-3	1.6 x 10-2
Pb-210	22.26 y	1.8 x 10-5	0.00	0.00	0.00
Pb-211	36.1 m	2.3 x 10-2	1.8 x 10-2	8.2 x 10-3	1.9 x 10-2
Pb-212	10.643 h	2.7 x 10-2	1.7 x 10-2	6.8 x 10-4	1.9 x 10-2
Pb-214	26.8 m	2.8 x 10-2	2.0 x 10-2	4.8 x 10-3	2.2 x 10-2
Bi-210	5.013 d	2.3 x 10-2	1.8 x 10-2	7.4 x10-3	1.9 x 10-2
Bi-211	2.13 m	8.8 x 10-4	7.1 x 10-4	2.2 x 10-4	7.4 x 10-4
Bi-212	60.55 m	1.5 x 10-2	1.2 x 10-2	6.5 x 10-3	1.3 x 10-2
Bi-213	45.65 m	2.3 x 10-2	1.9 x 10-2	8.1 x 10-3	1.9 x 10-2
Bi-214	19.9 m	2.3 x 10-2	1.9 x 10-2	9.6 x 10-3	2.0 x 10-2
Po-210	138.378 d	0.00	0.00	0.00	0.00
Po-211	0.516s	0.00	0.00	0.00	0.00
Po-212	2.98 x 10-7 s	0.00	0.00	0.00	0.00
Po-213	4.2 x 10-6 s	0.00	0.00	0.00	0.00
Po-214	1.637 x 10-4 s	0.00	0.00	0.00	0.00
Po-215	1.778 x 10-3 s	0.00	0.00	0.00	0.00
Po-216	0.146 s	0.00	0.00	0.00	0.00
Po-218	3.05 m	0.00	0.00	0.00	0.00
At-217	0.0323 s	0.00	0.00	0.00	0.00
Fr-221	4.8 m	1.5 x 10-3	9.5 x 10-4	1.6 x 10-5	1.1 x 10-3
Fr-223	21.8 m	2.6 x 10-2	1.9 x 10-2	6.6 x 10-3	2.0 x 10-2
Ra-223	11.434 d	8.2 x 10-3	5.1 x 10-3	2.4 x 10-4	5.5 x 10-3
Ra-224	3.62 d	2.9 x 10-4	2.2 x 10-4	1.5 x 10-5	2.4 x 10-4
Ra-225	14.8 d	1.5 x 10-2	8.3 x 10-3	2.9 x 10-4	9.5 x 10-3
Ra-226	1600 y	5.9 x 10-4	3.7 x 10-4	0.00	4.2 x 10-4
Ra-228	5.75 y	0.00	0.00	0.00	0.00
Ac-225	l0.0 d	2.1 x l0-3	2.6 x 10-4	0.00	4.5 x 10-4
Ac-227	21.773 y	0.00	0.00	0.00	0.00
Ac-228	6.13 h	2.7 x 10-2	1.8 x 10-2	6.3 x 10-3	2.0 x 10-2
Th-227	18.718 d	2.4 x 10-3	1.4 x 10-3	7.2 x 10-5	1.5 x 10-3
Th-228	1.9132 y	4.3 x 10-3	1.5 x 10-4	0.00	4.0 x 10-4
Th-229	7.34 x10-3 y	1.3 x 10-2	3.9 x 10-3	4.0 x 10-6	5.0 x 10-3
Th-230	7.7 x 10-4 y	9.1 x 10-4	0.00	0.00	0.00
Th-231	25.52 h	1.9 x 10-2	6.8 x 10-3	9.5 x 10-5	8.2 x 10-3
Th-232	1.405 x 10+10 y	2.7 x 10-4	1.5 x 10-5	0	1.8 x 10-5
Th-234	24.10 d	9.5 x 10-3	2.2 x 10-3	4.3 x 10-7	3.l x l0-3
Pa-231	3.276 x 10+4 y	1.3 x 10-3	5.7 x 10-4	4.5 x 10-5	6.6 x 10-4
Pa-233	27.0 d	2.6 x 10-2	1.4 x 10-2	8.7 x 10-4	1.6 x l0-2
Pa-234	6.70 h	7.8 x 10-2	3.9 x 10-2	5.0 x 10-3	4.7 x 10-2
Pa-234m	1.17 m	2.4 x 10-2	2.0 x 10-2	1.1 x 10-2	2.l x 10-2
U-232	72 y	2.8 x 10-4	2.5 x 10-5	0.00	3.0 x 10-5
U-233	1.592 x 10+5 y	4.6 x 10-5	1.7 x 10-6	0.00	6.8 x 10-6
U-234	2.445 x 10 y	6.5 x 10-5	1.6 x 10-5	0.00	2.1 x 10-5
U-235	7.038 x 10-8 y	3.1 x 10-3	8.0 x 10-4	2.9 x 10-7	1.1 x 10-3

Nuclide	Ha1f-life	4 mg/sq cm	8 mg/sq cm	40 mg/sq cm	7 mg/sq cm
U-236	2.3415 x 10-7 y	4.0 x 10-5	1.3 x 10-5	0	1.9 x 10-5
U-238	4.468 x 10+9 y	3.7 x 10-5	9.8 x 10-6	0.00	1.6 x 10-5
U-240	14.1 h	1.8 x 10-2	1.1 x 10-2	1.1 x 10-3	1.2 x 10-2
Np-237	2.14 x l0+6 y	4.3 x 10-3	4.3 x 10-4	0.00	6.8 x 10-4
Np-238	2.117 d	1.8 x 10-2	1.1 x 10-2	3.5 x 10-3	1.2 x 10-2
Np-239	2.35 5 d	3.6 x 10-2	2.0 x 10-2	1.2 x 10-3	2.3 x 10-2
Np-240	65 m	5.4 x10-2	2.6 x 10-2	5.0 x 10-3	3.0 x 10-2
Np-240m	7.4 m	2.3 x10-2	1.9 x 10-2	9.3 x 10-3	1.9 x 10-2
Pu-236	2.851 y	0.00	0.00	0.00	0.00
Pu-238	87.75 y	0.00	0.00	0.00	0.00
Pu-239	24131 y	3.8 x 10-6	0.00	0.00	0.00
Pu-240	6537 y	0.00	0.00	0.00	0.00
Pu-24 I	14.4 y	0.00	0.00	0.00	0.00
Pu-242	3.758 x 10+5 y	0.00	0.00	0.00	0.00
Pu-243	4.956 h	2.0 x 10-2	1.3 x 10-2	2.2 x 10-3	1.4 x 10-2
Pu-244	8.26 X 10+7 y	0.00	0.00	0.00	0.00
Am-241	432.2 y	4.8 x 10-4	1.1 x 10-5	0.00	2.2 x 10-5
Am-242	16.02 h	1.7 x 10-2	1.2 x 10-2	2.6 x 10-3	1.3 x 10-2
Am-242m	152y	3.2 x 10-5	0.00	0.00	0.00
Am-243	7.38 X 10+3 y	1.1 x 10-3	6.2 x 10-6	0.00	4.1 x 10-5
Cm-242	163.2 d	0.00	0.00	0.00	0.00
Cm-243	28.5 y	1.7 x 10-2	9.8 x 10-3	3.0 x 10-4	1.1 x 10-2
Cm-244	18.11 y	0.00	0.00	0.00	0.00
Cm-245	8.5 X 10+3 y	8.6 x 10-3	5.1 x 10-3	0.00	5.8 x 10-3
Cm-246	4.75 X 10+3 y	0.00	0.00	0.00	0.00
Cm-247	1.56 X 10+7 y	1.4 x 10-3	1 x 10-3	1.6 x 10-4	1.1 x 10-3
Cm-248	3.39 X 10+5 y	0.00	0.00	0.00	0.00
Bk-249	320 d	3.5 x 10-3	5.2 x 10-4	0.00	8.2 x 10-4
Cf-249	350.6 y	2.8 x 10-3	1.8 x 10-3	1.4 x 10-4	1.9 x 10-3
Cf-250	13.08 y	4.2 x 10-5	1.0 x 10-6	0.00	4.7 x 10-6
Cf-252	2.639 y	3.4 x 10-5	1.1 x 10-6	0.00	4.8 x 10-6

A-3

Data for Neutron Instrument Calibrations

This appendix contains some practical information to assist in the calibration of neutron survey instrumentation using isotopic neutron sources. First, conversion factors between the neutron flux and the dose equivalent rate, in SI units and the "old" units, are tabulated. Next, geometric correction factors for non-uniform field conditions are given for the case of a spherical remmeter. Finally, the emitted neutron spectra from a variety of commonly available isotopic neutron sources are presented graphically.

For purposes of neutron survey meter calibrations, it is necessary to know the conversion factor between the neutron flux and the corresponding dose equivalent rate. If reliable measured values are available for a particular in-house source, that data should be used. Otherwise, the following approximate values of conversion factor will give a reasonable estimate of the neutron dose rate for a source. Listed conversion factors are averages of various published values.

Table 1 - Flux to Dose Equivalent Rate Conversion Factors

Neutron Source	mSv/hr per n/cm^2-sec	mrem/hr per n/cm^2-sec
Ac-227:Be	1.3×10^{-3}	0.13
Am-241:B	1.4×10^{-3}	0.14
Am-241:Be	1.4×10^{-3}	0.14
Am-241:F	1.3×10^{-3}	0.13
Am-241:Li	0.64×10^{-3}	0.064
Cf-252	1.2×10^{-3}	0.12
Cf-252+15cm D_2O	0.35×10^{-3}	0.035
Po-210:Be	1.2×10^{-3}	0.12
Po-210:Li	0.63×10^{-3}	0.063
Pu-238:Be	1.4×10^{-3}	0.14
Pu-239:Be	1.4×10^{-3}	0.14
Ra-226:Be	1.4×10^{-3}	0.14

Due to the wide availability of spherical remmeters (e.g., Eberline NRD and PNR-4, Ludlum Model 12-4, etc.) some additional information will be presented here regarding their calibration. Chapter 12 discussed the need for a uniform calibration field over the sensitive volume of a detector. Then, correction factors were given for the special case of a cylindrical ion chamber calibrated “end on” at close distances. A similar approach can be taken with a spherical neutron detector.

As explained in Chapter 12, neutron survey instrument calibration is greatly complicated by air scatter, room scatter and ground scatter components in addition to the primary neutrons. The best way to eliminate these scatter components (short of performing the calibration in deep space outside a spacecraft) is to place the calibration source very close to the detector, violating the important “uniform field” rule.

However, in the special case of a spherical detector and a source with small dimensions compared to the sphere radius, the correction factor, F, for non-uniformity can be calculated. This arrangement has one further benefit. Smaller activity sources can be used to reach calibration points on the higher ranges of the remmeter. The corrected dose equivalent rate for a source under these conditions is given by:

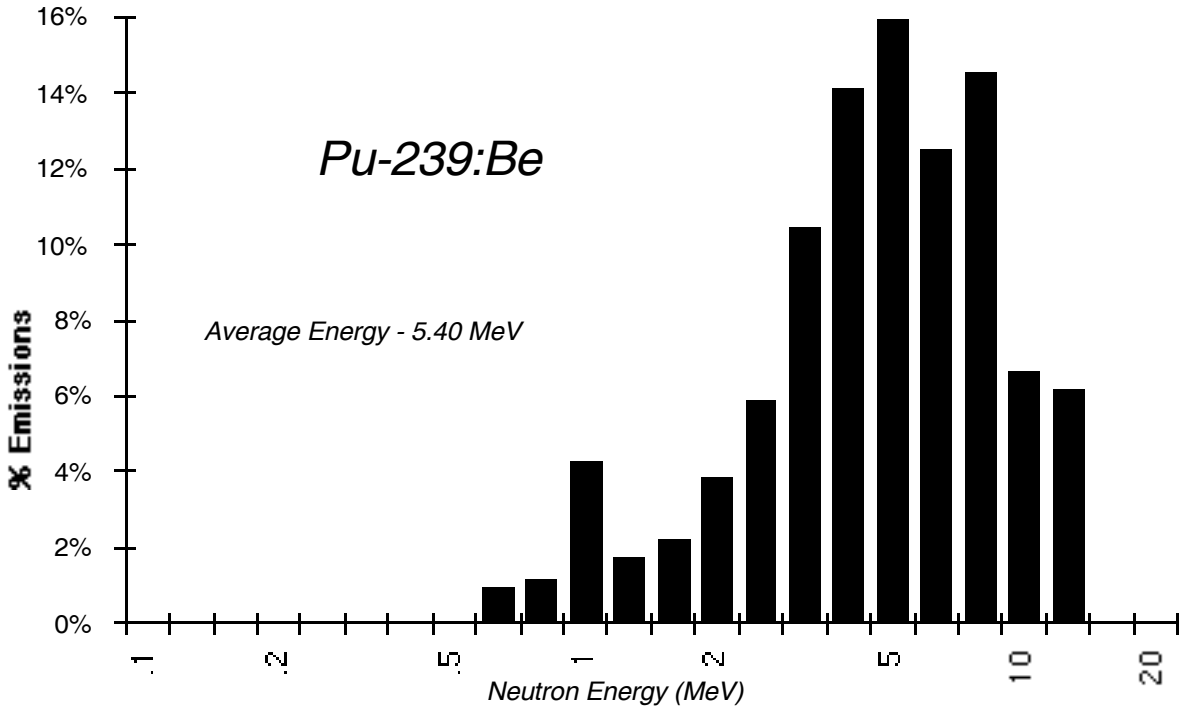
$$\mathbf{H/t\ (mrem/hr\ or\ mSv/hr)} = \frac{FCY}{4\pi r^2} \quad \textbf{[Eqn.1]}$$

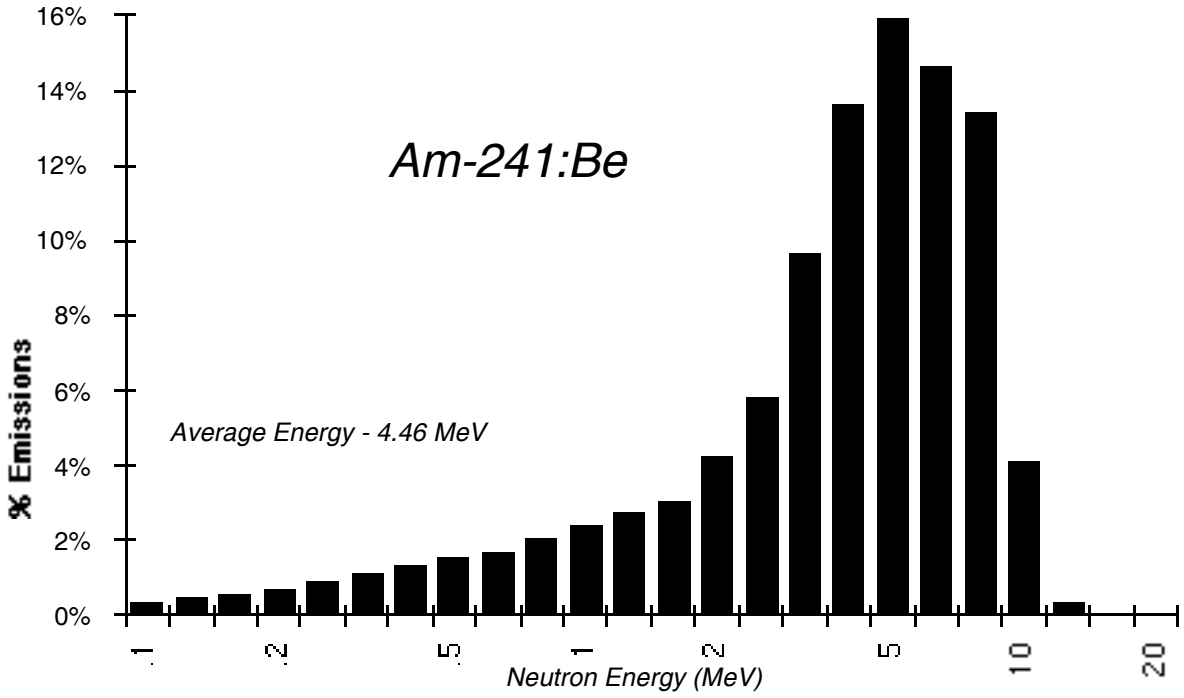
Here, C = the flux to dose equivalent rate conversion factor, Y = the neutron source output in neutrons/second, and r = the separation distance between the centers of the neutron source and the spherical detector in centimeters. The value of the non-uniformity correction, F, can be taken from the Table 2. Note that, for convenience, the distance in the first column is expressed as the separation in centimeters between the source center and the sphere front surface. Also, note that the sphere diameters are expressed in inches, not centimeters, since U.S. made detectors are usually supplied in inch increments.

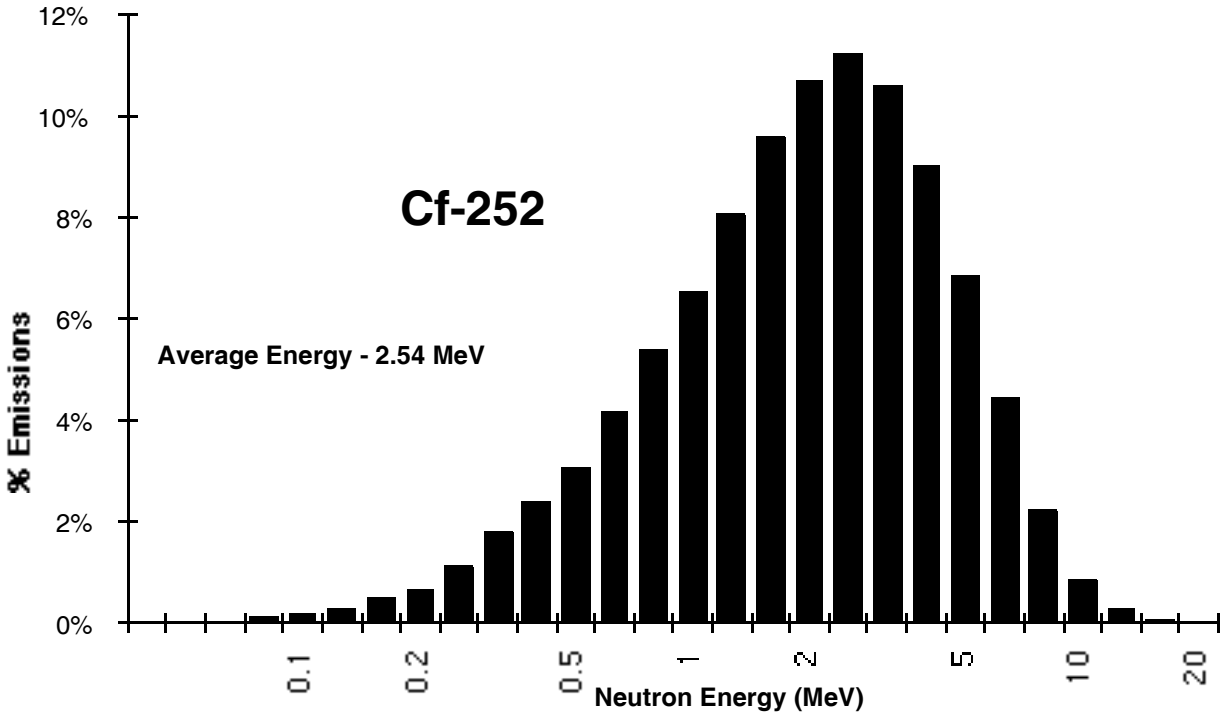
Table 2 - Correction Factor, F, for Spherical Neutron Detectors

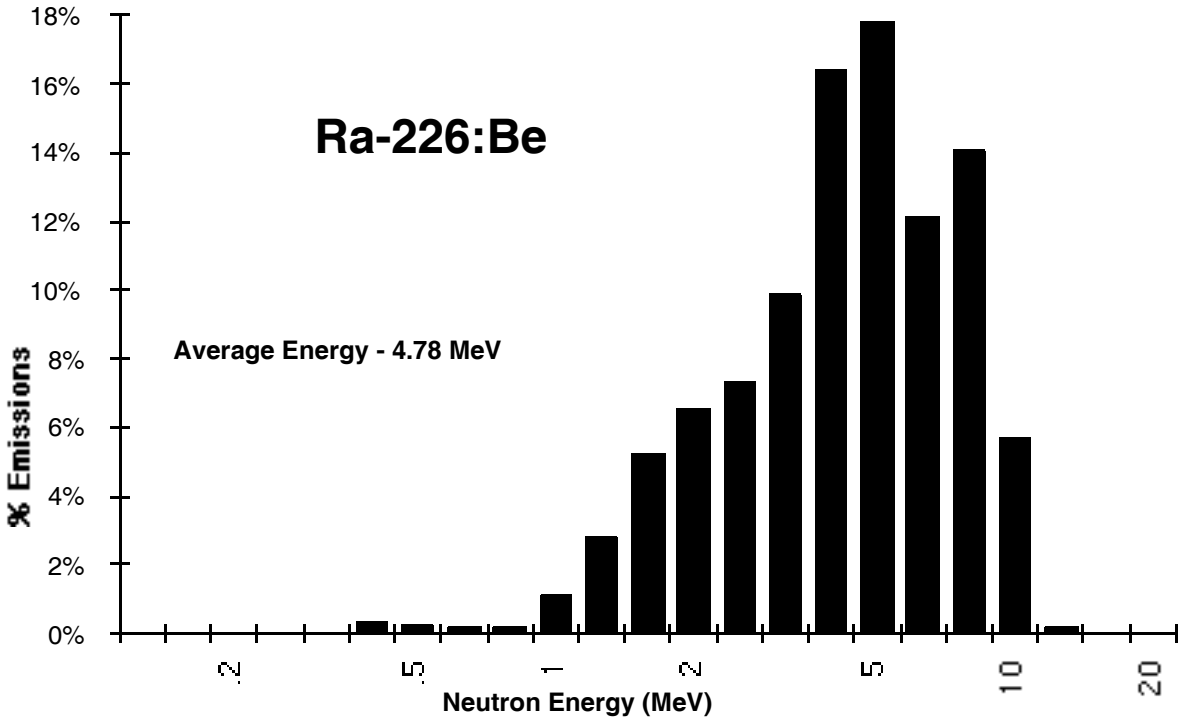
Source center to sphere front surface, cm	F for a 6 inch diameter sphere	F for an 8 inch diameter sphere	F for a 9 inch diameter sphere	F for a 10 inch diameter sphere
0	1.580	1.460	1.485	1.510
0.5	1.282	1.246	1.269	1.291
1	1.210	1.191	1.211	1.232
2	1.140	1.134	1.151	1.168
3	1.104	1.102	1.117	1.132
4	1.081	1.082	1.095	1.108
5	1.065	1.068	1.079	1.091

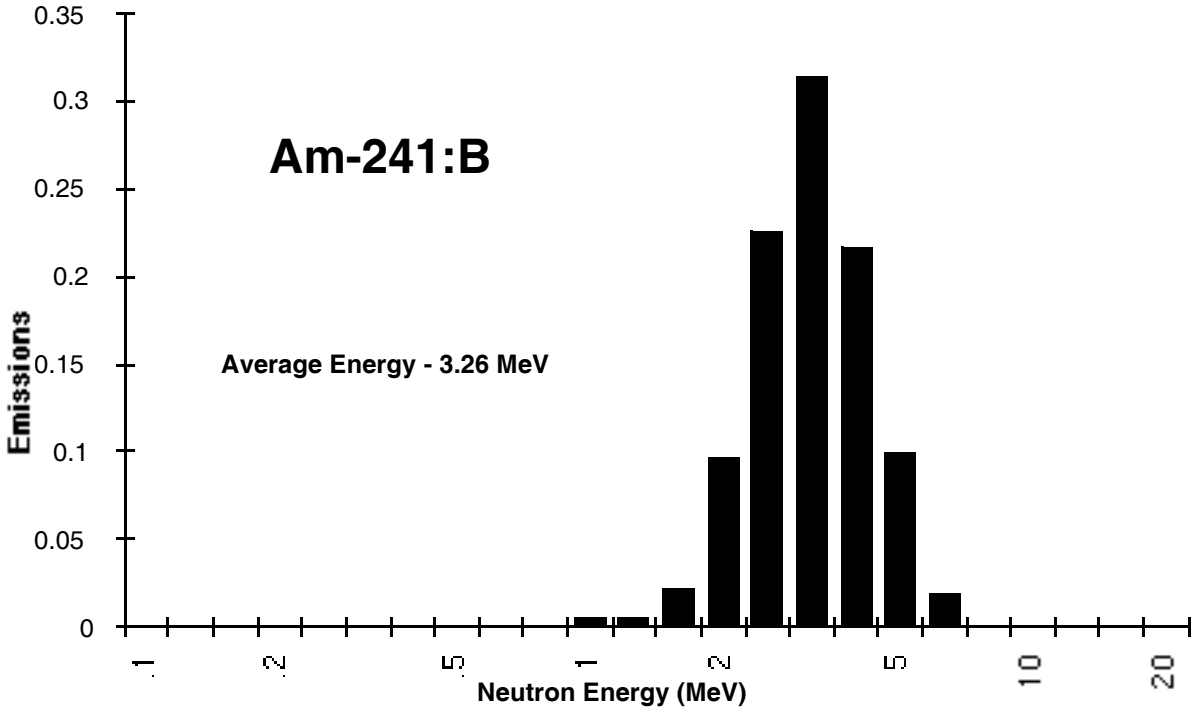
Finally, this appendix concludes with energy spectra of several commonly available isotopic and spontaneous fission sources. The targets selected are beryllium, lithium and boron to illustrate the effect that the target material has on the average neutron energy and spectrum shape.

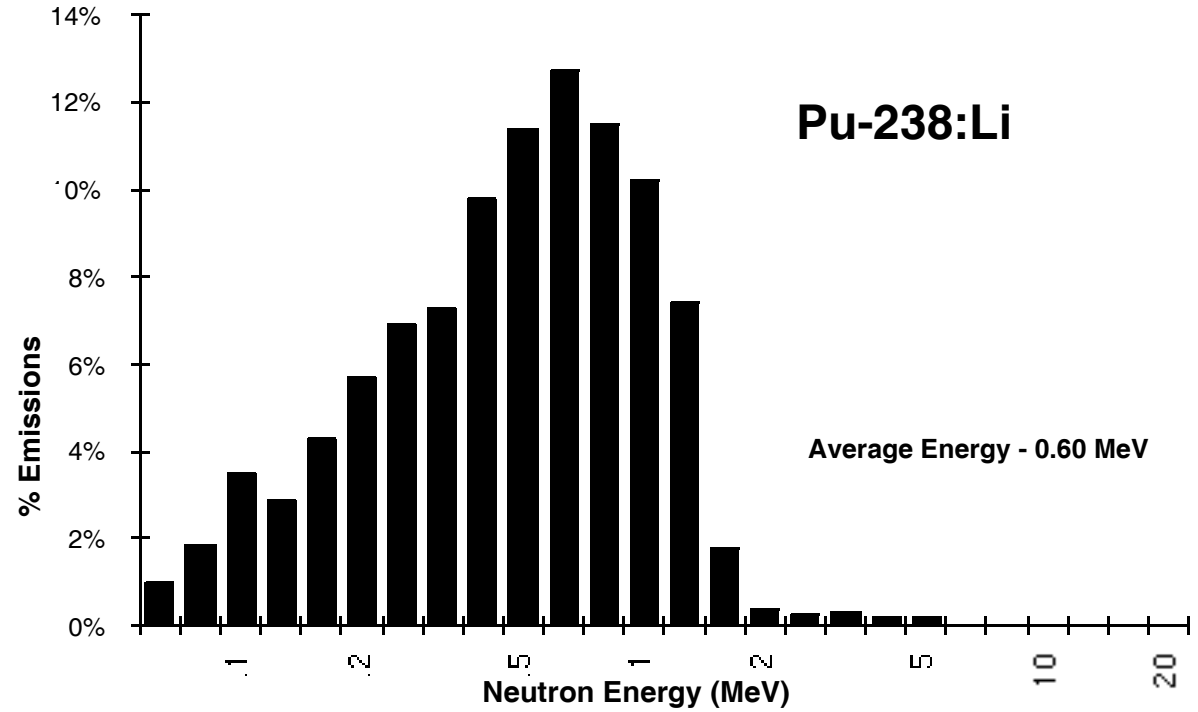

Pu-239:Be
Average Energy - 5.40 MeV
% Emissions
16%
14%
12%
10%
8%
6%
4%
2%
0%
.1
.2
.5
1
2
5
10
20
Neutron Energy (MeV)


Am-241:Be
Average Energy - 4.46 MeV
% Emissions
16%
14%
12%
10%
8%
6%
4%
2%
0%
.1
.2
.5
1
2
5
10
20
Neutron Energy (MeV)


Cf-252
Average Energy - 2.54 MeV
% Emissions
12%
10%
8%
6%
4%
2%
0%
0.1
0.2
0.5
1
2
5
10
20
Neutron Energy (MeV)


Ra-226:Be
Average Energy - 4.78 MeV
% Emissions
18%
16%
14%
12%
10%
8%
6%
4%
2%
0%
2
5
1
2
5
10
20
Neutron Energy (MeV)


Am-241:B
Average Energy - 3.26 MeV
Emissions
0.35
0.3
0.25
0.2
0.15
0.1
0.05
0
.1
.2
.5
1
2
5
10
20
Neutron Energy (MeV)


Pu-238:Li
Average Energy - 0.60 MeV
% Emissions
14%
12%
10%
8%
6%
4%
2%
0%
.1
.2
.5
1
2
5
10
20
Neutron Energy (MeV)

List of Acronyms Used

This section contains a list of all of the acronyms mentioned within this textbook. Each acronym is defined. The location in the text where each is found can be determined by looking up the acronym alphabetically in the Subject Index.

Acronym	Definition
A	
ABWR	Advanced Boiling Water Reactor
AEC	Atomic Energy Commission
AGN	Active Galactic Nucleus
ALARA	As Low As Reasonably Achievable
ALI	Annual Limit on Intake
ANI	American Nuclear Insurers
ANSI	American National Standards Institute
ATP	Adenosine Tri Phosphate
B	
BEIR	Biological Effects of Ionizing Radiation
BNFL	British Nuclear Fuels Limited
BWR	Boiling Water Reactor
C	
CAM	Continuous Air Monitor
CANDU	Canadian Deuterium Uranium
CAT	Computerized Axial Tomography
CD	Civil Defense
CDC	Centers for Disease Control and Prevention
CDE	Committed Dose Equivalent
CEDE	Committed Effective Dose Equivalent
CERN	European Organization for Nuclear Research
CFR	Code of Federal Regulations
CNS	Central Nervous System
CP-1	Chicago Pile #1
CRCPD	Conference of Radiation Control Program Directors
CRDM	Cracked Reactor Head Nozzles
CRR	Collective Reference Range
CSF	Colony Stimulating Factor
CSI	Criticality Safety Index
CT	Computed Tomography
CW-OSL	Continuous Wave - Optically Stimulated Luminescence
CZT	Zinc activated cadmium telluride
D	
DAC	Derived Air Concentration
DC	Direct Current
DCA	Dicentric Chromosome Aberration
DCGL	Derived Concentration Guideline Level
DDE	Deep Dose Equivalent
DHS	Department of Homeland Security
DNA	Deoxyribonucleic Acid
DOD	Department of Defense
DOE	Department of Energy

Acronym	Definition
DOELAP	Department of Energy Laboratory Accreditation Program
DOP	Dioctyl phthalate
DOT	Department of Transportation
DREF	Dose Rate Effectiveness Factor
DRF	Dose Reduction Factor
DS02	Dosimetry System of 2002
DS86	Dosimetry System of 1986
DTPA	Diethylene triamine pentaacetic acid
DU	Depleted Uranium
DWPF	Defense Waste Processing Facility
E	
EC	Electron capture
EDTA	Ethylene diamine tetraacetic acid
EHBP	Ethane hydroxy biphosphonate
EPA	Environmental Protection Agency
EPD	Electronic Personnel Dosimeter
EPR	Electron paramagnetic resonance
EPRI	Electric Power Research Institute
EPZ	Emergency Planning Zone
ESR	Electron Spin Resonance
F	
FDA	Food and Drug Administration
FIDLER	Field Instrument for Detection of Low Energy Radiation
FRMAC	Federal Radiological Monitoring and Assistance Center
G	
G-CSF	Granulocyte Colony Stimulating Factor
GI	Gastrointestinal
GM-CSF	Granulocyte Macrophage Colony Stimulating Factor
GSD	Genetically Significant Dose
H	
HDER	Homeland Defense Equipment Reuse
HEPA	High Efficiency Particulate
HEU	High Enrichment Uranium
HMR	Hazardous Materials Regulations
HPGe	High Purity Germanium
HPS	Health Physics Society
HRCQ	Highway Route Controlled Quantity
HSA	Historical Site Assessment
HTGR	High Temperature Gas Reactor
HTO	Tritiated water with one tritium atom
HZE	High Z and high energy

Acronym	Definition
I	
IAEA	International Atomic Energy Agency
ICAO	International Civil Aviation Organization
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiation Units and Measurements
IMDG	International Maritime Dangerous Goods
IMO	International Maritime Organization
IND	Improvised Nuclear Device
INEE	Idaho National Engineering and Environmental
INES	International Nuclear and Radiological Event Scale
INPO	Institute of Nuclear Power Operations
IRF	Intake Retention Factor
IRR	Individual Reference Range
ISFSI	Independent Spent Fuel Storage Installation
ISOE	Information System on Exposure
K	
kVp	Peak kilovoltage
L	
LANL	Los Alamos National Laboratory
LBGR	Lower Bound of the Gray Region
LD50/30	Lethal Dose to 50% of the exposed animals in 30 days
LD50/60	Lethal Dose to 50% of the exposed humans in 60 days
LES-URENCO	LA Energy Services - Uranium Enrichment Company
LET	Linear Energy Transfer
LHC	Large Hadron Collider
LLW	Low Level Waste
LOCA	Loss of Coolant Accident
LSA	Low Specific Activity
M	
MARLAP	Multi-Agency Radiological Laboratory Analytical Protocols
MARSSIM	Multi-Agency Radiological Site Survey and Investigation Manual
MCA	Multi-channel Analyzer
MCL	Maximum Contaminant Level
MDA	Minimum Detectable Activity
MDC	Minimum Detectable Concentration
MOSFET	Metal Oxide Semiconductor Field Effect Transistor
MOX	Mixed Oxide
N	
NASA	National Aeronautics and Space Administration
NATC	North American Technical Center
NCRP	National Council on Radiation Protection and Measurements
NEI	National Energy Institute

Acronym	Definition
NEST	Nuclear Emergency Search Team
NIOSH	National Institute of Occupational Safety and Health
NIST	National Institute of Science and Technology
NORM	Naturally Occurring Radioactive Material
NRC	Nuclear Regulatory Commission
NRF	National Response Framework
NRRPT	National Registry of Radiation Protection Technologists
NVLAP	National Voluntary Laboratory Accreditation Program

O

OD	Optical Density
ORNL	Oak Ridge National Laboratory
OSL	Optically Stimulated Luminescence

P

PAG	Protective Action Guide
PAPR	Powered Air Purifying Respirator
PCB	Polychlorinated biphenyl
PI	Performance Indicator
PIC	Pressurized Ion Chamber
PM	Photomultiplier
POSL	Pulsed Optically Stimulated Luminescence
PSE	Planned Special Exposure
PSN	Proper Shipping Name
PTSD	Post Traumatic Stress Disorder
PWSCC	Primary water stress corrosion cracking
PWR	Pressurized Water Reactor

R

RBE	Relative Biological Effectiveness
RBMK	Hi Power Channel Reactor (Reaktor Bolshoy Moshchnosti Kanalniy)
RCRA	Resource Conservation and Recovery Act
RDD	Radiological Dispersion Device
REAC/TS	Radiation Emergency Assistance Center/Training Site
RERT	Radiological Emergency Response Team
rms	Root mean square
RO	Rad Owl
RTR	Research and Test Reactor
RWP	Radiation Work Permit

S

SCO	Surface Contaminated Object
SDD	Superheated Drop Detector
SDP	Significance Determination Process
SI	International System
SIRAD	Self-indicating Instant Radiation Alert Dosimeter
SL-1	Stationary Low Power reactor #1

Acronym	Definition
SLAC	Stanford Linear Accelerator Center
T	
T2O	Tritiated water with two tritium atoms
TEDA	Triethylene diamine
TEDE	Total Effective Dose Equivalent
TENORM	Technologically Enhanced Naturally Occurring Radioactive Material
TEPCO	Tokyo Electric Power Company
TeV	Teraelectron Volt
TI	Transport Index
TL	Thermoluminescence
TLD	Thermoluminescent Dosimeter
TMI	Three Mile Island plant
TODE	Total Organ Dose Equivalent
TRIGA	Training, research, isotopes General Atomic
TRUPACT	Transuranic Package Transporter
U	
UMTRAP	Uranium Mill Tailings Remedial Action Project
UNSCEAR	United Nations Scientific Committee on the Effect of Atomic Radiation
USPS	United States Postal Service
V	
VSLI	Very Large Scale Integration
VVER	Water-Water Energetic Reactor (Vodo-Vodyanoi Energetichesky Reaktor)
W	
WINCO	Westinghouse Idaho Nuclear Company
WIPP	Waste Isolation Pilot Project
WL	Working Level
WLM	Working Level Month
wR	Radiation weighting factor
wT	Tissue weighting factor
Z	
ZEP	Zero Equivalent Point

Answers to Numerical Problems

This section contains just the final answers to the problems in the text which have a numerical solution. As in past editions, the complete step-by-step solutions to all of the problems are contained in the Student Manual which is another component of the Pacific Radiation Self Study Course. (For ordering information, please visit our website, www.pacificrad.com).

Chapter 1

1. 1.00098×10^{5}, 3.45×10^{-3}, and 2.71828×10^{0}
2. 6×10^{9}, 6,000,000,000
3. 12.0
6. t = 1.2
7. x = 3.96
8. 20.085537
9. 31.2 feet high
10. n=-0.063
12. 78 ft-lbs force
13. 28 feet per sec
15. 2.4×10^{-5} amps
16. 72 volts
17. 6.08×10^{8} A
18. 434 ohms
19. 8 times
23. 44.8 liters, 170 grams

Chapter 2

2. 10,000 times smaller
9. Beta = 0.758 MeV, Neutrino = 1.52 MeV
13. 0.329 MeV
14. 1 day
15. 8.3×10^{-4} per minute
16. 6.4×10^{15} dps
17. 0.39% left
18. A) 64 mCi and B) 0.06 μCi

S-2. 1.44 half-lives
S-3. Total = 92.16 MeV and average = 7.68 MeV per nucleon
S-4. 4.868 MeV
S-7. 4.5×10^{18} years

Chapter 3

4. 17 collisions
6. A) Low = 5.2 times, B) Medium = same #, C) High = 1.7 times
8. 1876.5 MeV
9. 2.42×10^{20} /second, 1.24×10^{-10} cm
10. 4.138 x E-15 eV-second
11. 4.5%
12. About 0.6 MeV
13. 33.9 eV/i.p.
16. Reduced 50%
18. P-32: 6.8 m air, 0.86 cm tissue
    Y-90: 9.1 m air, 1.1 cm tiss.
    C-14: 0.6 m air, 0.08 cm tiss.
19. 542 keV

S-2. 10 keV) 1.2% vs. 4% simulated
    25 keV) 74% vs. 60% simulated
S-4. Al = 0.29 cm, acrylic plastic = 0.67 cm

Chapter 4

9. 30 rad
12. 0.5 rem over 9 months
16. 425 rem and about 1,000 rem
19. Males = 20.94%; females = 21.17%

Chapter 5

3. 2.22 x 10^{12} dis/min
4. 27 Ci
9. 0.15 Gy; 0.15 Gy
11. USA: Q = 1, 2, 1. Others: Q = 1, 2.5, 2.
12. 2.15 rem
13. Underestimate by 35%
15. ± 20% accuracy
16. 7.05 x 10^{7} Bq
17. Cs-137 is 13% low; Co-60 is 4% low
20. 5.9 times more
21. A) 334 n/cm^2-sec; B) 147 n/cm^2-sec

S-3. 740 rads
S-5. 0.0056 Sv

Chapter 6

12. 39% as much
16. About 3 mSv
19. 11.4 mR/hr
21. 8.5 MeV
25. 3.8 x 10^{17} fissions

Chapter 7

4. A) 1.39 x 10^{5} electrons; B) 100%; C) 2.2 x 10^{-14} Coul.
9. A) 1.39 x 10^{11} electrons, B) 2.2 x 10^{-8} Coul.
23 . 7.2% resolution
24. 63 keV

S-4. 1.08

Chapter 8

5. A) OD = 3; B) 0.1% transmission

Chapter 9

7. B) About 100 days and about 10 days
14. 3160 μCi; 2394 μCi
17. 1.4 hours
18. CEDE = 5.7 rem
19. CEDE = 6.4 rem
20. Both ratios are 67
21. 227 Bq of Am-241
22. 0.78%

S-6. H = 1.76 mSv

Chapter 10

10. A) 2% of DAC; B) 1 mSv/yr CEDE
11. 3 x 10^{-12} μCi/ml
13. 13.7 days

Chapter 11

5. 6 feet
6. 14 mR; 7 mR
7. 8.9 hours/week
8. 7.2 cm of lead
11. 0.18 mSv/hr
12. 4 mm Al
14. 2.1 m of concrete
16. Every 3 months
19. 12 ml
20. 22,000 dpm/100cm^2
22. A) 1; B) 1
24. 10,000

S-6. 33% less for photons, 25% less for neutrons
S-10. 45 x the ALI

Chapter 12

5. 0.78 cm
8. 1.6 x 10^{9} atoms/cm^2

15. 128 Ci
17. 15 mR/hr @ 5 cm gamma and 298 mrad/hr contact beta
20. 0.067 μCi
21. Not over 6 months
23. 1.4 seconds per probe width
24. B) 400 ± 20 cpm; C) 4 minutes
26. 10,000
27. A) 1000 ± 32; B) 1000 ± 52; C) 1000 ± 63
28. A) 900 ± 33 cpm, 3.7%
    B) 900 ± 66 cpm, 7.3%
31. Bkg, 17 min; sample, 43 min
32. A) 343,000 cpm. B) 3.57×10^4 Bq

S-1. 8,000 films per week
S-2. 829 cpm
S-3. 322 dpm/100 cm^2
S-4. 1400 mrem/hr

Chapter 13

5. 0.4 cubic cm/year
6. B) About 1.6×10^6 Ci/T
7. 287% of 1990
16. 5 times longer
17. 250 bundles per year
18. A) 25 dry casks, B) 3181 square feet

S-2. About 0.013 mrem/hour

Chapter 14

7. A) 10 mile radius
12. 25 rem to the thyroid
16. About 1 death
18. 9.8×10^{-4} grams of U-235
19. 21.5 μR/hour

S-7. 0.288 R/hour or 4% lower than the 7:10 Rule

Chapter 15

10. $47,929
15. 3 ALI
16. 3 years
18. 35 millisieverts

S-1. 24.8% of limit
S-3. <22 mSv current year and < 38 mSv in remainder of 5 years.

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Y

Z

Chart of the Chemical Elements

Z #	Chemical Element	Sym-bol	Density (g/cm3)	Z #	Chemical Element	Sym-bol	Density (g/cm3)
89	Actinium	Ac	10.1	80	Mercury	Hg	13.6
13	Aluminum	Al	2.7	42	Molybdenum	Mo	10.2
95	Americium	Am	11.7	60	Neodymium	Nd	7.1
51	Antimony	Sb	6.6	10	Neon	Ne	0.0009
18	Argon	A	0.0017	93	Neptunium	Np	20.4
33	Arsenic	As	5.7	28	Nickel	Ni	8.9
85	Astatine	At		41	Niobium	Nb	8.6
56	Barium	Ba	3.5	7	Nitrogen	N	0.0012
97	Berkelium	Bk	14.8	102	Nobelium	No	
4	Beryllium	Be	1.8	76	Osmium	Os	22.5
83	Bismuth	Bi	9.8	8	Oxygen	O	0.0014
5	Boron	B	2.3	46	Palladium	Pd	12
35	Bromine	Br	3.1	15	Phosphorus	P	1.8
48	Cadmium	Cd	8.6	78	Platinum	Pt	21.5
20	Calcium	Ca	1.6	94	Plutonium	Pu	19.8
98	Californium	Cf	15.1	84	Polonium	Po	9.2
6	Carbon	C	2.2	19	Potassium	K	0.86
58	Cerium	Ce	6.9	59	Praseodymium	Pr	6.6
55	Cesium	Cs	1.9	61	Promethium	Pm	7.3
17	Chlorine	Cl	0.0032	91	Protactinium	Pa	15.4
24	Chromium	Cr	7.2	88	Radium	Ra	5.0
27	Cobalt	Co	8.9	86	Radon	Rn	0.0097
29	Copper	Cu	8.9	75	Rhenium	Re	20.5
96	Curium	Cm	13.5	45	Rhodium	Rh	12.4
105	Dubnium	Db		37	Rubidium	Rb	1.5
66	Dysprosium	Dy	8.6	44	Ruthenium	Ru	12.2
99	Einsteinium	Es	8.8	104	Rutherfordium	Rf	≅23
68	Erbium	Er	9.2	62	Samarium	Sm	7.7
63	Europium	Eu	5.2	21	Scandium	Sc	2.5
100	Fermium	Fm		106	Seaborgium	Sg	
9	Fluorine	F	0.0017	34	Selenium	Se	4.8
87	Francium	Fr	1.87	14	Silicon	Si	2.4
64	Gadolinium	Gd	8.0	47	Silver	Ag	10.5
31	Gallium	Ga	5.9	11	Sodium	Na	0.97
32	Germanium	Ge	5.4	38	Strontium	Sr	2.6
79	Gold	Au	19.3	16	Sulfur	S	2.1
72	Hafnium	Hf	11.4	73	Tantalum	Ta	16.6
2	Helium	He		43	Technetium	Tc	11.5
67	Holmium	Ho	0.00018	52	Tellurium	Te	6.2
1	Hydrogen	H	10.1	65	Terbium	Tb	8.3
49	Indium	In	8.9E-05	81	Thallium	Tl	11.9
53	Iodine	I	7.3	90	Thorium	Th	11.5
77	Iridium	Ir	4.9	69	Thulium	Tm	9.4
26	Iron	Fe	22.4	50	Tin	Sn	7.3
36	Krypton	Kr	7.86	22	Titanium	Ti	4.5
57	Lanthanum	La	0.0037	74	Tungsten	W	19.3
103	Lawrencium	Lr	6.2	92	Uranium	U	18.7
82	Lead	Pb		23	Vanadium	V	6.0
3	Lithium	Li	11.35	54	Xenon	Xe	0.0058
71	Lutetium	Lu	0.53	70	Ytterbium	Yb	7.0
12	Magnesium	Mg	9.7	39	Yttrium	Y	5.5
25	Manganese	Mn	1.7	30	Zinc	Zn	7.1
101	Mendelevium	Mv	7.4	40	Zirconium	Zr	6.5

BASIC RADIATION PROTECTION TECHNOLOGY

by Daniel Gollnick, Ph.D.

Dr. Gollnick earned his M.S. degree in Experimental Nuclear Physics at Michigan State University and his Ph.D. in Radiological Physics at the University of California, Los Angeles. He is President of Pacific Radiation Corporation, a technical consulting firm he founded to provide assistance, training and D and D services to the nuclear community. He is certified in Comprehensive Health Physics by the American Board of Health Physics. He served as Professor of Physics for 22 years at California State University, Los Angeles before he decided to devote full-time to consulting activities. He has taught radiation protection courses at both the technical and professional levels, at numerous sites around the world. He worked for the U.N. International Atomic Energy Agency as an international expert in Health Physics in Argentina. His current training efforts are focused on nuclear hazmat technology.

Having been employed for over 50 years in the radiation field, Dr. Gollnick recently decided to slow down a bit. He hung up his racecar driving suit and expends his spare energy in astronomy outreach programs. He presents a daytime show-and-tell program for grade school students in their classrooms. At night, he sets up his telescopes on the school playground and gives the school kids and their parents some quality eyes-on experience viewing the wonders of the night sky.

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436 Problems

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