Instrumentation for Environmental Monitoring RADIATION

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ABSTRACT

A comprehensive survey of instrumentation for environmental monitoring is being carried out by the Lawrence Berkeley Laboratory originally under a grant from the National Science Foundation and now by the Office of Health and Environmental Research of the Department of Energy. Instruments being investigated are those useful for measurements in Air Quality, Water Quality, Radiation, and Biomedicine related to environmental research and monitoring. Consideration is given to instruments and techniques presently in use and to those developed for other purposes but having possible applications to this work. The results of the survey are given as (a) descriptions of the physical and operating characteristics of available instruments, (b) critical comparisons among instrumentation methods, and (c) recommendations of promising methodology and development of new instrumentation. Information is also given regarding the pollutants to be monitored: their characteristics and forms, their sources and pathways, their effects on the ecosystem, and the means of controlling them through process and regulatory controls.

The survey material is compiled in seven loose-leaf binders which can be periodically updated.

<u>Volume</u>	<u>Parts</u>	<u>Category</u>
1	1 and 1A	Air Monitoring, Gases
1	2	Air Monitoring, Particulates
2 and 2A		Water Monitoring
3		Radiation Monitoring
4		Biomedical Monitoring

INDEXING METHOD

The sections of the survey are indexed by a series of mnemonic descriptors. The caption in the upper right-hand corner of each page provides the necessary filing information. On the first line, a three-character descriptor indicates the category of monitoring:

<u>Category</u>	<u>Mnemonic descriptor</u>
Air Monitoring	AIR
Water Monitoring	H2O
Radiation Monitoring	RAD
Biomedical Monitoring	BIO

Here, also, a second descriptor indicates the type of parameter under discussion. For example, alpha radiation monitoring instruments are identified by "RAD-ALP." A list of these descriptors is given in the contents on the right side in capitals.

On the second line, a third descriptor is used to identify the specialized topic, type of sampling or use; for example, "AIR-SO₂, Ambient" signifies instruments suitable for monitoring sulfur dioxide in the ambient atmosphere.

On the third line, for instrument notes, the manufacturer or developing laboratory is indicated as the fourth descriptor. Additional instrument notes in this category, parameter, and use by this manufacturer, are indicated by numerals after the manufacturer's name.

The last descriptor is the date the note was first issued. Revised notes bear a new date. Descriptions of equipment bearing a new model number will be issued as additional notes.

PROLOGUE

"I often say that when you can measure what you are speaking about and express it in numbers you know something about it, but when you cannot measure it, when you cannot express it in numbers, your knowledge is of a meager and unsatisfactory kind...it may be the beginning of knowledge, but you have scarcely, in your thoughts, advanced to the stage of science whatever the matter may be."

Lord Kelvin, 1883

The wilderness and the parched land shall be glad; and the desert shall rejoice, and blossom as the rose. It shall blossom abundantly, and rejoice, even with joy and singing;....And the parched land shall become a pool, and the thirsty ground springs of water;....

Isaiah 35: 1,2,7

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PREFACE

It is essential to the nation's program of environmental improvement that accurate and efficient measurements be made of the parameters affecting the environment in which we live and work. In many important areas the present state of the art of instrumentation for environmental monitoring is inadequate. There are many reasons for this inadequacy; more relevant, however, is the question: Where are we now and where do we go from here? The Lawrence Berkeley Laboratory, under grants from the National Science Foundation and the Department of Energy, is conducting a survey of instrumentation suitable for the measurement of the quality of our environment. The results of the survey appear in four volumes which discuss monitoring of air, water, and radiation, and treat biomedical problems due to environmental causes.

Air Monitoring, Volume 1, primarily considers the chemical parameters known to contribute (or suspected of contributing) to environmental degradation. Physical parameters, usually recorded to establish standard conditions of measurement, are also included.

Water Monitoring, Volume 2, deals with instrumentation suitable for fresh water and estuarine water; coastal and off-shore oceanic waters have been excluded from the survey. Chemical and biological contaminants have been included as well as physical parameters.

Radiation Monitoring, Volume 3, considers ionizing radiation. A description of the type of radiation present at various sources of radiation is followed by sections containing a discussion and the specifications for the instrumentation used to monitor its radiation.

Biomedical Instrumentation, Volume 4, deals with those parameters which affect human, animal and plant life presently considered to arise from environmental problems. Hospital instru-

mentation is not included. Biomedical problems arising from contamination of the air or water are discussed here, e.g., in the mercury section an overview describes the sources, metabolism, food chain and genetic effects, followed by a description of the methods for detecting mercury in biological samples.

A number of environmental contaminants are well identified, others are at present only suspect, still others are potentially insulting to the ecological balance, and the effects of some substances are unknown. Upon the advice of recognized authorities, the effort expended in compiling this survey has been distributed in proportion to the severity of the problem. If problems are well identified and existing instrumentation is considered satisfactory, presently available instruments are emphasized. If current methods of detection are inadequate--because, for example, of insensitivity, the time consumed in making the measurement or the ambiguity resulting from interfering parameters--present methodology is surveyed and new methods of detection and analysis are explored. Recommendations are made for development and exploitation of techniques that are employed in other disciplines and appear to be applicable in the environmental field.

Each volume contains (a) an overview of the basic problems, (b) comparisons among the basic methods of sensing and detection, and (c) instrument notes that summarize the characteristics and evaluate the presently available instruments and techniques. Steps that should be taken to alleviate the problems are recommended if appropriate.

Contact has been made with the staffs of federal, state, university, regional and local laboratories as well as with manufacturers interested in instrumentation for environmental monitoring. Due to the nation-wide concern

in this field, it is obvious that not all the experienced professionals in any given specialty could be contacted; however, as the survey continues, an endeavor will be made to utilize all the experience and expertise that comes to the attention of the group.

This survey deals predominantly with instrumentation which is available commercially from U.S. sources. The information is derived from manufacturers' catalogs and specifications, the periodic literature, and scientific and engineering reports. Instrumentation from outside the U.S. has been included whenever it has been found to be widely used or merits attention by our readers because of some particular characteristics.

The survey has been compiled from the aforementioned information sources. The limited scope of this program, however, has not allowed the in-house evaluation of operating instruments at this laboratory.

It is realized that no survey can be exhaustive and complete. If any existing or potentially useful instrument or technique has been omitted, it is urged that information about it be com-

municated to the Environmental Instrumentation Survey Group at once.

It is important to state that the explicit function of this survey is to aid in, but not to be the sole information relied upon in, the choice of instrumentation. Manufacturers' information and the periodic literature in particular must be considered by any potential user.

The technical information in this survey has been compiled from the best available sources and is believed to be correct. The mention of commercial products, their source or their use in connection with material reported herein is not to be construed as either an actual or implied endorsement of such products. The opinions expressed herein represent the opinions of the LBL Environmental Instrumentation Group and do not necessarily express the opinion of the National Science Foundation, the Department of Energy or the University of California.

Mary S. Quinby Hunt
Environmental Instrumentation Survey
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I.A. Radiation, Monitoring Programs, and Instrumentation

1. Sources of Radiation
2. Objectives of Environmental Monitoring Programs
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A. Radiation, Monitoring Programs, and
Instrumentation

Man has been continuously exposed to natural radiation since his appearance on earth, but has been aware of the presence of penetrating ionizing radiation only for the last eighty years. During this time, the use of radiation and radioactive materials, often man-made, has steadily increased, but a substantial body of evidence has accumulated that radiation also poses risks. For this reason, a major effort has been made to understand the attendant risks, to formulate standards for limiting radiation exposures, and also to develop means to monitor radiation and radionuclides in the environment.

This section indicates briefly the important sources of radiation, the objectives of monitoring programs, and some basic monitoring considerations. It also summarizes the organization of this volume. The following introductory sections define units (I.B), summarize the biological effects of radiation (I.C), discuss radiation standards and their origin (I.D), introduce radiation detection methods (I.E), and outline several general analytical considerations (I.F).

1. Sources of Radiation

Man's environment now has two sources of radiation, natural and man-made. Natural sources are the greater contributor to human radiation exposure (Refs. 1-5). Section II.F

discusses this contribution, which averages about 100 mrem/year per capita (Section I.B discusses units).

The important man-made causes of radiation exposures are the following:

- Diagnostic and therapeutic procedures (x-rays, radionuclides);
- Increased exposure to natural background due to man's activities (high-altitude flights, use of naturally radioactive substances in construction, etc.), commonly referred to as technologically enhanced;
- Nuclear weapons (fallout of radionuclides from weapons testing; direct exposure to a nuclear event);
- Nuclear fuel: mining, processing, and re-processing operations (internal radionuclide exposure to workers; release of gaseous, liquid and solid radiation during processing, transportation and storage);
- Nuclear reactors used for either power or research (release of radiation during normal operation; accidental releases);
- Particle accelerators (whole-body external exposure);
- Industrial use of radionuclides and x-rays for processing or quality control;
- Research instruments, such as x-ray diffractometers and photoelectric spectrometers.

Table 1. Summary of average annual exposures from various man made sources.
(Reproduced from LBL-5285, Ref. 2.)

Man Made Sources	Individual Annual Dose	Average Annual Dose
Weapons test fallout ^b	Variable	4 mrem
Medical exposures ^c	Highly variable	60 mrem
Nuclear power ^b		0.003 mrem
Miscellaneous (consumer products, jet travel) ^b	Highly variable	2 mrem
Occupational (mostly medical) exposures	Highly variable	0.8 mrem
TOTAL		70 mrem

^bFrom BEIR (Ref. 4), whole body exposures.

^cFrom NCRP 39 (Ref. 3) and BEIR (Ref. 4); number given is for "abdominal dose," roughly corresponding to genetically significant dose.

Table 1 gives typical doses from human activities; only medical procedures contribute average doses comparable to that from natural background radiation. In the past few decades, substantial effort has gone into the study of these sources to understand, isolate, control and reduce exposure to as many of them as possible. The problem of monitoring radiation levels is different in each case, and special techniques have been developed to achieve an understanding of the exposure levels involved. One aim of this survey is to gather together these techniques, categorized both by the radiation source and by the type of instrument.

First, each of the major sources is discussed separately, pointing out the type of radiation that needs to be measured. For example, the problem of monitoring nuclear reactor emissions is discussed in Section II.B. Once the radiation types (e.g., alphas, betas, x-rays, neutrons, and/or gammas) or radionuclides (e.g., tritium, plutonium, etc.) to be measured have been determined, the user can refer to Division III or IV for information on the various instruments available for the task as discussed in Section I.A.4 on instrument notes and discussion.

2. Objectives of Environmental Monitoring Programs

The radioactive materials man uses must be adequately contained for the protection of humans and the broader biosphere. Environmental monitoring (including that performed at the source) indicates the effectiveness of controls on radioactive releases and, where an abnormality occurs, the character of the failure and of the release.

IAEA Safety Series No. 41, a publication of the International Atomic Energy Agency (see Section I.C), states the primary objectives of environmental monitoring near installations using radioactive materials for normal situations as: (Ref. 6)

"a) Assessment of the adequacy of controls on the release of radioactive materials to the environment.

"b) Assessment of the actual or potential exposure to man to radiation or to radioactive materials present in his environment or, as a minimum, the estimation of the probable upper limits of such exposure.

"c) Demonstration of compliance with the applicable regulations, environmental standards, and other operational limits.

"d) The possible detection of any long-term changes or trends in the environment resulting from the operation of the installations."

Benefits received from such environmental surveys may include:

"a) An increase in knowledge concerning the correlation of levels of discharge and the environmental effect, thus improving the basis for future predictions and for the estimation of maximum or other levels that might arise in an emergency.

"b) The provision of data that may be used to provide the public with adequate information on environmental surveillance.

"c) The maintenance of a capability that will go at least some way to provide a means of dealing with a foreseeable emergency situation.

"d) The provision of data related to the behavior of elements in the environment, including the ecological, hydrological, geochemical, and meteorological aspects of such behavior."

The objectives of a survey under emergency situations differ from those for normal monitoring. The primary objectives of emergency surveys are:

"a) The rapid compilation of information, on a timely basis, on the magnitude and location of the possible hazards to the public for the purpose of defining the type and extent of any necessary countermeasures or other emergency procedures.

"b) The assessment of any hazard from inhalation or external radiation to serve as a basis for the planning of immediate countermeasures.

"c) The rapid determination of the possible contamination of food-stuffs, including milk and drinking water, as a basis for decisions on rejection or continued use.

"d) The provision of data necessary to assess the radiation doses actually received by members of the public, taking into account any countermeasures that have been applied.

"e) The provision of timely information for the public regarding the emergency situation.

"f) The collection of scientific information on the results of the emergency and on the behavior of the released radioactive material in the environment. Such information may be useful in checking the adequacy of the monitoring and accident mitigation systems and to determine whether any changes are required in the routine monitoring programs."

The most prominent example of a broad environmental monitoring program is that

established initially to measure fallout from nuclear weapons testing (Ref. 9). As a direct outgrowth of a program initiated by the U. S. Naval Research Laboratory in 1954, the DOE Environmental Measurements Laboratory (EML) [formerly the Health and Safety Laboratory (HASL)] since January 1963 has been conducting the Surface Air Sampling Program. The primary objective of this Department of Energy (DOE) program is to study the spatial and temporal distribution of specific natural and man-made radionuclides, and of trace metals in the surface air. The raw data are summarized quarterly in Environmental Quarterly published by EML. The Department of Energy also publishes the environmental monitoring reports from its contractors (Ref. 11). The U.S. Environmental Protection Agency (EPA) operates the Environmental Radiation Ambient Monitoring Systems (FRAMS) through its Office of Radiation Programs (Ref. 10). This is a surveillance program for measuring levels of radioactivity in air, air particulates, deposition, surface and drinking water, and milk in the United States and its territories. A tabulation of all raw data from each sampling network is reported quarterly in Environmental Radiation Data by the Eastern Environmental Radiation Facility. Other federal agencies conduct smaller and more specific programs in the radiation areas of concern to them (e.g., monitoring of mines by the Mine Health and Safety Administration). Moreover, agencies of many states conduct broad surveys of radiation and radionuclide levels, often with efforts concentrated around identifiable sources of radioactivity.

Such sources are discussed in Division II. However, we note that, in connection with commercial nuclear power operations, the EPA has issued an "Environmental Radioactivity Surveillance Guide" (Ref. 7) on surveillance outside light-water-cooled nuclear power plants. In addition, the U.S. Energy Research and Development Administration (ERDA), now part of the U.S. Department of Energy (DOE), issued a guide (Ref. 8) for environmental surveillance at its facilities. Comparable guides have been formulated by various entities for the specific sources of radiation and radioactivity, as indicated in the sections on these sources. In addition to "environmental monitoring", we discuss monitoring at the source. In any surveillance program the measurements required are usually low-level determinations which depend on reliable techniques of data gathering and analysis, supported by parallel theoretical computation (Ref. 12).

3. Radiation Measurements and Dose Assessment

Monitoring at a source or in the environment can have two basically different, but closely related, goals: either to characterize the radionuclide or radiation distribution, or

to characterize the resulting burdens or doses to which tissue would be exposed. The precise distinctions are suggested to some extent by the definitions given in Section I.B ("Units of Radiation and Radioactivity") and I.F ("General Analytical Considerations"). The fundamental purpose of a radiation protection program is to limit radiation doses, but the monitoring employed to indicate the effectiveness of this program may focus on the use, production, or emission of radioactivity, on the radiation fields resulting from these materials, or on the doses from this radiation.

As a practical matter, a distinction is often made on the basis of the exposure mode. Bodily tissue may be exposed to radiation from either external radioactivity, i.e., outside the body, or from internal sources, inside the body. The instrumentation used to determine radiation fields and absorbed doses depends on the mode of exposure.

In the case of external radiation, there are two generally accepted approaches:

- An area may be monitored to determine radionuclide activity concentrations or alternatively, to characterize the radiation field in terms of spatial distribution, particle flux, energy and angular properties, and intensities.

- The dose of radiation received by an individual can be determined by personnel dosimetry. The aim is either to determine the integrated dose-equivalent (see Section I.B) received over a period of time (e.g., the use of film badges), or alternatively, to determine the instantaneous rate at which the individual is receiving radiation, and other indications such as type and quality of the radiation to which the individual is exposed. These aims differ since in one case a post-hoc measurement is made of exposure already received, while in the other case the information is used as a measure of exposure in the present or immediate future.

In the case of internal radiation exposure, the determination of dose-equivalent can take either or both of the following approaches:

- Analysis of the radioisotope concentration in water, foodstuffs, or respirable air can be compared to maximum permissible concentration (MPC) levels; and

- Whole-body counting, specific organ activity determinations (for example, thyroid activity after ingestion of radioiodine) or excrement analyzers may be used for a post-hoc measurement of exposure levels.

It is the intention of this instrumentation survey to deal with each of the various methods of measuring radiation levels, namely

measurement of radionuclide distributions or radiation fields, personnel dosimetry, radio-isotope analyses of air, food, water, and whole-body or specific-organ activity analysis. An introduction to basic radiation detection methods is given in Section I.E. The ultimate purpose, in any event, is dose assessment.

4. Organization of Instrument Notes and Discussion

In Divisions III and IV, we present information on instruments that are commercially available for radiation monitoring. Each section contains both instrument notes, giving specifications of individual instruments, and discussion, giving background on the monitoring problem of interest.

Instrument notes have been grouped according to the radiation type being measured (Division III) and by radionuclide of interest (Division IV) to facilitate selecting the best instrument for the task. If an instrument performs more than one function, that fact is pointed out in the comparison or remarks section. The obvious comparisons of power, size, weight, cost, etc., can easily be made by the reader in the light of his particular application. In the discussions preceding the instrument notes, we concentrate on comparing the basic principles of operation of the different classes of instruments, and evaluating the adequacy (or inadequacy) of available instrumentation. If appropriate, recommendations for possible improvements, research and development, or conception of new instrumentation and techniques are provided.

At the beginning of the instrument notes for Division III and IV, green sheets are included, which tabulate all the available instrumentation listed in the notes, compiled according to function or type of instruments. These green sheets also list some of the more important parameters for each device. These

green sheets can be a useful starting point for persons interested in an instrument to perform a specific operation. After finding the instrument the reader needs, he may then look up according to the key mnemonics the specific detailed note on that instrument.

Each instrument within a section will be identified by its primary application or "class". Light-weight, portable, hand-held, rugged, field-survey monitors designed for use by semi-skilled personnel are to be distinguished from heavy, sophisticated calibration instruments for use by highly trained scientists.

Each instrument note also contains a brief description of the "Principle of Operation". A general discussion of the various detection methods has been provided in Section I.E. In the specific instrumentation sections a general familiarity with the physical, chemical, and engineering concepts which form the basis for their operation is assumed, and with the basic analytical chemistry techniques which underly the separation methods for specific radio-isotopes. These considerations are discussed whenever they are relevant to the performance of an instrument in actual monitoring situations. For more detailed discussions of these items, the interested reader is referred to the extensive bibliographies which accompany the various sections.

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I.B. Units of Radiation and Radioactivity

1. Absorbed Dose - The Rad
2. Exposure - The Roentgen
3. Rad - Roentgen Relationship
4. Activity - The Curie
5. Neutral Particles - The Kerma
6. Relative Biological Effectiveness (RBE)
and Quality Factor (Q)
7. Dose Equivalent - The Rem
8. Acknowledgments
9. References

B. Units of Radiation and Radioactivity

Radiation protection requires the development of quantitative concepts and their application to a complex biophysical process. In the initial period of radiation research, much confusion existed about the correct ways of quantifying radiological doses. Today, an accepted set of units exists: the various quantities are defined by the International Commission on Radiation Units (ICRU) in its Report 19, "Radiation Quantities and Units" (Ref. 1). Because radiation and its interaction with matter is at the microscopic level a random process, certain radiation quantities can only be adequately described in statistical terms. This mode of description is analyzed in some detail in the above report and in subsequent reports (Refs. 2, 3).

1. Absorbed Dose - The Rad

The basic concept in radiation dosimetry is the absorption of energy in living tissue. The basic unit of absorbed dose is called the "rad" (radiation-absorbed-dose) defined by the ICRU¹ as:

1 rad = 100 ergs of absorbed energy per gram of tissue (0.01 Joules per kilogram of tissue).

2. Exposure - The Roentgen

In the early days a need was felt for quantifying the effect of X- and gamma-radiation on tissue. However, various tissues absorb such radiation at different rates, so the convention arose of defining x-ray fields by the amount of ionization produced in a unit mass of standard air. Exposure is now defined by the ICRU as:

the quotient of ΔQ by Δm where ΔQ is the absolute value of the total charge of the ions of either sign produced in the air when all the electrons (negatrons and positrons) liberated by photons in a volume of air having the mass Δm are completely stopped in air.

The unit of exposure is the roentgen (R) defined as:

$$1 \text{ R} = 2.58 \times 10^{-4} \text{ coulombs per kilogram of air.}$$

Using the facts that the average energy required to produce one ion pair in air is $\sim 34 \text{ eV}$ ($5.5 \times 10^{-11} \text{ ergs}$), and that each charge in the pair is $1.6 \times 10^{-19} \text{ coulomb}$, one finds that that:

$$1 \text{ R} = 87.6 \text{ ergs per gram of air.}$$

*Sometimes the reference numbers will appear as superscripts.

3. Rad - Roentgen Relationship

The roentgen is only defined for photons (x-rays and gammas), and in fact is not used widely except below about 3 MeV. On the other hand, the rad, which measures energy absorbed in tissue, is applicable to all kinds of radiation. The question arises: When tissue is exposed to one roentgen, how much energy is absorbed? Measurements in various tissues (see Fig. 1) indicate that in the 1-MeV region one roentgen deposits 85-95 ergs/g in fatty or muscular tissue and about 75 ergs/g in bone. This is so close to 100 ergs/g, corresponding to 1 rad, that an exposure to 1 roentgen has often incorrectly been considered equivalent to an absorbed dose of 1 rad. Figure 1, taken from Cember (Ref. 4), indicates how significant the error becomes when bone, for example, is exposed to x-rays in the range of 10-50 keV.

Fig. 1. Energy absorption per roentgen for various tissues (from H. Cember, Ref. 4, with permission by Pergamon Press © 1969).

4. Activity - The Curie

A useful quantity in many contexts, including the specifications of radionuclide concentrations, is "activity". The activity of a quantity of material is defined as the number of spontaneous nuclear transformations occurring in this quantity per unit time. The unit of activity is the curie (Ci), where

$$1 \text{ Ci} = 3.7 \times 10^{10} \text{ transformations per second.}$$

In some contexts, the specific activity (i.e., activity per unit mass) is also useful. Thus, one gram of Cesium-134 (half-life = 2.06 years) has fifteen times the decay rate of one gram of Cesium-137 (half-life = 30.2 years). Cesium-134's specific activity is 15 times greater.

5. Neutral Particles: The Kerma

Energy loss (in the form of ionizations) does not occur directly from the passage of neutral particles through matter. However, neutral particles can produce local ionization through indirect means* (such as e^+/e^- pair production from high-energy γ -rays, or nuclear recoil from neutron collisions). The concept used to quantify neutral-particle effects in matter is the kerma. The following definition comes from ICRU Report 19:

The kerma, K , is the quotient dE_{+V}/dm where dE_{+V} is the sum of the initial kinetic energies of all the charged particles liberated by indirectly ionizing particles in a volume element of the specified material (erg/cm^3) and dm is the mass of the matter in that volume element (g/cm^3)

$$K = dE_{+V}/dm.$$

The unit of kerma is the rad ($1 \text{ rad} = 10^{-2} \text{ erg/g}$).

As defined, the incident neutral particles can impinge upon the mass of interest from all directions. The definition also includes within dE_{+V} the energy of all charged particles (such as Auger electrons) produced in secondary processes within the volume element. The kerma requires a specification of the type of matter being irradiated, and for example one may speak of "the kerma rate of 100 keV x-rays in fatty tissue".

6. Relative Biological Effectiveness (RBE) and Quality Factor (Q)

The absorption of a given amount of energy in a given mass of tissue (ergs/g) is not the only criterion for determining the dose-effectiveness of a given radiation field. As discussed in Section I.C, damage generally increases as the rate of linear energy transfer (LET) increases. This is generally attributed to the fact that as energy deposition becomes more and more dense and more localized, the probability of local damage increases. For

purposes of assessing the Relative Biological Effectiveness (RBE) of various radiations, a reference has been chosen as the effectiveness of 200 keV x-rays for creating the observed effect. For dose-deposition purposes, each different type of radiation is assigned a quality factor (Q^{**}) based upon the RBE. The relation of LET to Q is shown in Table 1, from Ref. 1. Table 2, based on Ref. 6, shows a summary of Q numbers of various typical radiations. More detailed numbers are available which treat the energy-dependence of the Q for different particles. The Q as a function of neutron energy is shown in Table 3, taken from Ref. 5 (NCRP Report No. 39.)

Table 1. Relationship Between Quality Factor and Linear Energy Transfer (from ICRU, Ref. 5, with permission by International Commission on Radiation Units and Measurements © 1971).

LET keV per micron in water	Q
3.5 (and less)	1
7.0	2
23	5
53	10
175 (and above)	20

Table 2. Quality Factor Values for Various Radiations (adapted from ICRU, Ref. 6).

Radiation	Q
Gamma-rays	1
X-rays	1
Beta-rays and electrons	1
Neutrons	10
Protons	10
Alpha particles	20
Heavy ions	20

7. Dose Equivalent - The Rem

Since the biological effectiveness of a given absorbed dose depends on the factors above, the absorbed dose is insufficient by itself to predict either the severity or the

* See Section I.E. for more details.

**Previously recommended abbreviation was QF, now it is Q. (Refs. 1, 2).

Table 3. Quality Factor Values for Various Neutron Energies (from NCRP, Ref. 5, with permission by National Council on Radiation Protection and Measurements © 1971).

Neutron Energy, MeV	Q
2.5×10^{-8} (thermal)	2
1×10^{-7}	2
1×10^{-6}	2
1×10^{-5}	2
1×10^{-4}	2
1×10^{-3}	2
1×10^{-2}	2.5
1×10^{-1}	7.5
5×10^{-1}	11
1	11
2.5	9
5	8
7	7
10	6.5
14	7.5
20	8
40	7
60	5.5
1×10^2	4
2×10^2	3.5
3×10^2	3.5
4×10^2	3.5

probability of the deleterious effects on health resulting from irradiation under unspecified conditions. In current radiation protection procedures, an indication of the effect upon a given organ is inferred by weighting the absorbed dose in that organ by certain modifying factors. This quantity correlates better with the more important deleterious effects of exposure to radiation particularly with the delayed stochastic effects (those for which the probability of an effect occurring, rather than its severity is re-

garded as a function of dose, without threshold.) The permissible levels for radiation protection are stated in terms of dose equivalent.

The dose equivalent (H^*) is the product $D \cdot xQ \cdot N$ at the point of interest in tissue, where D is the absorbed dose, Q is the quality factor and N is the product of any other modifying factors (now assigned a value of 1 by the ICRP).

The most frequently used unit of dose equivalent is the "rem" equal to 100 erg/g; thus, H is in rem if D is expressed in rads. However, the Standard International unit of dose equivalent is the Sievert[†] (Sv); 1 Sv = 1 J/kg (= 100 rem).

However, it should be noted that: a) dose equivalent is limited to radiation protection applications and may be used when H is in the region of, or below, the applicable maximum permissible dose equivalent (it should not be used for high level accidental exposures); b) the dependence of Q upon LET is specified by the International Commission on Radiological Protection⁶ and shown in Table 1; c) in the usual case where D is delivered by particles having a range of LET values, the dose equivalent is the total dose weighted by an average quality factor, $H = DQN$.

(Two new terms introduced by ICRU Report No. 19, Absorbed Dose Index and Dose Equivalent Index, are not germane to our discussion.)

The rem is applicable to all types of radiation and for both external and internal exposures. It is crucial to recognize, however, that to determine rem properly requires a knowledge of the absorbed dose (ergs/g), the composition of the radiation, and its energy distribution.

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* Formerly designated DE.

† However we will use rem throughout this text.

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- I.C. Biological Effects of Ionizing Radiation
 - 1. Physical and Biological Considerations
 - 2. Radiation Disease Effects
 - a. Somatic effects
 - b. Genetic effects
 - 3. Threshold and Dose-Response
 - 4. Acknowledgments
 - 5. References

C. Biological Effects of Ionizing Radiation

As a basis for setting limits on radiation exposures, quantitative concepts have been developed, as noted in Section I.B, for expressing radiation doses and their effects on living organisms. This section will discuss some of the biological factors that have affected the guideline setting process. Two standing committees review the biological effects of ionizing radiation, the United Nations Scientific Committee on Effects of Atomic Radiation (UNSCEAR) and the National Academy of Sciences/National Research Council Advisory Committee on Biological Effects of Ionizing Radiation (BEIR). The discussion of this section is based on reports of these committees,* as well as reports of the International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection and Measurements (NCRP). Additional information is developed through activities of the International Atomic Energy Agency and the World Health Organization (Ref. 3).

The only radiation effects discussed here are those in man, and related animal and cellular studies. The view of regulatory agencies, including the ICRP (Ref. 4) and NCRP, is that effects on forests, (Ref. 5), birdlife, (Ref. 6) and other parts of the biosphere (Ref. 7,8) are adequately controlled when man is given proper protection.

1. Physical and Biological Consideration

Ionizing radiation carries enough energy to ionize atoms and such ionization (or excitation) can damage cells. This radiation may arise from nuclear decay (e.g., alpha, beta, gamma or neutrons) or certain atomic processes (X-rays) occurring in radioactive materials, high voltage devices, accelerators, and reactors. Table 1 summarizes some important characteristics of radiation types.

As discussed in Section I.B, the concept that has arisen to measure ionizing radiation is absorbed dose (D), the energy deposited per unit mass, measured in rads. However, experiments have shown that one rad of absorbed dose does not always produce the same biological effect in all tissues. The concept of Relative Biological Effectiveness has been introduced to express the relative amounts of different radiation that produce the same amount of biological damage. (Usually x-radiation near 200 keV is used as the reference for specifying RBE). The radiation protection guides take RBE into account by establishing a Quality Factor (Q) that multiplies absorbed dose to yield dose equivalent (H). Considerable evidence indicates the importance of the instantaneous ionization of a charged particle per unit length of matter traversed, which is termed linear energy transfer (LET) (Ref. 9). Biological effects are

also found to depend on the rate at which the absorbed dose is delivered.

When a charged particle or other ionizing radiation passes through matter, energy is lost by the particle along its path, primarily by knocking electrons from atoms and molecules, causing ion pairs. The energy loss per distance travelled is called the linear energy transfer (LET), usually expressed in kiloelectronvolts (keV) per micron (μm). Since tissue is mostly water, water is the relevant medium for comparing the LET of different radiation. Densely ionizing particles have large LET values. The value of the LET depends not only on the particle but on its energy, as illustrated in Table 2. It is found that, in general, the relative biological effectiveness increases as LET increases. This was illustrated in Table 1 of Section I.B, in which the Quality Factor (Q) increased with LET. For further information on LET and biological effects the reader should be aware that the NCRP is currently studying this problem and may comment on it in the near future (Ref. 11).

The dependence of RBE on the time distribution of absorbed dose (the dose rate) has been demonstrated in a number of experiments and will be discussed after summarizing the health effects of radiation. Before going on, it is useful to note that considerations such as RBE and, more generally, radiosensitivity depend on both species and individuals:

Species Variance: Plants have usually shown a greater resistance to radiation damage than mammal life, although there are exceptions (such as pine trees) (Ref. 12). Moreover, different animal species show different sensitivities, causing difficulty in designing experiments and interpreting the results. However, at least for common experimental animals (mice, dogs, pigs), a consistent method of scaling the experimental results to man has been agreed upon (Ref. 13).

Individual Variance: Radiosensitivities have also been found, in some cases, to depend on sex [e.g., in mice (Ref. 14)], age (esp., young children vs. adults), and other population subgroups (Ref. 15).

2. Radiation Disease Effects

As radiation passes through tissue, ionization and the resulting formation of free radicals occurs very quickly, i.e., within microseconds. Subsequent chemical reactions then occur within a few seconds and may result in an early death of the cell, the prevention or delay of cell division, or a permanent modification which is passed on to daughter cells. The symptoms resulting from biological effects of radiation exposure manifest themselves in times from tens of minutes to tens of years de-

*Sometimes reference numbers appear as superscripts.

Table 1. Basic Types of Radiation and Its Instrumentation

Radiation	Symbol	Sources and Characteristics	Instrumentation
alpha particles	He^{++}, α	Highly ionizing, positively charged particles emitted by many naturally occurring radionuclides. Discrete energies. Short range. Atomic mass 4.	Ionization chambers, proportional counters, ZnS Scintillators; semiconductors
beta particles	e^-, e^+, β^-	Electrons/positrons emitted by all fission products and most thermal neutron activation products. Continuous distribution of energies from zero to characteristic maximum values. More penetrating than α -particles (for energies > 0.1 MeV). Mass $\sim (1/1850)1$.	Gas counters in Geiger-Müller and proportional voltage regions. Solid and liquid scintillation counters. Photographic film. Semi-conductor
gamma and X-rays	γ X	Relatively penetrating electromagnetic radiation. Interacts with matter via the photoelectric effect, pair production or Compton scattering, exhibiting characteristic peak energies in the first two instances. Absorption follows Beer's Law: $I = I_0 e^{-\mu x}$ Where I_0 = initial intensity I = intensity after traversing absorber of thickness $x(\text{cm})$ μ = attenuation coefficient (absorption coefficient) of the absorber	photon spectrometers Large crystal scintillators (eg NaI(Tl); > 0.1 MeV) Thin crystal scintillators Ge(Li) Detectors Proportional counters (Xenon for ~ 0.01 MeV X-rays) TLD
Neutron	n	Indirectly ionizing, neutral particle emitted in fission	BF_3 gas filled detector ^6Li loaded TLD pn junction diode Activation techniques

Table 2. Ionizing Particles of Radiobiological Interest (from D. J. Rees, *Health Physics*, First edition, 1967. Courtesy of Charles C. Thomas, Publisher, Springfield, Illinois.)

Particle	Charge	Energy (MeV)	LET (keV/ μ)
Electron	-1	0.001	12.3
		0.010	2.30
		0.100	0.42
		1.00	0.25
		200 k Vp x-rays	0.4-36
		Cobalt 60 -rays	0.2-2
Proton	+1	Small	92
		2	16
		5	8
		10	4
Alpha	+2	Small	260
		5	95

pending on the particular type of cell damage.

These effects may be divided into two classes (Ref. 16) somatic and genetic. The somatic effects arise from damage to the ordinary cells of the body and affect only the irradiated person. The genetic effects, on

the other hand, are due to damage to cells in the reproductive organs, the gonads, so that the damage may be passed on to the person's children and even to later generations.

(a) Somatic Effects

i. The Acute Radiation Syndrome. The characteristic group of symptoms shown by a person whose whole body, or a major portion of it, is exposed to a large dose of radiation in a short time is called the acute radiation syndrome. These symptoms have been studied in animal experiments and, more importantly, in cases of radiation administered to persons in hospitals, in the atomic bomb survivors at Hiroshima and Nagasaki, and in persons exposed accidentally to large doses of radiation. The effects of an exposure large enough to cause an acute radiation syndrome in an individual varies considerably from person to person. The general features of each exposure are listed in Table 3. As the exposure increases, the severity of the effects increases; a single dose of 600 rem to the whole body is likely to result in death.

When large doses of radiation are received in a relatively short time, the first symptoms observed are due to depletion of those cells

Table 3. Probable Effects of Acute Whole-Body Radiation Doses (from *Nuclear Reactor Engineering*, by S. Glasstone and A. Sesonske © 1963 by Litton Educational Publishing, Inc. Reprinted by permission of D. Van Nostrand Company.)

Acute dose (rem)	Probable Clinical Effect
0 to 25	No observable effects.
25 to 100	Slight blood changes but no other observable effects.
100 to 200	Vomiting in 5 to 50 percent within 3 hrs with fatigue and loss of appetite. Moderate blood changes. Except for the blood-forming system, recovery will occur in all cases within a few weeks.
200 to 600	For doses of 300 rem and more, all exposed individuals will exhibit vomiting within 2 hrs or less. Severe blood changes, accompanied by hemorrhage and infection. Loss of hair after 2 weeks for doses over 300 rem. Recovery in 20 to 100 percent within 1 month to a year.
600 to 1000	Vomiting within 1 hr, severe blood changes, hemorrhage, infection, and loss of hair. From 80 to 100 percent of exposed individuals will succumb within 2 months; those who survive will be convalescent over a long period.

which have a short life span. The most important of these are the white blood cells (lymphocytes and granulocytes) and the cells lining the intestines. Following an acute exposure to radiation the lymphocyte count will fall rapidly. If the animal (or man) is going to survive, the count will rise again after a few days, although full recovery may take several months. As seen in Table 3, the symptoms observed in an overexposed person depend quite particularly on the magnitude of the dose received. Table 4 lists the symptoms observed at various times after an exposure to a large whole-body dose of radiation (400-600 rem).

Only a small percentage of individuals would survive an acute whole-body dose of 1000 rads. Doses up to several thousand rad cause death within one or two weeks. Larger

doses result in death within days. In addition to affecting the gastrointestinal tract, very high doses damage the central nervous system, and an exposed individual can be almost immediately incapacitated. Death can be caused by circulatory or respiratory failure.

ii. Delayed Somatic Effects.

The delayed effects of radiation may be due to either a single acute overexposure or chronic low-level over-exposure. The latent period between exposure and appearance of a symptom can be months, years, or decades. Table 5 summarizes some of the long-term effects of radiation. Since these effects occur with normal spontaneous incidence in populations, it is often impossible to ascribe to radiation the occurrence of a symptom in a particular exposed individual. Delayed somatic effects due to radiation in a population appear to

Table 4. Symptoms Observed at Various Times After Exposure to a Dose of 400-600 rems (from Martin and Harbison, Ref. 28, with permission by Chapman and Hall, Ltd., © 1972).

Time after exposure	Symptoms observed
0-48 hours	Loss of appetite , nausea, vomiting, fatigue and prostration
2 days to 2-3 weeks	The above symptoms disappear and the patient appears quite well
2-3 weeks to 6-8 weeks	Purpura and haemorrhage, diarrhoea, loss of hair (epilation), fever and severe lethargy. It is during this period that fatalities occur
6-8 weeks to several months	This is the recovery stage during which surviving patients begin to show a general improvement and the severe symptoms tend to disappear

Table 5. Some Long-Term Effects of Radiation (from Martin and Harbison, Ref. 28, with permission by Chapman and Hall, Ltd., © 1972).

Effect	Mean Latent period	Evidence
Leukaemia	8-10 years	Atomic bomb casualties Medical X-ray treatment
Bone cancer	15 years	Radium luminous dial painters
Thyroid cancer	15-30 years	Atomic bomb casualties Medical treatment
Lung cancer	10-20 years	Mine workers
Life shortening		Experiments with mice
Cataract formation	5-10 years	Atomic bomb casualties

occur statistically; an individual in that population is presumably placed at a higher risk of injury.

Radiation carcinogenesis in man is the most important effect produced in the individual by low doses. It is at present impossible to determine from animal data the frequency at which the induction of malignancies by radiation carcinogenesis occurs in man. UNSCEAR (Ref. 21) has examined the frequency with which such changes are induced in different tissues. The review presented in UNSCEAR 77 is of estimates of the induction for malignancies derived from studies of human populations in which the whole body or individual organs have been irradiated; it updates the information presented in the 1972 report by UNSCEAR.

Fig. 1. Incidence of radiation-induced leukemia in humans. Curve A illustrates a linear hypothesis for the effect of low doses; Curve B represents a threshold-type hypothesis (from Hurst and Turner, Ref. 9, with permission by John Wiley & Sons Publishers © 1970).

Figure 1 shows schematically the correlation between high radiation doses and the probability per year of the occurrence of leukemia. The peak incidence of the disease among the survivors at Hiroshima and Nagasaki after the bombings in 1945 occurred during the period 1950 to 1952, an abnormally high incidence continuing beyond this time. Further studies of these exposed populations showed evidence of an increased incidence of other types of cancer.

An example of disease produced in humans, as a result of radiation exposure internally over a long period of time, is the bone cancers in the radium dial painters. Like leukemia, such effects have also been demonstrated in laboratory experiments with animals.

Studies of irradiated animals show a shortening of the life span not attributable to any specific cause. Experiments with mice indicate that a life shortening of 1 or 2% occurs as a result of acute radiation doses of about 100 rad.

The production of radiation-induced lens opacities and cataracts has been demonstrated. Acute exposure to radiation appears to be more serious than protracted exposure in producing cataracts, which may appear several years after the exposure. The x-ray dose required to produce clinically significant cataracts is probably at least 400 rad. Fast neutrons are much more effective. Neutron RBE values for cataract production in excess of 10 have been reported.

The irradiation of pregnant women can have adverse effects, some of which are manifested in subsequent years by the child. In addition to abnormally high still-birth and infant mortality rates among Japanese survivors pregnant at the time of the atomic bombings, there appears to be an above-normal amount of mental retardation in the offspring. Small children who are exposed also appear to lag somewhat behind their contemporaries in overall physical development. These effects are noticeable only among survivors who received doses of at least several hundred rad.

Doses of a few thousand rad have been given with beta rays to the skin of the body without affecting deeper-lying organs. In many cases a chronic abnormal condition of radiodermatitis developed over a period of years after exposure. The skin became thin and susceptible to ulceration and the formation of cancer.

(b). Genetic Effects. The genetic effects of radiation result from damage to the reproductive cells. These genetic changes are of different kinds: 2a) gene mutation, i.e., alteration in the function of individual genes; b) chromosome aberrations, resulting from breakage and reorganization of the chromosome; and c) changes in the number of chromosomes. Most of these mutations are lethal and result in miscarriage or still birth. Some of these changes result in offspring suffering abnormalities which may range from mildly detrimental to severely disabling or lethal disorders. It should be noted that gene mutations occur spontaneously. The rate in a population can also be increased by elevated temperatures, ultra-violet light, and certain chemical agents. It can in fact be said that no biological effect is specifically peculiar to radiation, other chemicals and toxic substances can cause some of these effects. No single substance, however, causes all the effects that radiation does.

For purposes of discussion it is useful to introduce the so-called "doubling dose" for mutation rates. This is the lifetime dose that will double the spontaneous rate. Estimates (Ref. 2) of this value range from 15 r to 150 r per lifetime. For purposes of radiation protection, the NCRP has accepted a value of 10 r and has recommended that the genetically significant dose be kept below this value.

Since there is a natural incidence of genetic disorders in man, the risk to the population is estimated in terms of an increase in incidence. For low dose or chronic irradiation, the expected increase will be about 1 per cent per rad (Ref. 2a) of added average dose to individuals during their lifetime. For these genetic hazards, their potential increase due to radiation does not mean that there is any present danger to an individual or his immediate descendants if he has received a doubling dose of radiation. It is the possible accumulation of harmful genes in the general population and their transmission to future generations which is the keystone of concern from a long term point of view of geneticists.

3. Threshold and Dose Response

At the present it is not known whether there exists a threshold or certain minimum absorbed dose below which no biological damage occurs. Although the effects just described are demonstrable with large amounts of radiation, doses can be made so low that it is virtually impossible to measure any effects, even with very large study populations. Figure 1 as an example, shows how the probability per year (incidence) of leukemia in man varies with radiation dose. Studies of Japanese survivors from the atomic bombs at Hiroshima and Nagasaki and of patients receiving therapeutic x-ray treatments indicate that the solid curve describes the relationship between dose and incidence when the dose is several hundred rad or greater. Below about 100 rad, too few cases of leukemia occur in the populations studied to demonstrate clearly an increased incidence due to radiation.

At doses of 10 rad or 1 rad, based on the observed incidence at high doses, the size of an exposed population needed to show a significant statistical increase would be enormous, and therefore experiments with an animal population are not usually feasible. One assumption frequently made is that the approximately linear solid curve at high doses should be extrapolated linearly to zero dose, as shown by the dashed Curve A in Fig. 1. According to this hypothesis, any amount of radiation, no matter how small, increases the probability of leukemia (or other effects) above the normal value by an amount proportional to the dose equivalent.

A contrary hypothesis is that the body can repair damage done by small amounts of radiation. According to this hypothesis, the body tolerates low doses, and a certain threshold dose is required before the probability of leukemia or other effects is increased above its normal value. This threshold hypothesis is illustrated by the dashed curve B in Fig. 1. Experimentally differentiating between a linear or a threshold response for any biological effect is complicated by the fact that radiation-induced effects occur with a natural incidence.

On the other hand, curves with shapes other than those of A and B have also been postulated for connecting the high dose region, in which data are statistically significant, to the region of low dose. The linear extrapolation may even underestimate the risk at low doses, particularly for high LET radiation (Ref. 17).

The EPA has commissioned a new report by the Advisory Committee on the Biological Effects of Ionizing Radiations of the National Academy of Science and National Research Council in order to look at the linear, no threshold, dose-effect curve in light of evidence within the past few years (Ref. 18).

Fig. 2. The effect of Dose Rate. The variation with Dose Rate of the LD-50 (the total absorbed dose required to produce 50% lethality) in 30 days in a population of mice (from Thompson and Fortellatte, Ref. 19, with permission by American Roentgen Ray Society © 1953).

Interpreting studies on biological effects of radiation is further complicated by a dose-rate dependence that has been demonstrated, both for early and latent effects. The data in Fig. 2 (from Ref. 19) show the variation with

dose rate of the LD-50 (the total absorbed dose required to produce 50 per cent lethality in 30 days) in a population of mice. At rates below about 250 rad/hour, the LD-50 is larger (about 1600 rad when the rate is 6 rad/hr). Genetic studies with mice and fruit flies have shown that fewer mutations per rad occur when the dose rate is low than when it is high. Figure 3 (from Ref. 20) shows experimental data for the mouse. Whether or not there is a threshold below which little or no effect occurs remains to be determined.

Fig. 3. Dose curves for specific locus mutations in the mouse. Solid points (90% confidence) represent results from acute x-rays (80-90 R/min); open points are chronic gamma doses (10-90 R/week). Lower 1000 R point represents a single exposure; top one, successive exposures of 600 and 400 R, 15 weeks apart (from Russell et al., Ref. 20, with permission by Taylor and Francis, Ltd. © 1960).

A study by R. O. McClellan, et al. (Ref. 21) of the Inhalation Toxicology Research Institute investigating the influence of variations in dose rate on the biological effects of inhaled beta-emitting radionuclides on beagle dogs found that higher initial dose-rate exposures were more effective than low dose-rate exposures on a per-rad basis in producing early effects.

That decreasing the dose rate decreases the RBE is well accepted for $Q = 1$ radiation (β and γ radiation). An exception seems to be the case of tritium (^3H , $t_{1/2} = 12.5$ yr, $E_{\text{max}} = 0.018$ MeV) where in cases of chronic exposures at low dose rates its RBE increased to ~ 1.5 (Ref. 22). However, that a given dose can be less effective when delivered at once than when administered at intervals is considered the exception rather than the rule. Some scientists feel that high-LET radiation exhibit this type of increasing RBE (Ref. 23). The response of an organism appears to depend in a complicated way on many factors. A number of theories of damage and recovery have been proposed to account for observed dose-rate effects. For instance, the Keller-Rossi "dual-action" theory (Ref. 24) predicts an increased RBE for neutrons at low dose rates, which seems to be the case (Ref. 25).

Another area where there is difficulty in designing experiments and interpolating the results is in the study of effects in cell cultures. The difficulty arises because some studies are done in isolation from the cells in their normal environment where they are a component in a complex biological system. In fact, interpreting outcomes of cellular effects often requires presumption of a repair mechanism in order to make predictions fit experimental results.

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D. Radiation Protection Philosophy Criteria,
Guides and Standards

This section reviews the development of radiation protection philosophy and presents the standards that have been promulgated or advised by the appropriate authorities. Fuller discussions can be found in the pertinent publications of the National Council on Radiation Protection and Measurements (NCRP) or International Commission on Radiological Protection* (ICRP). Several standard texts also provide relevant discussions; cf. Taylor, (Refs. 1, 2) Glasstone (Ref. 3) Morgan (Ref. 4) and Cember (Ref. 5). This discussion stresses standards aimed at the general population and individual members of the population. However, when necessary, occupational standards will be presented.

1. Development of Radiation Protection Criteria

After radiation was discovered by Becquerel and began to be used, it was quickly realized that there were definite biological effects. One of the earliest effects was Edison's and his assistants' complaint of a smarting of the eyes after doing an experiment with a cathode ray tube. However this recognition extended only to negative effects at high doses. Low dose effects were not recognized until the middle of this century. Early medical use resulted in some unfortunate overdoses leading sometimes to immediate results and thus the discovery of definite drastic biological endpoints. However, up until 1928, there was no large scale effort in radiation protection activities; most of the responsibilities fell to the societies and organizations of active radiologists.

The initial concerns of protection were focused primarily on radiologists to prevent gross biological effects from external radiation. As a result, shielding was the primary method recommended to cut down exposures. Fortunately, X-rays were found to be stopped more effectively by lead than any other common metal, resulting in its gaining early acceptance for radiation protection purposes. Due to this observation, along with the absence of any knowledge of radiation dosimetry, all the early protection recommendations were based on shielding factors rather than limiting dose.

During the 1920's, experience began to be gained concerning the effects of a bone-seeking α -emitter, ^{226}Ra . Radium dial painters would use their lips and tongues to keep a sharp tip on their brushes, thus ingesting radium. Over the years they have been studied, and it has

* These organizations and any others introduced will be discussed in detail at the end of this section.

been learned that a bone-seeking α -emitter can cause bone cancer, with a latent period of about 15 years. Through determining the body burdens of ^{226}Ra , a rough dose-effect curve began to take shape.

It was largely these "occupational" experiences at high dose levels in radiologists and radium dial painters that led to an awareness of the danger of high doses of radiation. However this experience was not put on a quantitative basis, such as has been done today,* because no measure of radiation exposure had been agreed upon. It was not until the 1930's that the roentgen was adopted as a measure of exposure. (An early attempt to quantify radiation dose was done in terms of how much radiation it took to cause reddening of the skin, which clearly exhibited a threshold.) In 1934, the ICRP adopted a recommendation specifying a tolerance dose expressed in roentgen, thereby replacing shielding as the sole protection concept. The first standards put forth by the ICRP after its establishment in 1928, were only intended for use with X-rays, since it wasn't certain the effects for γ -rays and α -particles were related to those of X-rays. Prior to about 1930 radiation protection developed haphazardly, but after that the creation of the International Commission on Radiation Units and Measurement (ICRU), ICRP, and NCRP, national and international supervision helped it develop more smoothly.

The experiences of the radium dial painters has shown that radiation could also pose an internal hazard. As the scope of the radiation hazard began to be realized, attention also shifted from simply protecting people who worked with radiation, to considering the safety of a patient. Except for scientific research (of which there certainly was not as much as there is today), the primary users of radiation up to about World War II were medical workers. This situation changed rapidly in the 1940's and 1950's. Even so, until the late 1950's radiation protection philosophy was aimed at prevention of the immediate acute effects of radiation exposure. Only slowly were the latent effects of radiation recognized, largely because of follow-up studies of the Hiroshima-Nagasaki survivors.

With the development of fission, and its attendant uses in weapons and for electricity production, it was recognized that in addition to exposures of large numbers of workers, many members of the general public might run the risk of exposures to fallout and the effluents from a nuclear fuel cycle. At the advent of the atomic age, it became obvious that, with such widespread use, the public needed

* See for examples BIER 72 (Ref. 24) UNSCEAR 77 (Ref. 12).

protection from ionizing radiation. It also became evident, that since internal radiation could cause cancer and long-lived radioactivity was becoming widely dispersed in waterways and the environment, the largest mode of exposure for the public was internal. In the late 1950's limiting internal exposures was considered by regulatory bodies. Concentration standards to be met in water and air were derived from the maximum permissible body burden calculated from a consideration of the particular internal dosimetry of certain radionuclides. Additional departures in the 1950's were: the introduction of a benefit-cost consideration, although not numerical; and the formal dropping of the assumption that the biological effects of radiation have a definite threshold, largely on the evidence that there did not seem to be any threshold for genetic damage.

Presently the primary standards for radiation protection are given in terms of limiting radiation dose to the whole body, or particular body organs, from all sources of radiation other than natural background and medical exposures. There seems to be a shift towards including medical exposures as a part of those sources of radiation to be limited. (See (Refs. 8, 9). Further derivations yield in turn the limits of annual intake that will keep organ and body burdens within these limits even after a lifetime of intake. If only single modes of exposure are considered, the maximum permissible concentrations in air and drinking water can be derived. Whenever there is more than one activity present, the MPC's must be altered appropriately.

When radioactive materials are released to environment, there may be a variety of pathways potentially leading to the radiation exposure of man. All of the ways that may contribute significantly to radiation exposure should be considered in estimating potential radiation doses. (See Section I.F.7 for a discussion of critical pathway analysis.) In determining the total detriment to mankind from a particular release of radioactive materials, one should be concerned with the total dose from that source to all persons both now and in the future. In this case the radionuclides may retain a potential for radiation exposure far into the future. A concept gaining increasing favor is to place the restriction on the "dose commitment" from the release rather than on the dose rate.

2. Current Radiation Protection Philosophy

One of the significant features of existing philosophy is the recognition of the fact that radiation protection criteria depend not only upon purely biomedical and physical considerations, but also upon value judgments (Ref. 2). These criteria presently include the concept that there may be some de-

gree of risk at any level of exposure. As the understanding of the radiation hazard grew, the acceptable levels grew smaller, as can be seen by looking at how radiation protection standards evolved numerically over the years (see Table 1).

Table 1. Evolving Radiation Standards Recommended by the NCRP

1934	0.1 R/d	(or 0.5 R/wk)	(=25 R/yr)
1949		0.3 rem/wk	(=15 rem/yr)
1957		0.1 rem/wk	(= 5 rem/yr)
Present			5 rem/yr

Up until 1971, the NCRP explicitly reviewed only the basic principles underlying the recommendations for external sources of ionizing radiation. The release of NCRP Report No. 39 (Ref. 6) rectified that apparent anomaly. That report and others will be quoted to review the general criteria under which the advisory and regulatory agencies operate.

"Radiation protection recommendations must set a pattern as scientific as possible, but in practice partially empirical; ideally, the rational use of radiation should conform to levels of safety to users and the public which are at least as stringent as those achieved for the other powerful agents. It is important to avoid unnecessary rigidity in any protection recommendations so that the advantages of new knowledge can be exploited quickly and easily. This basic posture must be supplemented by somewhat detailed guidelines for control in numerous specific applications; these guidelines may change frequently in the light of practical experience, availability of improved measuring methods and the complexity of competing applications involving radiation. Bodies promulgating recommendations must maintain cognizance of advances in the understanding of radiation effects so that the basic criteria may become better founded in scientific principles and less dependent on value judgements, whenever possible" (Ref. 6).

"The NCRP position is centered on the principle that the "lowest practicable" radiation level is the fundamental basis for establishing radiation standards, and on the assumption that the most important radiation health hazards do not have a dose threshold. On this basis, the setting of radiation protection standards requires consideration of compensatory trade-offs between currently assumed hazards and benefits (Ref. 7). A basic assumption of conventional benefit-cost analysis is that human welfare (which this procedure seeks to maximize) may be measured in monetary or materialistic terms. This assumption has fundamental problems in that it cannot handle the

situations in which the risks and benefits do not accrue to the same people. The BEIR Committee has examined the problem of evaluation of the total benefits derived from exposure to ionizing radiations for comparison with the total risks. The reader is referred to their recently completed report (Ref. 8) for a more detailed consideration of the myriad factors involved in health benefit-cost analyses.

"The NCRP continues to hold the view that risk estimates for radiogenic cancers at low doses and low dose rates derived on the basis of linear (proportional) extrapolation from the rising portions of the dose-incidence curves at high doses and high dose rates... cannot be expected to provide realistic estimates of the actual risks from low-level, low-LET radiations, and have such a high probability of overestimating the actual risk as to be of only marginal value if any, for purposes of realistic risk-benefit evaluation (Ref. 7).

"Numerical radiation protection guides or dose limits for the exposure of radiation workers or the general public are provided only as upper limits; all exposure should be kept to a practicable minimum" (Ref. 7,21).

The NCRP and the ICRP both base their recommendations on a system of dose limitations, the basic features of which are as follows (Ref. 9):

"(a) No practice shall be adopted unless the introduction produces a positive net benefit.

"(b) All exposure shall be kept as low as reasonably achievable, economic and social factors taken into account; and

"(c) the dose equivalent shall not exceed the limits recommended for the appropriate circumstances by the Commission".

Two additional conservative positions are at the basis of the recommendations of the ICRP and NCRP (Ref. 2).

"(a) All low doses delivered to an organ are completely additive, no matter at what rate they are delivered or what intervals there may be in delivery.

"(b) There is no biological recovery of radiation effect from low doses".

If indeed the conservative assumptions concerning radiation effects are true, then the root of the problem in setting radiation protection standards is what level of damage and deaths is acceptable and to whom.

3. Radiation Protection Organizations, Recommendations, Guides, and Standards

We will introduce the organizations in this section and then present their recom-

mendations, if appropriate, giving an explanation of where and how they are applicable. Since international bodies are influential in setting guidance for the national bodies, we will first discuss the former.

INTERNATIONAL AGENCIES. The International Commission on Radiological Protection (ICRP) was established in 1928, as the International X-ray and Radium Protection Committee, by the Second International Congress of Radiology. It assumed the present name and organizational form in 1950 in order to cover more effectively the rapidly expanding field of radiation protection. The ICRP has continued its close relationship with succeeding Congresses of Radiology, and it has also been looked to as the appropriate body to give general guidance on the more widespread use of radiation sources caused by the rapid developments in the field of nuclear energy. The ICRP maintains its traditional contact with medical radiology and the medical profession generally, and it also recognizes a responsibility to other professional groups and an obligation to provide guidance within the field of radiation protection as a whole. The policy adopted by the ICRP in preparing its recommendations is to consider the fundamental principles upon which appropriate radiation protection measures can be based while leaving to the various national protection bodies the responsibility of formulating the specific advice, codes of practice, or regulations that are best suited to the needs of their individual countries.

The ICRP has an official relationship with the World Health Organization and the International Atomic Energy Agency. Close working relationships are maintained with the United Nations Scientific Committee on the Effects of Atomic Radiation, The International Labour Office, the United Nations Environment Program, the Nuclear Energy Agency, and the European Economic Community by sending observers to meetings organized by these bodies. In addition, the above bodies are invited to send representatives to the ICRP's meetings with its committees.

As the purpose of the ICRP is not to promulgate dose limits per se, but only to suggest recommendations which can be used by various bodies who then promulgate dose limits or standards, their recommendations are not best represented in a set tabular format (Ref. 10). The current ICRP recommendations are presented in Appendix A by excerpting appropriate parts from ICRP Publication 26 (Ref. 9). Basically, their recommendations are to the effect that individual members of the public should not be exposed to a dose that entails a risk greater than that arising from 0.5 rem uniform irradiation of the whole body, even when individual organs are irradiated.

The International Commission on Radiological Units and Measurements (ICRU) is the

international advisory body which, since its inception in 1925, has primarily concerned itself with the development of acceptable recommendations regarding:

"(1) Quantities and units of radiation and radioactivity,

"(2) Procedures 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" (Ref. 11).

In addition, the ICRU considers and makes such recommendations for the radiation protection field. It does this aspect in close cooperation with the ICRP. The ICRU collects and evaluates the latest data and information pertinent to the problems of radiation dosimetry and measurement and recommends the most acceptable values and techniques for current use. In its operating policy, "The ICRU feels it is the responsibility of national organizations to introduce their own detailed technical procedures for the development and maintenance of standards. However, it urges all countries adhere as closely as possible to the internationally recommended basic concepts of radiation quantities and units."

The ICRU maintains close relationships with other organizations in much the same fashion as the ICRP. They have formal working arrangements with WHO, IAEA, UNSCEAR. Informal connections exist with many more organizations, such as the International Organization for Standardization.

The United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), since the 1950's, regularly reviews the scientific literature concerning biological effects of ionizing radiation and trends in the radiation exposure pathways (i.e., natural radiation background, nuclear fuel cycle, medical radiation, occupational and patient, etc.) These reviews have in the past caused some restructuring of the current ideas affecting radiation protection guidance. An example is the recognition that carcinogenesis is the main danger for low doses (Ref. 12). This reverses the previous view that the genetic component of damage should be the overriding concern in the setting of dose limits for the general population. These reviews contribute to ICRP development of radiation protection recommendations by providing estimates of the frequency of health effects.

The International Atomic Energy Agency (IAEA) was organized in 1956 to "accelerate and enlarge the contribution of atomic energy to peace, health, and prosperity throughout the world." Its membership is composed of

approximately 105 member states of the United Nations. It was "empowered to provide for the application of standards of safety for protection against radiation to its own operations and to operations making use of assistance provided by it or with which it is otherwise directly associated. To this end, authorities receiving such assistance are required to observe relevant health and safety measures prescribed by the Agency" (Ref. 13). The recommendations of the IAEA rely heavily on the basic recommendations of the ICRP and ICRU, and are set forth in manuals and codes on health and safety published in the Safety Series or Technical Report Series. These manuals serve as basic reference documents for the preparation of national health and safety documents covering the use of radionuclides.

Some work of the IAEA is done in collaboration with other U.N. Agencies. For instance, with the International Labor Organization the IAEA has issued general safety guidance for the mining and milling of radioactive ores (See Section II.E). In addition, in collaboration with the World Health Organization, the IAEA has issued guidance on handling of radionuclides, environmental monitoring, and other topics. The IAEA with the Food and Agricultural Organization monitors radiation contamination of food, and with the World Meteorological Organization monitors isotopes in precipitation.

Other international organizations also address nuclear issues. Some of these are Euratom, Inter-American Nuclear Energy Agency, and the Organization for Economic Cooperation and Development's Nuclear Energy Agency. An international professional organization that sponsors pertinent symposia is the International Radiation Protection Association.

NATIONAL AGENCIES: The National Council on Radiation Protection and Measurements (NCRP) provides for the United States the same sort of general guidance and recommendations that the ICRP and ICRU provide at an international level. The NCRP started in 1928 as an informal unit of the National Bureau of Standards (NBS) to provide a focus for the U. S. to give input to the ICRU and later, ICRP. It was chartered in 1964 by Congress as a non-profit corporation to:

"1. Collect, analyze, develop, and disseminate in the public interest information and recommendations about (a) protection against radiation and (b) radiation measurements, quantities, and units, particularly those concerned with radiation protection;

"2. Provide a means by which organizations concerned with scientific and related aspects of radiation protection and of radiation quantities, units, and measurements may cooperate for effective utilization of their

combined resources, and to stimulate the work of such organizations;

"3. Develop basic concepts about radiation quantities, units, and measurements, about the application of the concepts, and about radiation protection.

"4. Cooperate with the International Commission on Radiological Protection, the Environmental Protection Agency, Nuclear Regulatory

Commission, International Commission on Radiation Units and Measurements, and other national and international organizations, governmental and private, concerned with radiation quantities, units, and measurements and radiation protection" Ref. 6). Like ICRP and ICRU, the NCRP maintains working relationships with many national and international organizations.

The current recommendations of the NCRP are presented in Table 2. It is to be empha-

Table 2. Dose-limiting recommendations of NCRP (from NCRP, Ref. 6, with permission by National Council on Radiation Protection and Measurements © 1971).

Maximum Permissible Dose Equivalent for	
Occupational Exposure	
Combined whole body occupational exposure	
Prospective annual limit -	5 rems in any one year
Retrospective annual limit -	10-15 rems in any one year
Long term accumulation to age N years -	$(N - 18) \times 5$ rems
Skin -	15 rems in any one year
Hands -	75 rems in any one year (25/qtr)
Forearms -	30 rems in any one year (10/qtr)
Other organs, tissues and organ systems -	15 rems in any one year (5/qtr)
Fertile women (with respect to fetus) -	0.5 rem in gestation period
Dose Limits for the Public, or Occasionally Exposed Individuals	
Individual or Occasional -	0.5 rem in any one year
Students -	0.1 rem in any one year
Population Dose Limits	
Genetic -	0.17 rem average per year
Somatic -	0.17 rem average per year
Emergency Dose Limits - Life Saving	
Individual (older than 45 years if possible) -	100 rems
Hands and Forearms -	200 rems, additional (300 rems, total)
Emergency Dose Limits - Less Urgent	
Individual -	25 rems
Hands and Forearms -	100 rems, total
Family of Radioactive Patients	
Individual (under age 45) -	0.5 rem in any one year
Individual (over age 45) -	5 rems in any one year
Individual (over age 45) -

sized that these are designed to serve as reasonable guidance and are not to be adopted verbatim by the agencies.

The U. S. Environmental Protection Agency (EPA) is one of the primary regulatory bodies in the United States concerned with radiation protection. The authority of the Federal Radiation Council (FRC) to provide guidance for use by Federal agencies in developing radiation standards was transferred to the EPA with its formation on December 2, 1970 by Executive Reorganization Plan No. 3.

Under this authority, the EPA provides generally applicable guidance in the form of recommendations similar to those developed by the NCRP (Table 2), as well as guidance on maximum permissible concentrations. The EPA, in addition, formulates "protective action guides" applicable to accidental releases of radioactivity. The EPA is also required to establish radiological standards for the protection of the general environment through a transfer of authority from the Atomic Energy Commission. "Standards mean limits on radiation exposure of levels, or concentrations or quantities of radioactive materials in the general environment outside the boundaries of locations under the control of persons possessing or using radioactive material" (Ref. 14). The EPA recently promulgated standards (Ref. 15) to limit radiation doses to the general public and quantities of long-lived radioactive materials in the general environment attributable to planned releases from operations contributing to the nuclear generation of electrical power from the uranium fuel cycle. These standards apply to nuclear power plants and to most other operations within the fuel cycle, but exclude mining, transportation and waste management. The standards limit: (1) the annual dose equivalent to the whole body or any internal organ, except the thyroid, to 25 millirems, and the annual dose equivalent to the thyroid, to 75 millirems; and (2) the quantities of ^{85}Kr , ^{129}I , and plutonium and other alpha-emitting transuranic elements with half-lives greater than one year released to the environment per gigawatt-year of electrical power produced by the entire fuel cycle to 50,000 curies, 5 millicuries, and 0.5 millicuries, respectively. The rule is reproduced from the Code of Federal Regulations in Appendix D of Section II.B. These standards were proposed to achieve two main objectives: (1) assure protection of members of the public against radiation doses resulting from fuel cycle operations, and (2) to limit the environmental burden of long-lived radioactivity that may accumulate as a result of the production of electrical energy, so as to limit their long-term impact on both current and future generations. For detailed discussion of the data base and judgements upon which these standards are based, the reader is referred to the final environmental statement (Ref. 15).

The EPA has issued proposed guidance (Ref. 22) on dose limits for individuals exposed to transuranium elements from presently existing cases of environmental contamination or from future unplanned release. The recommendations states that:

"1. The annual alpha radiation dose rate to members of the critical segment of the exposed population as the result of exposure to transuranium elements in the general environment should not exceed either:

- a. 1 millirad per year to the pulmonary lung, or
- b. 3 millirad per year to the bone..."

In addition, the EPA's jurisdiction includes the environmental radiation protection activities of the Department of Health, Education, and Welfare covered under the Public Health Service Act, except those functions kept by the Bureau of Radiological Health (discussed later). In carrying out these functions, the EPA on July 9, 1976 promulgated Interim Primary Regulations for Radioactivity in Drinking Water (Ref. 16). These regulations replace the Public Health Service Drinking Water Standards of 1962. These regulations set forth maximum contaminant levels for ^{226}Ra , ^{228}Ra , and gross α -particle radioactivity as follows (Ref. 17):

(1) Combined ^{226}Ra and ^{228}Ra limited to 5 pCi/l in community water systems. (2) Gross alpha particle activity (including ^{226}Ra , but excluding radon and uranium) limited to 15 pCi/l. In addition, maximum contaminant levels for β -particle and photon radioactivity from man-made radionuclides in community water systems were set as the following:

● the average annual concentration of β -particle and photon activity from man-made radionuclides in drinking water shall not produce an annual dose equivalent to the total body or any internal organ greater than 4 millirem/year (0.004 rem/year).

● Except for the radionuclides [^3H and ^{90}Sr], the concentration of man-made radionuclides causing 4 millirem total body or organ dose equivalents shall be calculated on the basis of a 2 liter per day drinking water intake using the 168 hour data listed in Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air or Water for Occupational Exposure: NBS Handbook 69 as amended August 1963, U. S. Department of Commerce. If two or more radionuclides are present, the sum of their annual dose equivalent to the total body or to any organ shall not exceed 4 millirem/year" (Ref. 16). For ^3H and ^{90}Sr , the limits are 20,000 pCi/liter and 8 pCi/liter, respectively.

As these standards are of environmental importance, parts of National Interim Primary Drinking Water Regulations (Ref. 16) which provide further clarification and application of these standards are reproduced in Appendix B following this section.

The U. S. Nuclear Regulatory Commission (NRC), formed by the Energy Reorganization Act of 1974, is broadly speaking, the regulatory part of the old Atomic Energy Commission (AEC). The NRC is responsible for the licensing and regulatory functions relating to commercial nuclear facilities and implements radiation protection standards by defining specific requirements in the licenses of individual plants and also by enforcing them. As its other major responsibility the NRC regulates the following:

(a) "Source material", meaning uranium or thorium, or ores with above a certain concentration of either or both

(b) "by-product material" meaning any radioactive material (except below) yielded in, or made radioactive by, producing or using special nuclear material

(c) "Special nuclear material" meaning plutonium, uranium-233, uranium enriched in the isotope 233 or 235, any material enriched by the foregoing, and other material designated by the NRC."

Much of the regulation of these materials is actually carried out by the states rather than by the NRC itself. About half the states are "Agreement" states that meet certain federal stipulations in this regulatory function. The role of the NRC with respect to nuclear power reactors is presented in Section II.B.

The NRC is responsible for occupational standards for workers involved in an NRC-licensed activity. These standards are given in Title 10, Parts 19 and 20 of the Code of Federal Regulations and limit each worker to a quarterly dose of 3 rem and an annual dose of 5 rem. to the whole body. The NRC is currently considering amendments (Ref. 23) to remove a provision which allows certain workers in licensed commercial nuclear activities to receive a radiation dose of as much as 12 rem in one year.

The U.S. Department of Energy (DOE) superseded the Energy Research and Development Administration in October 1977. As the agency designated to take over the AEC's research and development functions, DOE is responsible for the great bulk of research on the biomedical, environmental, physical, and safety aspects of nuclear and other kinds of energy. It is also responsible for radiation health and safety and environmental protection at DOE owned facilities, such as the National Laboratories. It has statutory power over radiation guidelines in its own areas of concern, namely: the National Laboratories, DOE licensees and contractors, and the weapons

testing program. The publications in DOE's Technical Information Center provide valuable reference materials for radiation protection as a whole, and they are referenced where appropriate. The DOE Environmental Measurement Laboratory (EML) [formerly Health and Safety Laboratory (HASL)] performs routine measurements of radiation contamination in food, air, water and fallout. These data are published in environmental quarterlies (Ref. 18). EML also publishes a procedures manual (Ref. 19) for radioactivity measurements.

The Food and Drug Administration's (FDA) Bureau of Radiological Health (BRH) of the U.S. Department of Health, Education, and Welfare (HEW) has many general public health responsibilities associated with radiation protection (Ref. 20). It conducts an electronic product radiation control program, including the development and administration of performance standards. As the agency primarily responsible for radiation used in the healing arts, the Bureau develops criteria, recommendations, and standards relative to radiation use and exposure, as well as developing techniques, procedures and users' qualifications, in order to reduce unnecessary exposure. BRH also provides advice to the Bureau of Foods and the Bureau of Drugs on the control of radioactive materials and radiation in food and drugs. Other functions include research, technical assistance and training in occupational radiation exposure, research on health effects of radiation exposure, and participation in the development of model codes and recommendations.

The Mining Safety and Health Administration (MSHA) of the Department of Labor regulates occupational exposures in the mining industry. They are also responsible for developing and enforcing safety and health standards for mineral extraction operations. Their role is discussed in Section II.E. on mining and milling.

The U. S. Department of Transportation (DOT) regulates the transportation of all hazardous materials, including radioactive materials. DOT regulates both shippers and carriers who are engaged in interstate commerce, and prescribes packaging, marking, labelling, loading, and storage regulations.

Several other federal government agencies serve specialized functions in providing services or data useful to the aforementioned agencies. For instance, the Soil Conservation Service, part of the Department of Agriculture, monitors soil for radionuclides. The Occupational Safety and Health Administration, part of the Department of Labor, develops and enforces health and safety standards for those occupationally radiation-exposed groups not covered by other federal agencies. The National Bureau of Standards, part of the De-

partment of Commerce, serves a very important role in providing a basis for national measurement systems by providing standards for radioactivity and standard reference materials for analysis.

STATE AGENCIES. Under the constitution the ultimate responsibility in the United States for most aspects of health protection - including radiological health - is vested in the states. Many states accept this responsibility by protecting their citizens through basic public health law and by specific laws concerning specialized fields, e.g., radiological health and occupational health. In order to standardize the radiation control regulations in the states, the Councils of State Governments assisted by the AEC and BRH initially published in 1962 the first "Suggested State Regulations for Control of Radiation." In 1964, 1966, 1970, and 1974, the suggested regulations were revised and updated. Naturally occurring isotopes (radium, thorium, etc.) or machine-produced (e.g. accelerator) isotopes are under the direct control of states. To facilitate material assistance and cooperation between the states and various agencies of the federal government, a National Conference of Radiation Control Program Directors was established in 1968. One of the purposes of the Conference is to collect and make accessible to all radiation control program directors in official administrative positions such information and data as might be of assistance to them in the proper fulfillment of their duties.

In 1959, Section 274 of the Atomic Energy Act was enacted to spell out a State role in radiation activities and to provide a statutory basis under which the Federal Government could relinquish to the States portions of its regulatory authority. The mechanism to transfer some of NRC's authority to a State is an agreement between the Governor of the State and the NRC. The programs of "Agreement" States are routinely reviewed by the NRC to assure that they are effective. States may take advantage of NRC assistance through NRC's Office of State Programs.

Working cooperatively with various professional societies, many states are also involved with educational programs on radiation protection, providing activities such as special seminars, pamphlets, books, and lectures in schools of radiologic technology. Several states have minimum standards for the training and education of radiologic technologists.

LOCAL AGENCIES have a definite contribution to make in radiation protection, particularly through emergency preparedness activities for any radiation facilities within their jurisdiction.

NONGOVERNMENTAL ORGANIZATIONS. In addition to the previously discussed bodies there are organizations which are not governmental providing useful criteria for another aspect of radiation protection and that is the measurement of ionizing radiation and/or exposure. Of course, the NCRP, ICRU, and the IAEA also address this area of concern in some of their reports. The American National Standards Institute (ANSI), alone or in collaboration with the Institute for Electrical and Electronic Engineers (IEEE), provides standards to be followed voluntarily. The American Society for Testing Materials (ASTM) published a set of reference methods for measuring radiation in various media.

Other organizations that provide a variety of information on radiation protection matters are professional organizations, such as the American Chemical Society, Health Physics Society, IEEE Nuclear Science, American Physical Society, and the American Nuclear Society.

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APPENDIX A

Recommendations of the ICRP (from ICRP, Ref. 9,
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(101) The Commission's dose-equivalent limits are primarily intended to ensure adequate protection even for the most highly exposed individuals. In many occupations workers who are exposed near the dose-equivalent limits are unlikely to receive such doses each year of their occupational life and it would be their expected lifetime dose equivalent that would indicate their total individual risk. In this sense they are comparable with individuals who are exposed randomly to higher risks in "safe" occupations. Exposures consistently near the limits would be comparable with a situation where a higher-than-average risk has been identified for certain individuals in non-radiation industries.

(102) However, if the exposure of workers in any particular occupation were to be *planned* so that a large fraction of workers received dose equivalents which approached the annual limit, the average exposure could rise substantially above one-tenth of the limit. There would thus be a corresponding rise in the average risk, even though the annual dose-equivalent limit was not exceeded by any individual worker. Long-continued exposure of a considerable proportion of the workers at or near the dose-equivalent limits would only be acceptable if a careful cost-benefit analysis had shown that the higher resultant risk would be justified.

Recommended dose-equivalent limits. (103) The Commission's recommendations are intended to prevent non-stochastic effects and to limit the occurrence of stochastic effects to an acceptable level. The Commission believes that non-stochastic effects will be prevented by applying a dose-equivalent limit of 0.5 Sv (50 rem) in a year to all tissues except the lens, for which the Commission recommends a limit of 0.3 Sv (30 rem) in a year. These limits apply irrespective of whether the tissues are exposed singly or together with other organs, and they are

intended to constrain any exposure that fulfils the limitation of stochastic effects (see paragraphs 104-110).

(104) For stochastic effects the Commission's recommended dose limitation is based on the principle that the risk should be equal whether the whole body is irradiated uniformly or whether there is non-uniform irradiation. This condition will be met if

$$\sum_T w_T H_T \leq H_{wb,L}$$

where, w_T is a weighting factor representing the proportion of the stochastic risk resulting from tissue (T) to the total risk, when the whole body is irradiated uniformly, H_T is the annual dose equivalent in tissue (T), $H_{wb,L}$ is the recommended annual dose-equivalent limit for uniform irradiation of the whole body, namely 50 mSv (5 rem).

(105) The values of w_T recommended by the Commission are shown below:

Tissue	w_T
Gonads	0.25
Breast	0.15
Red bone marrow	0.12
Lung	0.12
Thyroid	0.03
Bone surfaces	0.03
Remainder	0.30

The value of w_T for the remaining tissues requires further clarification. For the reasons stated in paragraphs 58 and 59 the Commission recommends that a value of $w_T = 0.06$ is applicable to each of the five organs or tissues of the remainder receiving the highest dose equivalents, and that the exposure of all other remaining tissues can be neglected. (When the gastro-intestinal tract is irradiated, the stomach, small intestine, upper large intestine and lower large intestine are treated as four separate organs.)

(106) It is recognized that the risk associated with a given exposure will vary with the age and sex of the individual exposed. However, the

values of w_T given in paragraph 105 are recommended as appropriate for the protection of any worker, regardless of these sources of variability (see also paragraph 38).

(107) The values of w_T presented in paragraph 105 are intended as guidance for those concerned with calculating secondary and derived limits (see section F). In particular, they are used by ICRP Committee 2 in calculating values of annual limits of intake (ALI) for radionuclides, which take account of the dose equivalent in each tissue. In practical situations, however, it will usually suffice to use the two secondary limits that are applicable to external and internal exposure, namely, the limit to the dose-equivalent index H_I (see paragraph 108) and ALI (see paragraph 109).

(108) With *external exposures* to penetrating radiation, on those occasions when information is lacking concerning the actual distribution of dose equivalent in the body, it is possible to assess the *maximum* value of dose equivalent that would occur in a 30 cm sphere (the deep dose-equivalent index, H_I)*. The limitation of the dose-equivalent index to an annual value of 50 mSv would afford a level of protection that would be at least as good as that provided by the method recommended in paragraph 104.

(109) With *internal exposure* resulting from the intake of radionuclides protection can be based on annual limits of intake (ALI). These are calculated by ICRP Committee 2, from knowledge of the various organ committed dose equivalents per unit intake, by application of the principles discussed in paragraphs 104 and 105; such exposures are also subject to the limits for non-stochastic effects given in paragraph 103.

(110) When external and internal exposures are received together, the Commission's recommended dose limitation for stochastic effects will not be exceeded if:

$$\frac{H_I}{H_{wb,L}} + \sum_j \frac{I_j}{I_{j,L}} \leq 1$$

where, H_I is the annual dose-equivalent index, $H_{wb,L}$ is the annual dose-equivalent limit,

I_j is the annual intake of radionuclide j , $I_{j,L}$ is the annual limit of intake for radionuclide j .

(111) Although the Commission no longer proposes separate annual dose-equivalent limits for individual tissues and organs irradiated singly, the implied values of such limits may be obtained, if required, by dividing the dose-equivalent limit $H_{wb,L}$ (50 mSv in a year) by the relevant value of w_T . Such values would be subject to the limits, based on non-stochastic effects, given in paragraph 103.

(112) It should be recognized that the limits have been derived for application in average situations, for all adult ages and for both sexes and without regard to individual circumstances which might enhance the risk. The Commission believes that, for example, any variation in risk with age will not influence the total risk from a lifetime exposure unless the exposure is limited to a special group. Additional precautions and dose limitations may be necessary, however, to limit the irradiation of an embryo or foetus in the case of occupational exposure of pregnant women (see paragraphs 115 and 116).

Planned special exposures. (113) Situations may occur infrequently during normal operations when it may be necessary to permit a few workers to receive dose equivalents in excess of the recommended limits. In such circumstances external exposures or intakes of radioactive material may be permitted provided the dose-equivalent commitment does not exceed twice the relevant annual limit in any single event, and, in a lifetime, five times this limit. The Commission wishes to emphasize that external exposures or intakes of this magnitude are only justified when alternative techniques, which do not involve such exposure of workers, are either unavailable or impracticable (see also paragraph 171).

(114) Planned special exposures should not be permitted if the worker has previously received abnormal exposures resulting in dose equivalents in excess of five times the relevant annual limit. Planned special exposures should not be permitted for women of reproductive capacity. Dose equivalents resulting

*International Commission on Radiation Units and Measurements, The Conceptual Basis for the Determination of Dose Equivalent, ICRU Report 25, International Commission on Radiation Units and Measurements, Washington, 1976.

from planned special exposures should be recorded with those from usual exposures, but any excess over the limits recommended in paragraphs 103 *et seq.* should not by itself constitute a reason for excluding a worker from his usual occupation. (Accidental and emergency exposures are discussed in section G).

Occupational exposure of women of reproductive capacity. (115) When women of reproductive capacity are occupationally exposed under the limits recommended in paragraph 108, and when this exposure is received at an approximately regular rate, it is unlikely that any embryo could receive more than 5 mSv during the first 2 months of pregnancy. Having regard to the circumstances in which such exposures could occur, the Commission believes that this procedure will provide appropriate protection during the essential period of organogenesis.

Occupational exposure of pregnant women. (116) It is likely that any pregnancy of more than 2 months' duration would have been recognized by the woman herself or by a physician. For reasons described in paragraph 65, the Commission recommends that, when pregnancy has been diagnosed, arrangements should be made to ensure that the woman can continue to work only in Working Condition B, specified in paragraph 161.

Dose-equivalent limits for individual members of the public. (117) Radiation risks are a very minor fraction of the total number of environmental hazards to which members of the public are exposed. It seems reasonable therefore to consider the magnitude of radiation risks to the general public in the light of the public acceptance of other risks of everyday life. This acceptance (when related to risks that could not be reduced or avoided entirely) is motivated by the benefits that would not otherwise be received, by an assessment of the social cost of achieving a possible reduction of risk, or by an implicit judgment that the risk is negligible.

(118) The acceptable level of risk for stochastic phenomena for members of the general public may be inferred from consider-

ation of risks that an individual can modify to only a small degree and which, like radiation safety, may be regulated by national ordinance. An example of such risks is that of using public transport. From a review of available information related to risks regularly accepted in everyday life, it can be concluded that the level of acceptability for fatal risks to the general public is an order of magnitude lower than for occupational risks. On this basis, a risk in the range of 10^{-6} to 10^{-5} per year would be likely to be acceptable to any individual member of the public.

(119) The assumption of a total risk of the order of 10^{-2} Sv⁻¹ (see paragraph 60) would imply the restriction of the lifetime dose to the individual member of the public to a value that would correspond to 1 mSv per year of life-long whole body exposure. For the reasons given in the following paragraphs, the Commission's recommended whole body dose-equivalent limit of 5 mSv (0.5 rem) in a year, as applied to critical groups, has been found to provide this degree of safety and the Commission recommends its continued use under the conditions specified in paragraphs 120–128.

(120) The application of an annual dose-equivalent limit of 5 mSv to individual members of the public is likely to result in average dose equivalents of less than 0.5 mSv, provided that the practices exposing the public are few and cause little exposure outside the critical groups (see paragraph 85). When applying the dose-equivalent limit for members of the public, consideration must be given to the possibility that some individuals may belong to more than one critical group. In fact, due to the maximizing assumptions usually made in selecting critical groups, the doses actually received by the most highly exposed individuals will in most cases be considerably lower than the doses postulated for the critical group.

(121) The actual value of the average dose depends upon both the result of the optimization process of a large number of justified practices causing radiation exposures and the value selected to limit the individual doses. If at some time in the future the combined exposure resulting from optimized exposures resulted in average dose equivalents higher than 1 mSv in a year, the situation might still be justifiable, even though the average risk

for members of the public would be higher than the range 10^{-6} to 10^{-5} per year.

(122) In some cases optimization, which could be the overriding protection criterion, will necessitate a more realistic selection of the critical group, because to overestimate the risk would introduce a bias which would invalidate the optimization. In these cases, the dose-equivalent limit to individual members of the public, referred to in paragraph 120, would still adequately restrict the average dose equivalent, but the few individuals exposed to the dose-equivalent limit could run a risk in the range of 10^{-5} to 10^{-4} per year. This annual risk would then be one order of magnitude higher than the risk range quoted in paragraph 118. Since exposures at the dose-equivalent limit are not likely to be repeated over many years, however, an adequate restriction of the life-time dose is still likely to be achieved. In any rare cases where the doses to a few individuals were actually found to be received at high rates over prolonged periods, it would be prudent to take measures to restrict their life-time dose as implied in paragraph 119.

(123) In such cases, where optimization is the deciding protection criterion and where critical groups are selected on the basis of realistic assumptions, it would be appropriate to use as a boundary condition of any single optimization process an individual dose limitation such as indicated in paragraph 122. Even the few individuals who would actually receive doses of this magnitude over a long period of time would not be subject to risks appreciably above the level that is generally considered acceptable (see paragraph 118).

(124) In most applications, however, administrative planning is based on operational limits derived from very crude cost-effectiveness considerations of protective measures and quite conservative estimates of the resulting doses in critical groups. For such planning it is usually quite appropriate to assess the doses in the critical group in relation to the dose-equivalent limit of 5 mSv in a year, and the Commission has found no reason to recommend a change in this practice. The main reason for this decision is that the Commission's system of dose limitation, with the dose-equivalent limit of 5 mSv in a year, is found to provide the necessary degree of

safety, under the boundary conditions stated in paragraphs 120–123, and that a change in the upper limit would not be of the same importance from the radiation protection point of view as the rigorous application of the principle of keeping all doses as low as is reasonably achievable.

(125) For non-homogeneous exposures with organ doses H_T , the limitation condition is $\sum_T w_T H_T \leq H'_{wb,L}$, where $H'_{wb,L}$ is the dose-equivalent limit for uniform irradiation of the whole body for a member of the public. Due to different age distributions when members of the public are irradiated, the risks will not be quite the same as for occupational exposures. The *relative* risks, however, will differ but little and the Commission has not found the differences significant enough to recommend a separate set of weighting factors, different from the values of w_T given in paragraph 105 (see also paragraph 60).

(126) In order to prevent any one organ or tissue from receiving a total dose which could contribute significantly to the induction of non-stochastic effects, an overriding annual dose-equivalent limit of 50 mSv should apply. This limit is considerably lower than the corresponding limits recommended for prevention of non-stochastic effects after occupational exposure. The intention is to ensure that the longer exposure period and the practical difficulties in controlling the total exposure from all sources will not result in threshold doses for non-stochastic effects being reached. The weighting procedure recommended in paragraph 125 will generally provide the necessary limitation also for non-stochastic effects, but the special limitation is needed for the skin and the lens of the eye.

(127) In the calculation of the dose equivalent incurred by members of the public from intake of radionuclides, account must be taken of differences in organ size or metabolic characteristics of children. Data on such differences may be found in the report of the task group on Reference Man (*ICRP Publication 23*).

(128) As with workers (see paragraph 102) an increase in the average dose to members of the public could result from any large increase in the number of sources of exposure, even though each satisfactorily met the criteria of justification and optimization (see para-

graphs 68–76) and caused no exposures above the recommended limits. National and regional authorities should therefore keep under surveillance the separate contributions from all practices to the average exposure of the whole population so as to ensure that no single source or practice contributes an unjustified amount to the total exposure, and that no individual receives undue exposure as a result of membership of a number of critical groups.

Exposure of populations

(129) In its previous recommendations the Commission suggested a genetic dose limit, although it expressed reservations about proposing generally applicable dose limits for populations. It has become increasingly clear that the previously suggested level is not likely to be reached, and it is very improbable that responsible authorities would permit the average dose equivalent in a population to reach values that are more than small fractions of the former genetic dose limit of 5 rem in 30 years. Therefore, continuance of the former genetic dose limit could be regarded as suggesting the acceptability of a higher population exposure than is either necessary or probable, and a higher risk than is justified by any present or easily envisaged future development. Furthermore, knowledge gained over the past two decades indicates that genetic effects, while important, are unlikely to be of overriding importance, and would need to be related to the sum of all other effects.

(130) In these recommendations, therefore, the Commission does not propose dose limits for populations. Instead it wishes to emphasize that each man-made contribution to population exposure has to be justified by its benefits, and that limits for individual members of the public refer to the total dose equivalent received from all sources (except as already noted). The limit for irradiation of a whole population is thus clearly seen as the total reached by a summation of minimum necessary contributions, and not as a permissible total apparently available for apportionment. Consideration of the principles enunciated in paragraphs 117–125, together with the Commission's system of dose limitation, is likely to ensure that the average dose equivalent to the population will not exceed 0.5 mSv per year.

(131) In any cost–benefit assessment of a given practice (see paragraphs 60 and 68–76) the specification of the collective dose equivalent is a relatively straightforward procedure if the dose equivalent is limited to the population that shares the benefit of the source. If the exposure will involve other populations (or groups, etc.) the total collective dose equivalent should be kept below that which would have applied had the cost–benefit assessment been confined to the population that receives the benefit.

(132) For the purposes envisaged in paragraph 128 it would be useful to develop long-term forecasts of the trend of the various contributions to the total collective dose equivalent delivered from various sources of exposure. This would facilitate discussions of the appropriate national and international arrangements in regard to justification for particular contributions to population exposures. Contributions that are not primarily controllable nationally, or on an agreed regional basis, call for international agreement on levels of exposure or collective dose equivalents from various sources. In order to achieve these objectives, such authorities may find it appropriate to apply collective dose-equivalent limits for various practices or sources of exposure.

Accidents and emergencies

(133) Under conditions in which accidental exposures occur, questions arise as to what remedial actions may be available to limit the subsequent dose. In such cases, the hazard or social cost involved in any remedial measure must be justified by the reduction of risk that will result. Because of the great variability of the circumstances in which remedial action might be considered, it is not possible for the Commission to recommend “intervention levels” that would be appropriate for all occasions. However, with certain types of accident that are to some extent foreseeable it may be possible to gauge, by an analysis of the costs of the accident and of remedial action, levels below which it would *not* be appropriate to take action.

(134) The decision to initiate remedial action will have to take account of the particular circumstances that prevail. In general it will be appropriate to institute countermeasures only when their social cost and risk will be

less than those resulting from further exposure. Nevertheless, at the operational level (at a reactor establishment, for example) those responsible for the health and safety of workers, and of individuals outside the establishment, will need to have an emergency plan, including dose levels at which various countermeasures would have to be considered. The setting of such levels for particular circumstances is considered to be the responsibility of the national authorities. (See also section G.)

(135) It has been indicated in paragraph 81 that the Commission's dose-equivalent limits are agreed values, established so as to make it possible to plan and design operations

involving foreseen but acceptable radiation exposures. These limits are intended to apply only to those conditions where the source of exposure is under control, and relate to average workers without taking any account of age or child expectancy. The Commission's recommended limits are set at a level which is thought to be associated with a low degree of risk; thus, unless a limit were to be exceeded by a considerable amount, the risk would still be sufficiently low as not to warrant such countermeasures as would themselves involve significant risks or undue cost. It is therefore clear that it is not obligatory to take remedial action if a dose-equivalent limit has been or might be exceeded (see section G).

APPENDIX B

Drinking Water Standards (Ref, 16)

Section 141.15 Maximum contaminant levels for radium-226, radium-228, and gross alpha particle radioactivity in community water systems.

The following are the maximum contaminant levels for radium-226, radium-228, and gross alpha particle radioactivity:

- (a) Combined radium-226 and radium-228—5 pCi/l.
- (b) Gross alpha particle activity (including radium-226 but excluding radon and uranium)—15 pCi/l.

Section 141.16 Maximum contaminant levels for beta particle and photon radioactivity from man-made radionuclides in community water systems

(a) The average annual concentration of beta particle and photon radioactivity from man-made radionuclides in drinking water shall not produce an annual dose equivalent to the total body or any internal organ greater than 4 millirem/year.

(b) Except for the radionuclides listed in Table A, the concentration of man-made radionuclides causing 4 mrem total body or organ dose equivalents shall be calculated on the basis of a 2 liter per day drinking water intake using the 168 hour data listed in “*Maximum Permissible Body Burdens and Maximum Permissible Concentration of Radionuclides in Air or Water for Occupational Exposure*,” NBS Handbook 69 as amended August 1963, U.S. Department of Commerce. If two or more radionuclides are present, the sum of their annual dose equivalent to the total body or to any organ shall not exceed 4 millirem/year.

TABLE A.—Average annual concentrations assumed to produce a total body or organ dose of 4 mrem/yr

Radionuclide	Critical organ	pCi per liter
Tritium	Total body	20,000
Strontium-90	Bone marrow	8

Section 141.25 Analytical Methods for Radioactivity.

(a) The methods specified in *Interim Radiochemical Methodology for Drinking Water*, Environmental Monitoring and Support Laboratory, EPA-600/4-75-008, USEPA, Cincinnati, Ohio 45268, or those listed below, are to be used to determine compliance with § § 141.15 and 141.16 (radioactivity) except in cases where alternative methods have been approved in accordance with § 141.27.

(1) Gross Alpha and Beta—Method 302 “Gross Alpha and Beta Radioactivity in Water” *Standard Methods for the Examination of Water and Wastewater*, 13th Edition, American Public Health Association, New York, N.Y., 1971.

- (2) Total Radium—Method 304 “Radium in Water by Precipitation” Ibid.
- (3) Radium-226—Method 305 “Radium-226 by Radon in Water” Ibid.
- (4) Strontium-89, 90—Method 303 “Total Strontium and Strontium-90 in Water” Ibid.
- (5) Tritium—Method 306 “Tritium in Water” Ibid.
- (6) Cesium-134—ASTM D-2459 “Gamma Spectrometry in Water,” 1975 *Annual Book of ASTM Standards, Water and Atmospheric Analysis*, Part 31, American Society for Testing and Materials, Philadelphia, PA. (1975).
- (7) Uranium—ASTM D-2907 “Microquantities of Uranium in Water by Fluorometry,” Ibid.

(b) When the identification and measurement of radionuclides other than those listed in paragraph (a) is required, the following references are to be used, except in cases where alternative methods have been approved in accordance with § 141.27.

(1) *Procedures for Radiochemical Analysis of Nuclear Reactor Aqueous Solutions*, H. L. Krieger and S. Gold, EPA-R4-73-014. USEPA, Cincinnati, Ohio, May 1973.

(2) *HASL Procedure Manual*, Edited by John H. Harley. HASL 300, ERDA Health and Safety Laboratory. New York, N.Y., 1973.

(c) For the purpose of monitoring radioactivity concentrations in drinking water, the required sensitivity of the radioanalysis is defined in terms of a detection limit. The detection limit shall be that concentration which can be counted with a precision of plus or minus 100 percent at the 95 percent confidence level (1.96σ where σ is the standard deviation of the net counting rate of the sample).

(1) To determine compliance with § 141.15(a) the detection limit shall not exceed 1 pCi/l. To determine compliance with § 141.15(b) the detection limits shall not exceed 3 pCi/l.

(2) To determine compliance with § 141.16 the detection limits shall not exceed the concentrations listed in Table B.

TABLE B—DETECTION LIMITS FOR MAN MADE BETA PARTICLE AND PHOTON EMITTERS

<i>Radionuclide</i>	<i>Detection limit</i>
Tritium	1,000 pCi/l.
Strontium-89	10 pCi/l
Strontium 90	2 pCi/l.
Iodine 131	1 pCi/l.
Cesium-134	10 pCi/l.
Gross beta	4 pCi/l.
Other radionuclides	1/10 of the applicable limit.

(d) To judge compliance with the maximum contaminant levels listed in sections 141.15 and 141.16, averages of data shall be used and shall be rounded to the same number of significant figures as the maximum contaminant level for the substance in question.

Section 141.26 Monitoring Frequency for Radioactivity in Community water systems.

(a) Monitoring requirements for gross alpha particle activity, radium-226 and radium-228.

(1) Initial sampling to determine compliance with § 141.15 shall begin within two years of the effective date of these regulations and the analysis shall be completed within three years of the effective date of these regulations. Compliance shall be based on the analysis of an annual composite of four consecutive quarterly samples or the average of the analyses of four samples obtained at quarterly intervals.

(i) A gross alpha particle activity measurement may be substituted for the required radium-226 and radium-228 analysis *Provided*, That the measured gross alpha particle activity does not exceed 5 pCi/l at a con-

fidence level of 95 percent (1.65σ where σ is the standard deviation of the net counting rate of the sample). In localities where radium-228 may be present in drinking water, it is recommended that the State require radium-226 and/or radium-228 analyses when the gross alpha particle activity exceeds 2 pCi/l.

(ii) When the gross alpha particle activity exceeds 5 pCi/l, the same or an equivalent sample shall be analyzed for radium-226. If the concentration of radium-226 exceeds 3 pCi/l the same or an equivalent sample shall be analyzed for radium-228.

(2) For the initial analysis required by paragraph (a) (1), data acquired within one year prior to the effective date of this part may be substituted at the discretion of the State.

(3) Suppliers of water shall monitor at least once every four years following the procedure required by paragraph (a) (1). At the discretion of the State, when an annual record taken in conformance with paragraph (a) (1) has established that the average annual concentration is less than half the maximum contaminant levels established by § 141.15, analysis of a single sample may be substituted for the quarterly sampling procedure required by paragraph (a) (1).

(i) More frequent monitoring shall be conducted when ordered by the State in the vicinity of mining or other operations which may contribute alpha particle radioactivity to either surface or ground water sources of drinking water.

(ii) A supplier of water shall monitor in conformance with paragraph (a) (1) within one year of the introduction of a new water source for a community water system. More frequent monitoring shall be conducted when ordered by the State in the event of possible contamination or when changes in the distribution system or treatment processing occur which may increase the concentration of radioactivity in finished water.

(iii) A community water system using two or more sources having different concentrations of radioactivity shall monitor source water, in addition to water from a free-flowing tap, when ordered by the State.

(iv) Monitoring for compliance with § 141.15 after the initial period need not include radium-228 *except when* required by the State, *Provided*, That the average annual concentration of radium-228 has been assayed at least once using the quarterly sampling procedure required by paragraph (a) (1).

(v) Suppliers of water shall conduct annual monitoring of any community water system in which the radium-226 concentration exceeds 3 pCi/l, when ordered by the State.

(4) If the average annual maximum contaminant level for gross alpha particle activity or total radium as set forth in § 141.15 is exceeded, the supplier of a community water system shall give notice to the State pursuant to § 141.31 and notify the public as required by § 141.32. Monitoring at quarterly intervals shall be continued until the annual average concentration no longer exceeds the maximum contaminant level or until a monitoring schedule as a condition to a variance, exemption or enforcement action shall become effective.

(b) Monitoring requirements for man-made radioactivity in community water systems.

(1) Within two years of the effective date of this part, systems using surface water sources and serving more than 100,000 persons and such other community water systems as are designated by the State shall be monitored for compliance with § 141.16 by analysis of a composite of four consecutive quarterly samples or analysis of four quarterly samples. Compliance with § 141.16 may be assumed without further analysis if the average annual concentration of gross beta particle activity is less than 50 pCi/l and if the average annual concentrations of tritium and strontium-90 are less than

those listed in Table A, *Provided*, That if both radionuclides are present the sum of their annual dose equivalents to bone marrow shall not exceed 4 millirem/year.

(i) If the gross beta particle activity exceeds 50 pCi/l, an analysis of the sample must be performed to identify the major radioactive constituents present and the appropriate organ and total body doses shall be calculated to determine compliance with § 141.16.

(ii) Suppliers of water shall conduct additional monitoring, as ordered by the State, to determine the concentration of man-made radioactivity in principal watersheds designated by the State.

(iii) At the discretion of the State, supplies of water utilizing only ground waters may be required to monitor for man-made radioactivity.

(2) For the initial analysis required by paragraph (b) (1) data acquired within one year prior to the effective date of this part may be substituted at the discretion of the State.

(3) After the initial analysis required by paragraph (b) (1) suppliers of water shall monitor at least every four years following the procedure given in paragraph (b) (1).

(4) Within two years of the effective date of these regulations the supplier of any community water system designated by the State as utilizing waters contaminated by effluents from nuclear facilities shall initiate quarterly monitoring for gross beta particle and iodine-131 radioactivity and annual monitoring for strontium-90 and tritium.

(i) Quarterly monitoring for gross beta particle activity shall be based on the analysis of monthly samples or the analysis of a composite of three monthly samples. The former is recommended. If the gross beta particle activity in a sample exceeds 15 pCi/l, the same or an equivalent sample shall be analyzed for strontium-89 and cesium-134. If the gross beta particle activity exceeds 50 pCi/l, an analysis of the sample must be performed to identify the major radioactive constituents present and the appropriate organ and total body doses shall be calculated to determine compliance with § 141.16.

(ii) For iodine-131, a composite of five consecutive daily samples shall be analyzed once each quarter. As ordered by the State, more frequent monitoring shall be conducted when iodine-131 is identified in the finished water.

(iii) Annual monitoring for strontium-90 and tritium shall be conducted by means of the analysis of a composite of four consecutive quarterly samples or analysis of four quarterly samples. The latter procedure is recommended.

(iv) The State may allow the substitution of environmental surveillance data taken in conjunction with a nuclear facility for direct monitoring of man-made radioactivity by the supplier of water where the State determines such data is applicable to a particular community water system.

(5) If the average annual maximum contaminant level for man-made radioactivity set forth in § 141.16 is exceeded, the operator of a community water system shall give notice to the State pursuant to § 141.31 and to the public as required by § 141.32. Monitoring at monthly intervals shall be continued until the concentration no longer exceeds the maximum contaminant level or until a monitoring schedule as a condition to a variance, exemption or enforcement action shall become effective.

Section 141.27 Alternative analytical techniques.

With the written permission of the State, concurred in by the Administrator of the U.S. Environmental Protection Agency, an alternative analytical technique may be employed. An alternative technique shall be acceptable only if it is substantially equivalent to the prescribed test in both precision and accuracy as it relates to the determination of compliance with any maximum contaminant level. The use of the alternative analytical technique shall not decrease the frequency of monitoring required by this part.

Section 141.28 Approved laboratories.

For the purpose of determining compliance with § 141.21 through § 141.27, samples may be considered only if they have been analyzed by a laboratory approved by the State except that measurements for turbidity and free chlorine residual may be performed by any person acceptable to the State.

Section 141.29 Monitoring of consecutive public water systems.

When a public water system supplies water to one or more other public water systems, the State may modify the monitoring requirements imposed by this part to the extent that the interconnection of the systems justifies treating them as a single system for monitoring purposes. Any modified monitoring shall be conducted pursuant to a schedule specified by the State and concurred in by the Administrator of the U.S. Environmental Protection Agency.

Impact of Maximum Contaminant Levels for Man-made Radionuclides

Though man-made radioactivity in public water systems is sometimes a matter of concern it is important to recognize that unlike the case for radium, current ambient concentrations are less than the proposed limits because of regulatory concern for these radionuclides. Drinking water is not a major pathway for exposure from nuclear power plants. The Agency has reviewed all the Environmental Impact Statements for power reactors currently available. Based on the design of these reactors the estimated total body doses due to drinking water served by public water systems from these facilities range from 0.00001 to 0.3 millirem per year with 90% of the expected doses less than 0.04 millirem per year. The average total body dose is 0.3 millirem per year. Thyroid doses are somewhat larger, ranging from 0.0003 to 0.8 millirem per year, with an average annual dose of 0.08 millirem per year.

Data on ambient levels in public water systems indicate that almost all of the radioactivity in the aquatic environment is due to residual radioactivity from nuclear weapons testing. The historical trend of radioactivity in the Great Lakes and in other waterways shows this source of radioactivity is diminishing (9).

The maximum contaminant level for man-made radionuclides is expressed in terms of the annual dose rate (millirem per year) from continuous ingestion. Specifying maximum contaminant levels in terms of radioactivity concentration (pCi per liter) was considered but rejected in view of the short length of time such limits would be appropriate, since presently available dose conversion factors for ingested radioactivity are obsolescent and the ICRP is developing new dose models. When appropriate models for doses due to environmental contamination become available, the Agency will revise the Interim Regulations to permit the use of newer data. The concentrations yielding 4 millirem annually, given in Appendix IV, are based on NBS Handbook 69 as required by the Interim Regulations, 41 FR 133, p. 28402, July 9, 1976.

Monitoring for Radioactivity in Community Water Systems

The Agency has developed monitoring requirements for radioactivity with two ends in view. Information must be available to the supplier so he can control the quality of the water he serves. However, the cost of the monitoring should not result in an undue economic burden in terms of other financial requirements for safe operation of the system. To an extent these are conflicting requirements since more information can always be purchased for more money. The Agency has tried to limit the monitoring to that which is essential for determining compliance with maximum contaminant limits under most conditions. As State capability for effective monitoring is aug-

mented, States are encouraged to introduce more rigorous monitoring of particular supplies because of local knowledge of their potential for radiocontamination. In addition Federal monitoring requirements for radioactivity are limited to community water systems as defined in Section 141.2 of the Interim Regulations. Since the proposed limits are based on lifetime exposure, any radiation risk to transient populations is minimal.

In general, the Interim Regulations call for quarterly sampling. In the case of naturally occurring radioactivity it is often thought that a single sample can be used to determine the average annual concentrations. This is not the case for some ground water sources where the annual discharge cycle of the aquifers has a pronounced effect on radium concentration. In such cases, a single yearly grab sample could show a low concentration, resulting in the acceptance of water containing more than a maximum contaminant level. Conversely, an abnormally high level could lead to the institution of expensive control measures where the annual average concentration is really acceptable. Although sampling at monthly intervals might be advisable in certain locations and situations (and should be required by the State where necessary) the Agency believes quarterly sampling will be sufficient to determine the average annual concentration in most cases. Where the average annual concentration has been shown to be less than one-half the relevant maximum contaminant level, a yearly sampling procedure is permitted by the regulations.

In order to reduce monitoring costs, the Interim Regulations allow composited samples to be radioassayed, usually at yearly intervals. In such cases care must be taken to prevent the loss of activity by means of absorption on container walls. Acidification with 1 milliliter of 16N HNO₃ per liter of sample is a method suggested in "Interim Radiochemical Methodology for Drinking Water" (10). In the case of iodine-131, hydrochloric rather than nitric acid should be used for acidification and sodium bisulfite should be added to the sample. In some cases State laboratories may prefer to count quarterly samples rather than keep track of quarterly aliquots. If so, the estimated costs given below will be exceeded. The increased cost is not large, however, and quarterly measurements are recommended, particularly for the monitoring of gross beta activity.

It should be noted that from the definition of "maximum contaminant level" in the Interim Regulations, section 141.2(c), samples should be collected from free flowing outlets, not at the source of supply water. Since in some cases, several sources may contribute water to the system, samples should be taken at representative points within the system so as to truly reflect the maximum concentration of radioactivity received by users. In cases where more than one source is utilized, suppliers shall monitor source water, in addition to water from a free flowing tap, when ordered by the State.

Although monitoring a typical community water system is relatively inexpensive, less than five dollars per year, the total national cost of monitoring for radium-226, radium-228, and gross alpha particle activity is not trivial because of the large number of supplies involved, 40,000. In order to minimize cost, the Agency is proposing that a water supplier initially obtain a relatively low cost analysis of gross alpha particle activity. In most cases this test will indicate that no significant activity is present and additional tests will not be required. However, when the gross alpha measurement indicates the alpha particle activity may exceed 5 pCi per liter, a further test for radium-226 is required.

Although not in the same decay chain, radium-228 sometimes accompanies radium-226. Only rarely, however, does the radium-228 concentration exceed that of radium-226. Therefore, a radium-228 analysis, which is relatively expensive, is only required when the radium-226 concentration exceeds 3 pCi per liter. In localities where radium-228 may be present in drinking water, it is recommended that the State require radium-226 and/or radium-228 analyses when the gross alpha particle activity exceeds 2 pCi/l.

The Interim Regulations require sampling and measurement at quarterly intervals where the limits are exceeded so that the water supplier can follow the variation of radium concentration through the yearly cycle and thereby institute remedial measures, such as additional dilution or treatment, during periods when concentrations are unusually high. Monitoring at quarterly intervals shall be continued until the annual average concentration no longer exceeds the maximum contaminant level or until a monitoring schedule as a condition to a variance, exemption or enforcement action shall become effective.

Dosimetric Calculations for Man-Made Radioactivity

A. Calculations Based on NBS Handbook 69

The dose rate from radioactivity in drinking water is calculated on the basis of a 2 liter daily* intake. Except for tritium and strontium-90, see below, the concentrations of man-made radionuclides causing 4 millirem per year have been calculated using the data in NBS Handbook 69 (1) and are tabulated in Table IV-2A and IV-2B. The dose models used in preparing Handbook 69 are outlined in reference 2. Maximum Contaminant Levels are defined in terms of the annual dose equivalent to the total body or any internal organ. Handbook 69 lists the critical organ for each radionuclide. Often the total body is listed as the critical organ. The 168 hour maximum permissible concentrations for ingestion in Handbook 69 are not calculated on the basis of the same annual dose to each critical organ as in the Interim Regulations, rather different organ doses are permitted by occupational radiation protection limits (ORL), Table IV-1.

TABLE IV-1. Occupational Radiation Limits

(ORL)	
Critical Organ	ORL (rems)
Total body	5
Gonads	5
Thyroid	30
Bone	29.1 (a)
Other Organs	15

(a) Based on the alpha energy deposited in bone by 0.1 μ Ci of radium-226.

The maximum permissible concentrations for a 168 hour week, MPC, in Handbook 69, assume ingestion at 2.2 liters per day and are in units of μ Ci per cc. The various numerical factors can be combined to find C_4 , the concentration causing 4 mrem per year from 2 liters daily ingestion of drinking water as follows:

$$C_4 = 4.4 \times 10^6 \times \frac{\text{MPC} \dots \text{pCi per liter}}{\text{ORL}}$$

Critical organs are identified by boldface type in Handbook 69 so that an appropriate ORL can be selected from Table IV-1.

To illustrate, a sample calculation, taken from page 24 of Handbook 69 is given:

*The recent ICRP publication #23, "Report of the Task Group on Reference Man," (3) gives the total daily water intake as 3 liters, 1.95 liters by fluid intake, the balance by food and food oxidation. Almost all of the fluid intake is from tap water and water based drinks (Page 360).

$$C_4 = 4.4 \times 10^6 \times \frac{0.02}{15} \frac{\text{pCi}}{\text{l}} = 5867 \text{ pCi/l}$$

$$= 6000 \text{ pCi/l}$$

Rounding is appropriate since the values in Handbook 69 are given to one significant figure.

Calculation of the dose resulting from the ingestion of drinking water containing a known mixture of radionuclides is straightforward. Let $A, B, \dots$ be the concentrations, in pCi per liter, of isotopes $a, b, \dots$ in the water and let $C_4^A(X)$ be the average annual concentrations of isotope A yielding 4 millirem per year to organ X , $C_4^B(X)$ the same quantity for B , etc. The total annual dose to organ X in one year is, then

$$\left[\frac{A}{C_4^A(X)} + \frac{B}{C_4^B(X)} + \dots \right] \times 4 \text{ millirem}$$

Therefore, the 4 millirem limit is not exceeded if

$$\left[\frac{A}{C_4^A(X)} + \frac{B}{C_4^B(X)} + \dots \right] \leq 1.0$$

It should be noted that although limits for the various radionuclides may be based on different critical organs, the resultant dose is additive with respect to a specific organ when the total body is the designated critical organ for one of the radionuclides. For example, consider drinking water which has on an annual basis a strontium-90 concentration of 4 pCi/l and a tritium concentration of 15,000 pCi/l. The annual dose to bone marrow from the strontium-90 is 2 mrem. The total body dose from the tritium is 3 mrem annually. Even though the annual concentration of each contaminant alone is permissible, the total dose to bone marrow is 5 mrem annually and therefore the MCL is exceeded. Tabular values for C_4 for photon and beta emitters are listed in Table IV-2A and IV-2B below.

B. The Dose from Tritium and Strontium-90 in Drinking Water

For the majority of radionuclides, the models given in Handbook 69 to estimate doses to occupationally exposed workers are also appropriate for environmental contaminants. They are not, however, appropriate for all man-made radionuclides, particularly tritium and strontium-90. Concentrations yielding 4 millirem annually for these radionuclides are given in Table A of the Interim Regulations and listed in Table IV-2A.

Some radionuclides are isotopes of elements which are incorporated into organic molecules within the body so that the single exponential excretion models assumed in the development of Handbook 69 underestimate the dose. An example is tritium where two or three exponentials may be needed to describe the dose-time relationship of ingested tritium (4). Some investigators have estimated that following chronic ingestion organically bound tritium may increase the dose by a factor of 1.4 to 1.5 over that predicted by Handbook 69 (5). Such estimates are too high because organically bound tritium irradiates the total body mass, and not just the mass of body water, as assumed in the model used in Handbook 69 (2).

Consideration of the daily intake of hydrogen and water shows that the tritium concentration (specific activity) in any organ is no greater than 120% of the tritium concentration in body water. The concentration of tritium in body water following chronic ingestion is $T/3$ where T is daily intake of tritium in pCi and the total water intake, including that in food, is 3 liters per day (3). Water content by weight of any organ does not exceed 80 percent (4). Therefore, equilibrium concentration of tritium in any organ due to its water content, can not exceed $0.8 T/3 = .267 T$ pCi/kg.

Because of organically bound hydrogen an organ's hydrogen (and tritium) content is greater than that due to water alone. The daily hydrogen intake is .35 kg per day (3) and, since no organ contains more than 11 percent hydrogen by weight (4), the maximum tritium concentration in any organ following chronic ingestion is $.11 \text{ T}/.35 = .314 \text{ T pCi/kg}$. The specific activity of tritium in any organ due to bound and unbound hydrogen exceeds that due to its water content alone by the ratio $.314/.267 = 1.18$. Therefore, the dose to any organ due to organically bound tritium exceeds the dose to body water, given in Handbook 69, by no more than about twenty percent.

The Agency is aware that the ICRP is developing new tritium dose models more suitable for environmental sources of tritium exposure than the model used in Handbook 69. Until these models are published and recommended by the Agency, the maximum contaminant level for tritium is calculated on the basis of 80 percent of the value calculated using NBS Handbook 69.* For tritium in drinking water:

$$C_4 = 0.8 \times 4.4 \times 10^6 \times \frac{0.03}{5} = 21,120 \text{ pCi/l} \\ = 20,000 \text{ pCi/l}$$

The maximum contaminant level for strontium-90 in the Interim Regulations is based on the dose model used by the Federal Radiation Council (FRC) to predict the dose to bone marrow (6). According to the FRC model a continuous daily intake of 200 pCi per day of strontium-90 will result in a body burden of 50 pCi per gram of calcium in bone. The annual dose rate to bone marrow from this body burden would be 50 mrem per year (7). Therefore, continuous ingestion of 16 pCi per day would result in 4 mrem per year, the limit for man-made radionuclides in drinking water. For two liters ingestion of water per day:

$$C_4 = \frac{16 \text{ pCi}}{2} = 8 \text{ pCi/l}$$

C. Concentrations yielding an Annual Dose of 4 Millirem

Tables IV-2A and IV-2B give C_4 the annual average concentrations for man-made radionuclides which are assumed to yield an annual dose of 4 millirem to the indicated organ. Table IV-2A comprises those nuclides having half-lives greater than one day. Table IV-2B contains shorter half-life radionuclides not expected to appear in drinking water supplies. Ingestion at a rate of 2.0 liters per day is assumed. The values shown were calculated from the Maximum Permissible Concentrations listed in Handbook 69 (1) as outlined above.

TABLE IV-2A. Annual Average Concentrations Yielding 4 Millirem per Year for a Two Liter Daily Intake

(Half-life greater than 24 hours)		
Radionuclide	Critical Organ	C_4 (pCi/l)
Tritium	Total Body	20,000
$^4\text{Be}^7$	GI (LLI)	6,000
$^{60}\text{Co}^{14}$	Fat	2,000
$^{11}\text{Na}^{22}$	Total Body	400
$^{15}\text{P}^{32}$	Bone	30
$^{16}\text{S}^{35}$	Testis	500
$^{17}\text{Cl}^{36}$	Total Body	700
$^{20}\text{Ca}^{45}$	Bone	10
$^{20}\text{Ca}^{47}$	Bone	80
$^{21}\text{Sc}^{46}$	GI (LLI)	1,000
$^{21}\text{Sc}^{47}$	GI (LLI)	300

*n.b. In accordance with current guidance to Federal agencies, a quality factor of 1.7, as in Handbook 69, is used in this calculation.

²¹ Sc ⁴⁸	GI (LLI)	80
²³ V ⁴⁸	GI (LLI)	90
²⁴ Cr ⁵¹	GI (LLI)	6,000
²⁵ Mn ⁵²	GI (LLI)	90
²⁵ Mn ⁵⁴	GI (LLI)	300
²⁶ Fe ⁵⁵	Spleen	2,000
²⁶ Fe ⁵⁹	GI (LLI)	200
²⁷ Co ⁵⁷	GI (LLI)	1,000
²⁷ Co ⁵⁸	GI (LLI)	9,000
²⁷ Co ⁶⁰	GI (LLI)	100
²⁸ Ni ⁵⁹	Bone	300
²⁸ Ni ⁶³	Bone	50
³⁰ Zn ⁶⁵	Liver	300
³² Ge ⁷¹	GI (LLI)	6,000
³³ As ⁷³	GI (LLI)	1,000
³³ As ⁷⁴	GI (LLI)	100
³³ As ⁷⁶	GI (LLI)	60
³³ As ⁷⁷	GI (LLI)	200
³⁴ Se ⁷⁵	Kidney	900
³⁵ Br ⁸²	GI (LLI)	100
³⁷ Rb ⁸⁶	Total Body	600
³⁷ Rb ⁸⁷	Pancreas	300
³⁸ Sr ⁸⁵	GI (SI)	21,000
³⁸ Sr ⁸⁹	Bone	20
³⁸ Sr ⁸⁹	Bone Marrow (FRC)	80
³⁸ Sr ⁹⁰	Bone Marrow (FRC)	8
³⁹ Y ⁹⁰	GI (LLI)	60
³⁹ Y ⁹¹	GI (LLI)	90
⁴⁰ Zr ⁹³	GI (LLI)	2,000
⁴⁰ Zr ⁹⁵	GI (LLI)	200
⁴¹ Nb ^{93m}	GI (LLI)	1,000
⁴¹ Nb ⁹⁵	GI (LLI)	300
⁴² Mo ⁹⁹	Kidney	600
⁴³ Tc ⁹⁶	GI (LLI)	300
⁴³ Tc ^{97m}	GI (LLI)	1,000
⁴³ Tc ⁹⁷	GI (LLI)	6,000
⁴³ Tc ⁹⁹	GI (LLI)	900
⁴⁴ Ru ⁹⁷	GI (LLI)	1,000
⁴⁴ Ru ¹⁰³	GI (LLI)	200
⁴⁴ Ru ¹⁰⁶	GI (LLI)	30
⁴⁵ Rh ¹⁰⁵	GI (LLI)	300
⁴⁶ Pd ¹⁰³	GI (LLI)	900
⁴⁶ Pd ¹⁰⁹	GI (LLI)	300
⁴⁷ Ag ¹⁰⁵	GI (LLI)	300
⁴⁷ Ag ^{110m}	GI (LLI)	90
⁴⁷ Ag ¹¹¹	GI (LLI)	100
⁴⁸ Cd ¹⁰⁹	GI (LLI)	600
⁴⁸ Cd ^{115m}	GI (LLI)	90
⁴⁸ Cd ¹¹⁵	GI (LLI)	90
⁴⁹ In ¹¹⁵	GI (LLI)	300
⁵⁰ Sn ¹¹³	GI (LLI)	300
⁵⁰ Sn ¹²⁵	GI (LLI)	60
⁵¹ Sb ¹²²	GI (LLI)	90
⁵¹ Sb ¹²⁴	GI (LLI)	60
⁵¹ Sb ¹²⁵	GI (LLI)	300
⁵² Te ^{125m}	Kidney	600
⁵² Te ^{127m}	Kidney	200
⁵² Te ¹²⁷	GI (LLI)	900
⁵² Te ^{129m}	GI (LLI)	90
⁵² Te ¹²⁹	GI (S)	2,000
⁵² Te ^{131m}	GI (LLI)	200
⁵² Te ¹³²	GI (LLI)	90
⁵³ I ¹²⁶	Thyroid	3
⁵³ I ¹²⁹	Thyroid	1
⁵³ I ¹³¹	Thyroid	3
⁵⁵ Cs ¹³¹	Total Body	20,000
⁵⁵ Cs ¹³⁴	GI (S)	20,000
⁵⁵ Cs ¹³⁵	Total Body	900
⁵⁵ Cs ¹³⁶	Total Body	800

⁵⁵ Ce ¹³⁷	Total Body	200
⁵⁶ Ba ¹³¹	GI(LLI)	600
⁵⁶ Ba ¹⁴⁰	GI(LLI)	90
⁵⁷ La ¹⁴⁰	GI(LLI)	60
⁵⁸ Ce ¹⁴¹	GI(LLI)	300
⁵⁸ Ce ¹⁴³	GI(LLI)	100
⁵⁹ Pr ¹⁴³	GI(LLI)	100
⁶¹ Pm ¹⁴⁹	GI(LLI)	100
⁶² Sm ¹⁵¹	GI(LLI)	1,000
⁶² Sm ¹⁵³	GI(LLI)	200
⁶³ Eu ¹⁵²	GI(LLI)	60
⁶³ Eu ¹⁵⁴	GI(LLI)	200
⁶³ Eu ¹⁵⁵	GI(LLI)	600
⁶⁴ Gd ¹⁵³	GI(LLI)	600
⁶⁵ Tb ¹⁶⁰	GI(LLI)	100
⁶⁶ Dy ¹⁶⁶	GI(LLI)	100
⁶⁷ Ho ¹⁶⁶	GI(LLI)	90
⁶⁸ Er ¹⁶⁹	GI(LLI)	300
⁶⁹ Tm ¹⁷⁰	GI(LLI)	100
⁶⁹ Tm ¹⁷¹	GI(LLI)	1,000
⁷⁰ Yb ¹⁷⁵	GI(LLI)	300
⁷¹ Lu ¹⁷⁷	GI(LLI)	300
⁷² Hf ¹⁸¹	GI(LLI)	200
⁷³ Ta ¹⁸²	GI(LLI)	100
⁷⁴ W ¹⁸¹	GI(LLI)	1,000
⁷⁴ W ¹⁸⁵	GI(LLI)	300
⁷⁵ Re ¹⁸³	GI(LLI)	2,000
⁷⁵ Re ¹⁸⁶	GI(LLI)	300
⁷⁵ Re ¹⁸⁷	GI(LLI)	9,000
⁷⁶ Os ¹⁸⁵	GI(LLI)	200
⁷⁶ Os ¹⁹¹	GI(LLI)	600
⁷⁶ Os ¹⁹³	GI(LLI)	200
⁷⁷ Ir ¹⁹⁰	GI(LLI)	600
⁷⁷ Ir ¹⁹²	GI(LLI)	100
⁷⁸ Pt ¹⁹¹	GI(LLI)	300
⁷⁸ Pt ^{193m}	GI(LLI)	3,000
⁷⁸ Pt ¹⁹³	Kidney	3,000
⁷⁸ Pt ¹⁹⁷	GI(LLI)	300
⁷⁹ Au ¹⁹⁶	GI(LLI)	600
⁷⁹ Au ¹⁹⁸	GI(LLI)	100
⁸¹ Tl ²⁰⁴	GI(LLI)	300
⁸² Pb ²⁰³	GI(LLI)	1,000
⁸³ Bi ²⁰⁶	GI(LLI)	100
⁸³ Bi ²⁰⁷	GI(LLI)	200
⁹¹ Pa ²³³	GI(LLI)	300

TABLE IV - 2B
Annual Average Concentrations Yielding 4 Millirem
per Year for a Two Liter Daily Intake
(Half-life less than 24 hours)

<i>Radionuclide</i>	<i>Critical Organ</i>	<i>C_i</i> (pCi/I)
⁹ F ¹⁸	GI(SI)	2,000
¹⁴ Si ³¹	GI(S)	3,000
¹⁷ Cl ³⁸	GI(S)	1,000
¹⁹ K ⁴²	GI(S)	900
²⁵ Mn ⁵⁶	GI(LLI)	300
²⁷ Co ^{58m}	GI(LLI)	300
²⁸ Ni ⁶⁵	GI(LLI)	300
²⁹ Cu ⁶⁴	GI(LLI)	900
³⁰ Zn ^{69m}	GI(LLI)	200
³⁰ Zn ⁶⁹	GI(S)	6,000
³¹ Ga ⁷²	GI(LLI)	100
³⁸ Sr ^{85m}	Total Body	900
³⁸ Sr ⁹⁰	GI(LLI)	200
³⁸ Sr ⁹²	GI(ULI)	200
³⁹ Y ^{91m}	GI(SI)	9,000
³⁹ Y ⁹²	GI(ULI)	200
³⁹ Y ⁹³	GI(LLI)	90

⁴⁰ Zr ⁹⁷	GI (LLI)	60
⁴¹ Nb ⁹⁷	GI (ULI)	3,000
⁴³ Tc ^{96m}	GI (LLI)	30,000
⁴³ Tc ^{99m}	GI (ULI)	20,000
⁴⁴ Rh ¹⁰⁵	GI (ULI)	300
⁴⁵ Rh ^{103m}	GI (S)	30,000
⁴⁹ In ^{113m}	GI (ULI)	3,000
⁴⁹ In ^{114m}	GI (LLI)	60
⁴⁹ In ^{115m}	GI (ULI)	1,000
⁵³ I ¹³²	Thyroid	90
⁵³ I ¹³³	Thyroid	10
⁵³ I ¹³⁴	Thyroid	100
⁵³ I ¹³⁵	Thyroid	30
⁵⁵ Cs ^{134m}	Total Body	80
⁵⁹ Pt ¹⁴²	GI (LLI)	90
⁶⁰ Nd ¹⁴⁹	GI (LLI)	900
⁶³ Eu ¹⁵²	GI (LLI)	200
⁶⁴ Gd ¹⁵⁹	GI (LLI)	200
⁶⁶ Dy ¹⁶⁵	GI (LLI)	1,000
⁶⁸ Er ¹⁷¹	GI (ULI)	300
⁷⁴ W ¹⁸⁷	GI (LLI)	200
⁷⁵ Re ¹⁸⁸	GI (LLI)	200
⁷⁶ Os ^{191m}	GI (LLI)	9,000
⁷⁷ Ir ¹⁹⁴	GI (LLI)	90
⁷⁸ Pt ^{197m}	GI (ULI)	3,000
⁸¹ Tl ²⁰²	GI (LLI)	300

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I.E. Basic Radiation Detection Methods

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E. BASIC RADIATION DETECTION METHODS

1. Types of Ionizing Radiation

Radiation types are briefly reviewed in this section, leaving discussion of their origin or of the structure of matter to standard texts such as those of Evans (Ref. 1), Harvey (Ref. 2), Segrè (Ref. 3), and Marmier and Sheldon (Ref. 4).

a. Particulate

i. Alpha (α) particles are helium nuclei (${}^4\text{He}^{++}$) principally emitted by heavy radionuclides such as naturally occurring uranium and thorium and their daughters. The kinetic energy of the alpha particles generally lies between 2-7 MeV and they are typically emitted with discrete energies characteristic of the radionuclide.

ii. Beta (β) particles are high energy electrons or positrons created and emitted by certain radionuclides. Beta particles are emitted with a continuous spectrum (e.g., see Fig. 1) and tables of β -energies commonly list the maximum energy of emission that is characteristic of the particular radionuclide. For many purposes, the approximation of a mean energy equal to one-third of the maximum energy is satisfactory.

iii. Neutrons (n) are uncharged particles that — together with protons — are the major constituents of nuclei. They have no Coulomb interaction with atomic electrons or nuclear protons.

(1) Thermal neutrons have the same average kinetic energy (and therefore the same temperature) as atoms in the surrounding matter. At room temperature, this corresponds to a Maxwellian distribution with an average kinetic energy of 0.025 eV.

(2) Intermediate neutrons are often taken to be those with energy from 0.5 eV to 10 keV. Neutrons having energies less than 100 eV are sometimes referred to as slow neutrons.

(3) Fast neutrons are generally defined as those having energies between 10 keV and 10 MeV.

(4) Relativistic neutrons have energies greater than about 10 MeV, so that the mass equivalent of the kinetic energy becomes a significant fraction of the total mass-energy of the neutrons, and relativistic corrections should be applied in analyses of neutron interactions.

b. Electromagnetic. Both X- and gamma (γ) rays are electromagnetic radiation of very short wavelength, making them no different

than visible light except they are constituted of photons of much higher energy. Gamma rays are emitted with discrete energies characteristic of the radionuclide. X-ray emission is of two types, characteristic radiation (discrete energies) and continuous spectra (such as bremsstrahlung). Characteristic X-rays are emitted because of transitions between energy levels of the inner electrons in an atom, while γ -rays are produced by transitions from higher to lower energy states within the same nucleus (as opposed to α and β decay, which yield a different radionuclide upon decay).

2. Interactions of Ionizing Radiation with Matter

Radiation can cause ionization either directly or indirectly.

a. Directly Ionizing Radiations.

These include all charged particles of sufficient energy to produce ionization in matter. This includes α -particles, heavier ions, and β -particles. Such particles lose energy by interaction with the orbital electrons or nuclei of atoms in the materials they traverse. These collisions may be either elastic (Rutherford scattering) or inelastic. Any or all of the possible collision processes may take place, reducing the energy and altering the direction of the particles. (Nuclear reactions and strong interactions require the particle to overcome the nuclear Coulomb barrier, usually tens of MeV, and will be ignored because the energies of the principal radioactivities with which we are concerned are much lower.) Since atoms are bigger than nuclei, collision with the electrons are far more probable. Orbital electrons in atoms or molecules may be excited by these collisions with the emission of electromagnetic radiation from subsequent return to ground state. Alternatively the atom may be ionized, resulting in the creation of an ion pair. In air an average of 34 eV is expended in the creation of each ion pair. If the charged particle is much heavier than the electron, its direction will not be appreciably altered by any interaction and will travel through matter in almost straight lines. The electrons that are liberated (sometimes called delta rays) may have sufficient kinetic energy to induce further ionization or excitation in atoms along their paths.

Since heavy charged particles move in almost straight lines and lose energy in a large number of small steps (allowing fluctuations in energy loss to "average out"), they have definite ranges. The range of such a particle is a function of its initial energy, its mass and charge, and of the properties of the stopping medium. The particle's energy loss per unit path length is given in classical terms as (Refs. 1-4):

Fig. 1. The beta ray spectrum and average beta energy of selected isotopes.
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$$-\frac{dE}{dx} \approx \frac{8\pi z^2 e^4 N Z}{mv^2} \ln \frac{2mv^2}{I},$$

where z = charge of particle
 v = velocity of particle
 m = mass of electron
 N = number atoms/cm³ in stopping medium
 Z = atomic number of stopping medium
 I = average energy required for ionization.

This equation can be represented in the following approximate form

$$-\frac{dE}{dx} \propto \frac{Mz^2}{E}.$$

Where E = energy of moving particles
 M = mass of moving particles.

Thus the rate of energy loss increases as the particle slows down, so that the density of ionization along the path increases. However, when its velocity is very low, the particle can capture an electron from the stopping medium. It may lose the electron later, but the average value of z will decrease causing $-dE/dx$ to decrease, and the ionization becomes less dense. Figure 2, which is called a Bragg curve, shows the density of ionization along the path of an α particle moving in air.

Fig. 2. Bragg curve of specific ionization by alpha particles in air at standard temperature and pressure (from H. Cember, Ref. 17, with permission by Pergamon Press, Ltd. © 1969).

Although electrons induce ionization and excitation in stopping media in the same way as heavy charged particles, an electron can lose up to half its energy in a single encounter, and may also suffer considerable deflection. Therefore, because energy is lost in a random process and the paths of electrons

are not straight lines, their range is not well defined. (Another source of energy loss in a stopping medium is emitted bremsstrahlung, but since it is only important at high energies, we do not treat it.)

Because β -decay yields a continuous energy spectrum, even when the maximum energy is high, the spectrum will contain low-energy electrons that a thin absorber can stop. Thus a thin absorber can cause a reduction in the measurement by electron flux. For monoenergetic heavy particles, a reduction does not occur until the absorber thickness approaches the range of the particles.

As a positron slows, it may interact with an electron in an annihilation wherein their total mass energy is converted to electromagnetic radiation.

b. Indirectly Ionizing Radiation.

This includes some types of electromagnetic radiations and neutrons. These radiations interact with matter by giving rise to secondary radiation which is ionizing. Indirect ionizing radiations lose energy by interaction with electrons, or atomic nuclei, and the charged particles resulting collide in turn with the orbital electrons or nuclei.

i. Electromagnetic Radiation.

There are three main processes of importance in the interaction of photons with matter: the photoelectric effect, the Compton effect, and pair production.

The most important energy loss mechanism for low-energy photons in the range of hundreds of electron volts is the photoelectric effect. In this process, the low energy photon interacts with a bound electron in an orbital shell of the atom and disappears by the transfer of its entire energy to the electron, which is then ejected from the atom as a photoelectron. This photoelectron then produces most of the ion pairs formed. The photon energy must be greater than the binding energy of the electron in order for the photoelectric effect to occur. The cross section for this process decreases with increasing photon energy until at high energy it no longer plays a significant role. However, the cross section increases with increasing atomic number (Z) of the absorber. Very roughly, the photoelectric absorption follows the equation:

$$\text{Probability of photoelectric absorption} \propto \frac{Z^5}{E^{7/2}}$$

Therefore, it is most important for the absorption of low energy γ -rays in heavy elements. While the photon energy must be higher than the binding energy, the process is most likely to occur when the photon energy is only slightly higher than the binding energy. The most likely electron to be removed is the most tightly bound one in the atom, i.e. the

K electron, provided the photon energy is sufficient to remove it from its orbit.

In the Compton effect, the incident photon undergoes an elastic collision with a free or loosely bound electron, thereby giving up energy. The probability of this process depends on the number of electrons present in the material. It is the dominant absorption process for intermediate energy γ -rays: 1-5 MeV for lead, 0.1 - 15 MeV for aluminum. The Compton cross section decreases monotonically with increasing photon energy.

Pair production occurs for photon energies exceeding 1.02 MeV. Then the photon may interact with the Coulomb field of the nucleus or, less frequently, with that of an electron to produce a positron-electron pair (the inverse of positron-electron annihilation). The energy of the photon in excess of 1.02 MeV appears as the kinetic energy of the two particles created. The cross section varies approximately as Z^2 and increases with photon energy to become the predominant mode of interaction at about 10 MeV for elements of high atomic number (Z) (e.g., lead) and at much higher energies for low Z elements (e.g. aluminum).

Figure 3 illustrates how these three processes interact throughout a large range of photon energies.

Fig. 3. Linear absorption coefficients (in cm^{-1}) versus gamma-ray energy for silicon, germanium, and sodium iodide (from Capellani and Restelli, Ref. 18, with permission by The Institute of Physics © 1968).

ii. Neutrons. Neutrons interact with matter in quite different processes from that of other ionizing radiation, in each case reacting directly with nuclei:

● **Elastic scattering:** The neutron shares its initial kinetic energy with the nucleus, which recoils without being excited. The smaller the mass of the nucleus, the larger the fraction of kinetic energy given to it.

● **Inelastic scattering:** The neutron collides with a nucleus which is left in an excited state, and the neutron emerges with a reduced energy.

● **Capture:** The neutron is captured by a nucleus, forming a compound nucleus that is usually excited and emits γ -rays with typically a total energy of about 6-18 MeV.

● **Particle emission:** The neutron interacts with a nucleus, causing the emission of particles such as protons, neutrons, or α 's. Since outgoing charged particles must overcome the Coulomb barrier, this type of reaction is most probable for light nuclei (with a small Coulomb barrier) and fast neutrons (with a large amount of energy).

● **Fission:** The compound nucleus that is formed with the incoming neutron splits into two lighter nuclei (fission fragments) with the emission of, on the average, about two neutrons. Fission in ^{235}U , ^{239}Pu , and ^{233}U may even be induced with thermal neutrons. It may be induced in a larger group of heavy nuclides by fast neutrons.

3. Basic Detection Methods

The detection of ionizing radiation may depend on: 1) the collection of the ions formed; 2) the amplification of ionization to give a more detectable pulse; 3) the fluorescence of a phosphor that has absorbed radiation; 4) a chemical reaction that is radiation induced; or, more rarely, 5) the measurement of the heat produced by the absorption of radiation. We shall restrict our discussion primarily to the first three because most commercial instruments for environmental and source monitoring are based upon them. The ionizing radiation field to be detected and measured may be characterized by four parameters: radiation type, its energy, flux or rate of arrival, and time/spatial distribution. Usually the type of radiation to be detected is known and determines the type of instrument required. The following discussion is based on generic instrument types. For more detailed information, the reader is referred to Divisions III and IV and to the extensive literature on radiation detectors (Refs. 5-9, and 12-15.)

a. Gas-Filled Detectors. Gas-filled detectors are based upon one or both of the first two processes mentioned above. A voltage is placed across two physically separated electrodes, and the region between them is filled with an easily ionized gas. Ions formed therein are collected by the electrodes and read as a current or an electronic pulse, depending upon the associated electronics. The number of ions initially formed (by the interaction processes mentioned previously) is generally proportional to the energy of the radiation.

The three common gas-filled detectors are referred to as the ionization chamber, proportional counter, and Geiger-Müller (GM) tubes. Their difference can be explained with the aid of Fig. 4. At voltages below Region I there is competition between the loss of ion pairs by recombination and their collection on the electrodes. With an increasing electric field the drift velocity of the ions increases, decreasing the time for recombination, thereby causing the fraction of ion pairs collected to become larger. In Region I (saturation region or ionization chamber region) the recombination loss is negligible and the charge collected is directly related to the number of ion pairs formed. In Region II, the collected charge is increased by a factor through the phenomenon of gas multiplication: electrons released in the primary ionization are accelerated sufficiently to produce additional ionization adding to the collected charge. At the onset of Region II, the multiplication factor for a given voltage is independent of the initial ionization, allowing preservation of the proportionality of charge collected. This is known as the proportional region. In Region III, the charge collected is independent of the ionization initiating it. This is the G-M region.

XBL 796-1728

Fig. 4. Ideal response versus voltage.

As a whole, gas-filled counters are not inherently very sensitive to x-rays and γ -rays without special design; the probability that an energetic photon will produce an electron within the volume of the counter is very low. The thickness of the detector entrance wall can cut off α -particles and any but high energy β -particles. Sometimes in order to discriminate between β and γ -radiation, a very thin entrance window is additionally provided. When the thick window is in place, only photons can reach the sensitive detector volume.

(i) Ionization Chamber. Since many of the principles involved in the ionization chamber are applicable to other detectors, we will discuss the ion chamber in some detail.

The principal advantage of this instrument is that the exposure or absorbed dose can be calculated directly from the instrument reading provided the design is optimized and there is rudimentary knowledge of the spectrum of the radiation to which the chamber is exposed. Equally, since the roentgen is the unit of exposure and is defined in terms of the release of total electric charge resulting from the irradiation of a unit volume of air at STP, the ion chamber method is the most direct method of measuring exposure in these units. A simple ionization chamber is presented in Fig. 5. The electrons would be collected on the upper electrode. For practical sized ion chambers the collection time is on the order of microseconds. For filling the chamber, almost any gas is suitable, so long as it can be ionized.

XBL 796-1726

Fig. 5. Simple ionization chamber.

As a particle passes through the chamber, ion pairs, consisting of positive and negative charges of size e , are formed. Each pair corresponds to a loss in particle energy of approximately 34 eV. If particles pass through the chamber at a rate N , each losing total energy E , then (positive or negative) charge is formed at the rate

$$Q = \frac{NEe}{34 \text{ eV}} = \frac{NEe \times 10^6}{34} \text{ Coulombs/sec}$$

for N in particles per second, E in MeV, and e in Coulombs. If the strength of the electric field is sufficient, it is possible to collect all of the electrons and positive ions so that a current (J) flows in the circuit equal to $Q/2$. Usually a few hundred volts is sufficient to collect all of the charge.

An ammeter used to measure the current, as shown in Fig. 5, will not count the arrival of each group of electrons corresponding to the instantaneous arrival of particles, because an ammeter's response is usually too slow. Instead it is calibrated to respond to the effect of many particles by giving a reading that is proportional to the number of particles arriving per second, and to the average energy loss in the gas.

The roentgen is defined as the total ionization produced by the secondary electrons formed in a volume of air being measured. Since some of these secondary electrons may have ranges of several meters, large and cumbersome apparatus would be required to measure their total ionization and hence the exposure in roentgen. To avoid this problem, air equivalent, or thimble chambers, have been developed. Their operation is based on the principle that when a tiny cavity, such as a small ionization chamber, is placed in a large homogeneous absorbing medium that is uniformly irradiated, the flux of secondary electrons in the cavity is identical to the flux which existed in the medium before the cavity was introduced. If the chamber gas is air and if the walls are composed of materials having an atomic number near that of air, then the energy loss per gram of air in the chamber will be substantially the same as the energy loss per gram of air at the point where the chamber is located. An ideal chamber can cover only a limited range of energies, since many factors enter into the construction and use of such an instrument. For example, the use of "approximately" air-equivalent walls, the use of metallic electrodes, the absorption of low energy radiation in the chamber walls, and the lack of sufficient wall thickness for electron equilibrium at high energies are some of the factors.

Sometimes, instead of a flux measurement, individual particles need to be measured, particularly for α -particles. If the ionization chamber of Fig. 5 is connected to the circuit shown in Fig. 6, it may be used to detect the arrival of individual particles.

Electrical charge collects on the plates in about one millisecond or less and a pulse of current flows through resistor R . Since the voltage across R is proportional to the current flowing through it, the passage of a single ionizing particle causes a voltage pulse, as discussed below. R is usually a very large resistance, about 10^9 ohms, so that even a

XBL 796-1727

Fig. 6. Ionization chamber and circuit for detection of individual particles.

very small current may produce an easily measurable voltage pulse. The pulse is amplified, and is finally recorded by an electric counting circuit.

The electrodes of an ionization chamber behave as a parallel plate capacitor of capacity C . The charge collected in the short time following the passage of particle leaks away through resistance R with a time constant (decay time) equal to the product RC . (If R is in ohms, C is in farads, RC will be units of seconds.) If RC were sufficiently large, the voltage signal across R would have the time variation shown in Fig. 7. The fast rise at A is due to the collection of the fast moving electrons, the slower rise at B to the collection of slow moving positive ions, and the slow exponential fall is just the time constant of the circuit, RC .

Fig. 7. Signal from ionization chamber with $RC \gg 10^{-3}$ second (from Ref. 2, Bernard G. Harvey, Introduction to Nuclear Physics and Chemistry, 2nd edition, © 1969, pg.334. Reprinted by permission of Prentice Hall, Inc., Englewood Cliffs, New Jersey.)

Several disadvantages arise from long pulses of this type. The chamber and the pulse amplifier cannot accept another pulse until a long time has elapsed, so that only low counting rates can be accepted; the resolving time is said to be long. Electrical noise reduces the energy-resolving power of the counter-amplifier system since the pulse voltage carries information about the energy of the particle. To minimize these problems, the amplifier used in this application is designed with a narrow bandwidth to discriminate against the slowly rising pulse due to collection of positive ions (B in Fig. 7). In this way the counter-amplifier system is restored within a few microseconds to a condition suitable for the detection of a second particle so that very high counting rates can be achieved. In addition, maximum discrimination against electrical noise is obtained. A common practice is to differentiate the pulse, that is, to create a pulse whose voltage is proportional to the time derivative of the pulse from the ionization chamber. There are a number of ways to do this, both active and passive. The short pulse produced by differentiation can be further amplified, and then recorded by an electronic counting circuit.

Although the ionization chamber and circuit of Fig. 6 is reasonably satisfactory for the detection of charged particles, it is not satisfactory for the measurement of the energy of these particles. The commonest such application is measurement of the energies of α -particles from a source placed on the negative electrode. The particles are emitted into the gas in all directions, as shown in Fig. 8 (lines A and B). The electrons are quickly removed from the tracks and travel rapidly towards the collecting electrode C. The positive ions remain behind and they induce a small charge on C whose magnitude depends on the distance be-

Fig. 8. Operation of ionization chamber with grid (from Ref. 2, Bernard G. Harvey, *Introduction to Nuclear Physics and Chemistry*, 2nd edition, © 1969, pg. 335. Reprinted by permission of Prentice Hall, Inc., Englewood Cliffs, New Jersey.)

tween C and the "center of gravity" of the positive ion track. The distance is clearly more for A than for B. The pulse produced will be greater for track A, even if each particle initially produced the same number of ion pairs. To minimize this effect, the grid G is placed as an electrostatic shield between the positive ions and the collector. It is operated at a potential just sufficiently negative to prevent electrons from being attracted to it, but since it consists of very fine wires (0.005 in. diameter) spaced well apart, the electrons can readily pass through it. Thus a grid ionization chamber gives an output pulse that is proportional to the number of ion pairs produced by an alpha particle, and thus to its energy, regardless of the orientation of the track.

If the counter and amplifier are designed to detect only a fast rising electronic pulse, the electrons must be free to move rapidly through the gas of the counter. In some gases, such as those containing oxygen and halogen vapors, electrons rapidly become attached to neutral molecules to form slow moving negative ions. These gases must be carefully excluded from the chamber. The gas most commonly used is argon, usually mixed with a few percent of methane or carbon dioxide for reasons that are discussed below.

The time required for collection of the electrons depends on their drift velocity through the gas towards the collector. The initially fast electrons are slowed down by collisions with the gas atoms, but argon is a poor stopping gas for low energy electrons. It is a tightly bound atom with a large ionization potential, so that there are no low energy states that can be formed by inelastic collisions with the electrons (the lowest excited state of argon is at 11.5 eV). A quantum mechanical phenomenon called the Ramsauer effect produces a very large increase in the average distance between collisions with argon atoms for electrons with kinetic energies below 10 eV, but in argon it is not easy for electrons to slow down to energies as low as this. The velocity v with which electrons drift through a gas of pressure P under the influence of an applied electric field of E volts per cm is

$$v = \frac{e E \lambda}{P \sqrt{2m \epsilon}}$$

where m is the electron mass, ϵ is their average kinetic energy, and λ is the mean free path between collisions.

If a few percent of methane or CO_2 is added to Ar, inelastic collisions become more probable because these molecules have many low-energy (vibrational and rotational) states that can take up energy in collisions with

electrons. The average energy ϵ of the electrons is only one tenth as great in argon containing these molecules as it is in pure argon. Thus, because of the Ramsauer effect, λ becomes much larger, and both the reduction in ϵ and the increase in λ cause v to increase. It can be seen then that the addition of about 10% of CH_4 or CO_2 decreases the time required for collection of electrons in an argon-filled ionization chamber, and gives a corresponding decrease in the resolving time.

More information on the design of ionization chambers for specific applications, and on α and β particle detection can be found in the IAEA Technical Reports Series No. 150, 'Measurement of Short-Range Radiations' (Ref. 9).

(ii) Proportional Counters.

The proportional counter has the same basic design features as an ionization chamber, except that the electric field strength near the central electrode is much higher. This high field strength causes the electrons released in ionization to be accelerated toward the central electrode, and to gain sufficient energy to ionize further gas atoms by collisions. Thus, the charge collected, and hence the initial voltage developed, is amplified considerably. This type of detector is particularly valuable because the magnitude of the voltage pulse is proportional to the amount of the primary ionization. Therefore, such a detector can be used to measure the ionizing power of radiations.

In order to get a high electric field the electron-collecting electrode in a gas filled counter is made of thin wire (typically 0.001 inch). If the counter has the design shown in Fig. 9, the electric field

Fig. 9. Proportional counter schematic.

strength E at a distance r from the wire is

$$E = \frac{V \text{ volts}}{r \ln(d_1/d_2) \text{ cm}}.$$

V is the positive voltage of the central wire relative to the outer cylinder and d_1 and d_2 are the counter and wire diameter. At small distances r , large values of E are obtained, but the smallest value of r is

limited by the wire radius. Hence the field strength is greatest adjacent to a thin wire, even though thick wires give greater values of E for a fixed value of r .

By this arrangement an avalanche of ionization can thus be initiated by even a single electron. Most of the avalanche occurs very close to the wire where the electric potential gradient is high. The size of the output pulse is proportional to the gas multiplication factor, and to the number of primary electrons (and hence to the energy lost by the particle). The gas multiplication factor (maybe 10^2 or more) depends on V , the high voltage on the wire, as shown in Fig. 10. However it cannot be increased without limit as the gas becomes unstable at very high gains ($G > 100-200$). If the energy of a particle is to be measured in a proportional counter, the gas multiplication factor must be kept constant by using a very stable high voltage supply. Energy resolution of about 1% can be achieved. This counter is occasionally used for dosimetry but is more commonly used for x-ray spectrometry up to about 20 keV at which energy the low absorption of the gas causes the efficiency to become very low.

Proportional counters are often used for the detection of α , β^- , β^+ , and (EC) electron capture decays even when measurement of the energy is not required. The sample may be placed inside the counter and the air is removed by means of a continuous flow of gas, usually methane. Figure 11 shows a typical arrangement with sample external to the counter.

Fig. 10. Gas multiplication factor as function of applied voltage for argon pressures 10 and 40 cm Hg in counter 0.87 in. in diameter and wire 0.01 in. in diameter (from Rossi and Staub, Ref. 19, with permission by McGraw Hill © 1949).

Fig. 11. Cut-away view of typical proportional counter (from Ref. 2, Bernard G. Harvey, *Introduction to Nuclear Physics and Chemistry*, 2nd edition, © 1969, pg.348. Reprinted by permission of Prentice-Hall, Inc., Englewood Cliffs, New Jersey).

Figure 12 shows the additional circuits used with such a counter. The discriminator rejects pulses unless they exceed a certain voltage, which can be adjusted. The number of pulses which arrive are recorded by the scaler and register. Since the counter can detect as many 10^5 particles per second, an electronic rather than mechanical register must be used, since mechanical systems are far too slow.

Fig. 12. Block diagram of typical circuit used with a proportional counter.

Proportional counters are used for the detection of β or γ radiation, and can be used for α radiation also, but for the latter they have poorer energy resolution than ion chambers. Proportional counters, while less sensitive than the Geiger counter, will give a good indication of the particle or photon energy. Furthermore, their response to photon energy can be made to resemble that of human tissue.

This is advantageous when precise physiological measurements are to be made. However, the proportional counter is as complicated and expensive to use as the scintillation counter (discussed later) and very much less sensitive.

The proportional counter can be designed to do internal counting of radioactive gases; it can also count solid sources. Special designs have been made for surface monitoring (Ref. 9).

(iii) Geiger-Mueller Tubes.

The Geiger-Mueller tube is the simplest and most widely used detector. It is moderately sensitive, but gives no indication of the energy of the radiation being detected.

The Geiger counter is one of the oldest counting devices, and it has many shortcomings. In this type of detector, the amplification of the voltage pulse described in the case of proportional counters is carried one step further by increasing the potential of the central electrode. The amplification is sufficient to cause a breakdown discharge along the entire length of the central electrode (usually in the form of a wire) being used in the chamber, thus making the size of the ultimate pulse completely independent of the extent of the primary ionization.

As the initial avalanche of electrons approaches the central wire, the positive ions produced in the counter gas move towards the outer wall. When these ions hit the wall, electrons are ejected, resulting in a second avalanche and similarly in successive avalanches. Special precautions called "quenching" are required to stop the avalanche from turning into a continuous discharge.

One way to quench a tube in order to restore it to its original quiescent state is to remove the high voltage momentarily. This can be done electronically and is called external quenching. Another method more commonly used today is to employ internal quenching by mixing a small quantity of a polyatomic gas with the counter gas. The positive argon ions become neutralized by removing electrons from the polyatomic molecules. The polyatomic ions then drift relatively slowly away from the wire towards the counter wall, but their energy of recombination with electrons from the counter wall goes into decomposing the polyatomic atom rather than into the emission of new electrons. In this way, the avalanche is prevented from becoming self-sustaining. If a substance like alcohol or butane is used, the tube is said to be "organic-quenched." Such a tube has a useful life of about 10^8 or 10^9 counts because the molecules are decomposed irreversibly. "Halogen-quenched" tubes utilize chlorine, bromine and their compounds as quenching agents.

These tubes have a much longer life because the atoms normally recombine. These counters give larger pulses and operate at considerably lower voltages than organic quenched GM tubes.

The net behavior of a Geiger-Mueller tube during an ionizing event is the result of two opposing groups of factors, those tending to perpetuate the discharge and those tending to limit discharge, as discussed in the preceding paragraphs. After a GM tube has been exposed to a high intensity gamma source, the background of the tube may be abnormally high for some time after the source has been removed. The reason for this "memory" is not entirely understood but it is probably due to the formation of activated molecules of gas within the tube.

Once ionization of a GM tube is initiated, the tube becomes insensitive for a short interval of time. This is due to the time it takes for the positive ions to drift away from the central electrode ($\sim 10^{-3}$ seconds). This interval is called the dead time and sets a lower limit to the resolving time. It represents the time during which two or more ionizing particles striking the sensitive portion of the tube will be counted as a single particle. As a result of this phenomenon, the number of counts will be less than the actual number of particles passing into the tube. This could be a potential problem in a very high radiation field. The difference between the true and observed count is known as the coincidence loss.

GM tubes are not equally sensitive to alpha, beta, and gamma radiations. To initiate discharge of the tube, the radiation must first reach the sensitive volume. Alpha particles, being the least penetrating, may be absorbed by the window unless it is very thin. Beta particles are more penetrating and gamma rays are very penetrating. Thus thicker windows can be tolerated with the latter two types of radiation.

Considering only the radiation reaching the sensitive volume of the tube and assuming zero coincidence loss, it is found that the efficiency of the GM tube is essentially 100% for alpha particles, nearly 100% for beta particles but only 1 or 2% for gamma radiation. Ironically the property of gamma rays which enables them to penetrate the window of the tube so efficiently to reach the sensitive volume, i.e., their ability to pass through relatively thick layers of matter without interacting with the matter, now works against them. Most of them pass through the sensitive volume of the gas without interacting.

GM tubes are available in a variety of forms, a common form being the end-window variety. End windows consist of a thin film ($\sim 2 \text{ mg/cm}^2$) of mica, mylar or other suitable

substance especially selected to retain the gas in the tube while offering minimum interference to the passage of radiation. Obviously this window is extremely delicate and must not be touched with the fingers. Broken windows frequently account for GM tube casualties in the laboratory. Side windows are normally about ten times thicker than end-windows, since the window itself serves to support the end of the tube. Side window tubes are useless for alpha particle detection, but are quite useful for the detection of energetic betas. On the other hand side window tubes are more sensitive to gamma radiation than are end window tubes presenting the same cross section to the radiation. The thicker window increases the probability of interaction of the gamma rays to eject an ionizing electron. Certain GM tubes especially designed to be gamma sensitive have bismuth coated windows which serve as cathodes, and also sometimes steel windows. For these tubes the gamma efficiency is often as high as 6 or 8%.

Uncompensated GM tubes do not have anything like a uniform energy response. Examples are shown in Figs. 13 and 14. If an accurate exposure rate is to be determined with the Geiger tube, the energy response of the particular Geiger tube as well as the energy distribution of the source must be known. Moreover, it should not be assumed that the radiation detected by the GM tube is monoenergetic just because the source is monoenergetic. In most applications, measurements will be taken of scattered radiation rather than a direct beam measurement from a monoenergetic source. Under most conditions, the Geiger tube will detect radiation that has bounced off everything from concrete to steel beams. Under typical scattered radiation conditions, the actual dose rate, when using an uncompensated Geiger tube, can be off by a factor of three or four. Geiger tubes can be compensated in several ways to make them exhibit a reasonably flat energy response from 60 keV at 2 MeV; however, this normally eliminates the detector's sensitivity to beta particles.

The advantage of the GM tube is that it produces a large pulse requiring little further amplification, but the pulse size is independent of the nature of the incident radiation. Geiger counters are cheap and simple. They find their main use as portable instruments in surveying laboratories for possible radioactive contamination, and in mineral prospecting. Their other advantages are high sensitivity, small size, and simple reliable electronic circuitry.

b. Scintillation Detector. In certain organic and inorganic materials, the passage of a particle or photon produces excited atomic states which return to ground

EBERLINE INSTRUMENT CORPORATION GEIGER
COUNTER MODEL E-400 SERIAL NO. 389

EBERLINE INSTRUMENT CORPORATION GEIGER
COUNTER MODEL E-500A SERIAL NO. 182

Fig. 13. Energy response for Geiger counters (from Ref. 20).

LUDLUM MEASUREMENTS, INC.
COUNT RATE METER MODEL 12 SERIAL NO. 667

LUDLUM MEASUREMENTS, INC.
GEIGER COUNTER MODEL 14 SERIAL NO. 85

Fig. 14. Energy response for Geiger counters (from Ref. 20).

state by the emission of light in or near the visible part of the spectrum. In the family of scintillation counters, this light is converted to electrical pulses by means of which the incident particles are counted.

Scintillation counters are at least an order of magnitude more sensitive than the Geiger counter. Furthermore, they indicate the photon energy of the radiation flux. Like the Geiger counters, they measure the rate of arrival of the photons, which must be integrated to determine the received dose. However, they are more complicated than Geiger counters, more expensive, and usually require a trained operator. Recently, with the introduction of microprocessors and better electronics, simple survey monitoring instruments have been introduced.

When scintillation devices are used in dose rate measuring instruments, they are operated in a special way somewhat foreign to the normal operation of scintillation and proportional counters. The detector is compensated to yield a reasonably flat response from approximately 80 keV to 1,300 keV. If such a device is used, the user should be aware of the 80 keV cutoff because scattered radiation, as well as bremsstrahlung, can be well below 80 keV. These detectors are used as spectrum analyzers as well as a dose rate measuring device.

The first scintillation detectors of this class, under the name spinthariscopes, were primarily used to detect individual particles by observing the flash they produced. Rutherford used a ZnS(Ag) screen as a detector in his famous α -scattering experiments. Although this was the earliest detector, it did not attain widespread use because it needed a human observer of the flashes to catalogue them. It was with the development of efficient photomultipliers that they came into full use. Also, the discovery of new scintillating materials such as anthracene and the alkali halides, which are transparent to their own radiations and can therefore be used in varying thicknesses, led to increased interest and widespread use.

These materials are useful as scintillators because they are luminescent, i.e., energy absorbed may be reemitted as visible or near visible radiation. The emission that follows absorption of nuclear radiation originates from excitation and ionization produced in the substance by the radiation. If there are allowed transitions between the resulting excited state and ground state, then a light emission accompanying deexcitation occurs on the order of time typical for the lifetime of an allowed transition for atomic states ($\sim 10^{-8}$ sec $\rightarrow 10^{-10}$ sec). On the other hand, if the excited state is metastable, the emission is delayed and the process is known as phosphorescence. The

duration of phosphorescence varies with the type of luminescent material and may be from nanoseconds to hours. Only those scintillators with the shorter durations are useful in scintillation counters for the detection of ionizing radiation. Generally scintillation materials are divided into five classes:

- 1) organic crystals
- 2) liquid solutions of organic materials
- 3) solid solutions of organic materials (plastic scintillators)
- 4) inorganic crystals
- 5) noble gases.

We will not discuss the theories of the scintillation process. The reader is referred to Ref. 5 for details. For the present, it suffices to say that the number of photons produced is proportional to the energy absorbed in the scintillator.

A scintillator counting system is illustrated in Fig. 15. Since the light emission is isotropic, in order to increase efficiency, the scintillator is covered on all sides with a reflector, except where a light pipe connects to the photomultiplier. The photomultiplier is described in the discussion of associated electronics, as are the choices of additional equipment for the performing specialized functions pertinent to the possible ways of operating the scintillator detector.

Fig. 15. Schematic Diagram of Scintillator Detector.

c. Semiconductor Detector. In principle, a semiconductor detector can be referred to as a solid-state ionization chamber. The charge carriers produced by the ionizing radiation instead of being ion pairs are electron-hole pairs. The electrical signal is obtained by separating and collecting the electrons and holes, whose numbers are proportional to the energy deposited by the incident radiation in the sensitive region of the detector.

For a more in-depth discussion of nuclear radiation detection with semiconductors see the book by Bertolini and Coche¹³ and a more recent article by Goulding and Pehl.¹⁴

The semiconductor detector is usually made of Si or Ge (although there is progress in developing detectors based on HgI_2 , CdTe , etc.), and various procedures such as drifting of Li, or extreme purification, are used to make possible a large effective zone with reverse biasing. They are in general very similar in operation.

Regardless of the particular detector, general comparisons can be made with a gas ionization chamber:

1. The energy required to form an electron-hole pair in silicon is about 3.5 eV compared with 34 eV per ion pair in air, allowing greater resolution.
2. Although the resolution due to the statistics of measurement is better than the gas counter, effects of noise give a limit to the resolution of about 150 eV and limit the lowest attainable energy (by thermal noise) to about 2 keV.
3. The higher density of the semiconductor detector material allows for higher detection efficiencies than in a gas counter; the atomic number also helps.

A semiconductor detector must typically be operated at low temperatures (77°K, Liquid N temperature) to prevent the drift of Li from Ge(Li) or Si(Li) and to reduce the thermal noise contribution, even for Hyperpure (Hp) Ge. (Si surface barrier detectors do not have to be cooled, although cooling improves the resolution.) In addition, this low temperature requires a vacuum which in turn sets a limit on how thin the necessary window between the source and detector can be. Performance-wise this means that it is difficult to obtain a sufficiently thin window to work much below 5 keV.

Two types of semiconductor detectors useful to our applications are the junction/surface barrier detector, and the lithium drifted detector. Junction detectors can be formed with depletion depths up to 2 mm and can be made with very thin windows, making them suit-

able for the spectrometry of low-energy β and α radiation. Typically the detection efficiencies for α -particles is about 25% with good rejection of γ -radiation. Surface barrier detectors are 100% efficient for α 's, without rejection. Lithium drifted detectors can be made with a large volume ($\sim 100 \text{ cm}^3$) making them suitable for γ -ray and x-ray spectroscopy. In addition, hyperpure Ge detectors are available for photon spectroscopy from about 10 to 100 keV.

For spectroscopy purposes, the energy resolution of these detectors is an important figure. Typically for γ -spectroscopy Ge (Li) yields about 1.8 keV as seen in Table 1.

A pulse-height analyzer is needed to perform spectroscopy. The electronics is discussed later and in the γ -spectroscopy section (Section III.E.5.a); both discussions are also appropriate to α and β spectrometry.

d. Thermoluminescence dosimeter.

In contrast to gas counters, scintillation, and semiconductor detectors, which are devices that given an immediate indication of the presence of radiation, the devices which use thermoluminescence integrate the effects of all the doses received and only produce a signal when heat is applied. Basically, in thermoluminescent (TL) materials, the electrons and holes produced by radiation are trapped and later released by heating to a temperature of a few hundred degrees centigrade. During the heating process light is emitted and detected with a photomultiplier tube. Similarly in thermally stimulated electron emission (TSEE) the heating process releases electrons from the surface of the material which can be measured by a proportional counter. We will discuss only TL dosimeters.

Exposing a thermoluminescent phosphor to radiation causes electrons to be raised from the valence band into the conduction band and holes are left in the valence band. These electrons (and holes) can move through the crystal lattice until they either return to the valence band or are captured by lattice imper-

Table 1. Properties of Some Common Semiconductor Detectors

Detector	Spectroscopy/Resolution (FWHM)		
	X- γ -ray	α	β
Ge	180 eV @ 5.9 keV	15 keV @ 5.8 MeV	maximum 10 keV
Si			
Ge (Li)	1.8 keV @ 1332 keV		
Si (Li)	154 eV @ 6.542 keV

fections. Those electrons that are caught in shallow traps (imperfection) are returned to the conduction band fairly rapidly by thermal effects at room temperature, whereas those in the deeper traps require considerably more energy to escape and only do so when the material is heated. These released electrons return to the ground state with the emission of visible light. It is the deep traps that are most useful in dosimetry since they are least affected by a change in ambient conditions. As a single rapid rise in temperature does not generally empty all the traps, it is normally necessary to anneal the TL material before reusing.

The TL material may be in the form of a powder, crystal, or polycrystalline disc. The readout instrument for TL material consists of a heater with a thermocouple attached to measure the temperature, a photomultiplier tube with perhaps a filter to shift or remove certain wavelengths of the radiation given off, an amplifier and generally an x-y plotter. A curve plotted with the x-axis as the temperature measured by the thermocouple and the output current from the photomultiplier displayed on the y-axis is known as a glow curve and is illustrated by Fig. 16. For dosimetry purposes, a fixed heating cycle can be used and the output of the photomultiplier tube converted to a digital signal which is integrated between suitable points on the glow curve.

Fig. 16. Typical Glow Curve for TLD.

In dosimetry, the user should be aware of the response of the TL material to different energies of radiation. For instance, below

50 keV LiF exhibits an enhancement of 50% in response. The dynamic range of response is also important; for LiF it is from 10 mrad to ~1 krad. The response should be independent of dose rate, but in any case should be known. For LiF, the response is independent of the dose rate up to at least 10^{11} rads/sec. Above an integrated total dose of about 1000 rads, the response is supralinear, and it starts to saturate with a limit of 10^5 rads, beyond which radiation damage results in a loss of reading. It is also important to know how the TL material responds to different radiations (gamma, neutron, etc.).

e. Photographic detection. The latent image of the track from ionizing radiation traversing a silver halide emulsion is relatively permanent. Thus when the film is developed a record is obtained. Depending upon the technique used, a photographic emulsion can be used as a single event detector by recording the track of a single ionizing particle or can be used to locate sites of radioactivity on a surface. Its most prominent use in dosimetry is as a film badge where it is used to integrate many events over a period of time, after which the change in optical density is measured.

An important property of the emulsion is its grain size, since this determines how much and what type of radiation is measured. The user should, as always, be aware of the energy response, dose-rate dependence, the dynamic range, and any radiation type dependence to response.

f. Solid-State Track Detectors. Charged particles impinging on an object such as a film of a high polymer or glass plate produce trails of damage which can be chemically etched into tracks that can be examined individually or counted with an optical microscope or by spark counting. Since these detectors find primary use for measuring heavy charged particles, fission products, and other ions from accelerators or in cosmic radiation, they will not be discussed here. For further information the reader is referred to Ref. 15.

4. Associated Electronics

a. Electrometers. Electrometers are basically charge reading devices. There are several kinds, such as the quartz fiber electrometer, the vibrating-reed electrometer, an electrometer vacuum tube, and metal-oxide semiconductor field effect transistor (MOSFET) electrometers. Most of the electrometer circuits used in portable meters follow the block diagram in Fig. 17.

The circuits consist of a detector (e.g. an ionization chamber), a power supply for the detector, an electrometer amplifier where necessary, a power supply for the amplifier

XBL 796-1731

Fig. 17. Block diagram of electrometer.

section and a metering circuit. New components have enabled the manufacturer to reduce size and increase sensitivity of the electrometer.

The MOSFET and the vibrating reed offer various advantages and disadvantages when compared with the simple electrometer tube. The main advantages of the electrometer tube are low cost and low noise. However, its disadvantages include shock sensitivity (microphonic), poor long and short term stability (drift), and a poor temperature coefficient. The MOSFET has features such as low cost, shock insensitivity, good battery life, and high reliability; however, it does have some faults such as a high offset current, high noise, and poor overload characteristics (easily burned out). The vibrating reed has a variety of advantages such as excellent stability, very low offset current, low temperature coefficient, low noise, and good overload capabilities. However, it does have some shortcomings such as high cost. reason-

ably large size, poor battery life, and relatively complex circuitry. A quartz fiber electrometer is less sensitive than the two others mentioned above, and also requires visual readout. However, it is rugged, small and relatively inexpensive.

Drift is one of the major disadvantages of an ion chamber survey meter using an electrometer tube input amplifier. The electrometer tube survey meter on the most sensitive range will drift as much as 6% with time and temperature; therefore, it should be checked out and zeroed frequently. After the first hour of operation, it is usually stable to within 5% provided there is no significant temperature change. MOSFET and vibrating reed instruments are very stable with both temperature and time, as are GM and proportional counters.

b. Pulse-Height Analyzer (PHA).

Pulse height analyzers are a necessary component for the performance of spectroscopy. In order to be useful, the signal it receives from the detector must be proportional to, or at least a monotonic function of, the initial energy deposited by the ionizing radiation. The simplest PHA is a single channel analyzer, which consists basically of a window for energy discrimination. The top and bottom of the window usually can be set at varying energies. The lower edge is set just below the peak of interest while the upper is set above that peak. The single channel analyzer (SCA) then accepts only those pulses lying between the

Fig. 18. Spectrum of aluminum-28 as measured with a Ge(Li) detector (from Adams and Dams, Ref. 21, with permission by Pergamon Press © 1970).

two energies set by the edges. A many-channel pulse height analyzer can be basically thought of as many SCAs, each recording only those pulses falling in their domain. The channels are stored separately, contributing to a spectrum such as Fig. 18. The spectrum can be stored on magnetic tape for later data analysis, read out on an x-y plotter, or punched out on paper tape.

For a more detailed discussion the reader is referred to Section III.E.5.a.

c. Photomultiplier tubes. A photomultiplier consists of three essential components, namely (1) a photocathode (2) a secondary emission multiplier structure and (3) an electron collector system as output. When a flash of light falls on the photocathode, electrons are released in proportion to the number of photons in the flash.

At the present there are two types of multiplier structures. One consists of an array of secondary emitting dynodes or stages. Electrons strike the first stage and are increased in number by secondary emission, since each initial electron produced was increased in energy by an accelerating potential. These electrons are guided to the next stage where their number is again increased. This is repeated until amplification reaches the required level. Then the electrons are collected by the anode system.

The second type of photomultiplier structure consists of a plate made up a large number of tiny tubes coated with a secondary emitting material. An electron entering any one of these tubes or microchannel is multiplied by striking the wall and traveling through an increasing potential. These multiplier structures are very fast, very linear (if the light flashes are not too large), and unlike the first mentioned structure, quite insensitive to magnetic fields.

Typically a photomultiplier of either type will have an electron (current) gain of the order of 10^5 or 10^6 . Their response time is fast, on the order of 100-200 picoseconds for high speed photomultipliers.

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I.F. General Analytical Considerations

1. Tissue Equivalence
2. Particle Equilibrium
3. Radionuclide Decay Schemes
4. Calibration
5. Instrumental Response to Environmental Conditions
6. Critical Pathway Analysis
7. Reference Methods
8. Acknowledgments
9. References

F. General Analytical Considerations

This section emphasizes analytical considerations that are often important in radiation measurements.

1. Tissue Equivalence

A general consideration is the concept of tissue equivalence. Since one desires to measure rad and rem as energy deposited in tissue, any measurement of ionization in another medium (for example, in an air ionization chamber) must be interpreted with caution. One possible approach is to correct such measurements to yield the tissue dose. More often, however, uncorrected measurements are given, and they must be recognized as such.

An alternate possibility, less often chosen, is to make a detector out of tissue-equivalent material, which is strictly defined as material whose atomic composition matches exactly that of human muscle (10.2% H, 12.3% C, 3.5% N, 73.0% O, 2.0% others).

A true tissue-equivalent ion chamber would give an interpretable reading independent of the radiation composition (Ref. 2). There have been several attempts to construct tissue-equivalent plastics, and Table 1 shows the atomic composition of a commonly used material. Such material, which may be used for fabricating ionization chambers and auxiliary devices,* matches the hydrogen and nitrogen components closely, but replaces much of the muscular oxygen with carbon. It has been shown (Ref. 1) that for both neutrons and photons with energies from 0.5 to 14 MeV the use of carbon

*For practical guidance from the ICRU, see Ref. 4.

instead of oxygen induces errors of less than about 5%. Tissue-equivalent gases may, on the other hand, have significant oxygen content (Ref. 5).

The key element in the tissue-equivalent material is hydrogen, especially for neutron dosimetry below about 5 MeV; at higher energies the heavier elements become increasingly more important (Ref. 3). "One method of separating the gamma and neutron contributions in an absorbed dose measurement employs two ionization chambers, one with walls and filling of hydrogen-rich material responding strongly to neutrons, and the other with walls and filling devoid of hydrogen responding mostly to gamma radiation" (Ref. 3). The most common tissue-equivalent devices are ionization chambers, of which those of Rossi (Ref. 5) and Shonka (Ref. 7) are typical: in each case, both the walls and the gas are composed of tissue-equivalent materials. A gas proportional counter made of tissue-equivalent material has also been constructed for use around a high-energy accelerator (Ref. 8).

2. Particle Equilibrium

An important consideration, particularly for high energy radiation, is the possibility that particle interactions in tissue may significantly alter the radiation and ionization spectra. The concept which is important here is particle equilibrium, which occurs when primary particles traversing through a medium are accompanied by an equilibrium spectrum of secondary particles produced by interactions with the medium.

For this reason it is essential to differentiate between radiation at the surface and in the interior of a body. The distinction is clearly important, for example, for a pure,

Table 1. Elemental composition of TE plastic (from ICRU, Ref. 4, with permission by International Commission on Radiation Units and Measurements © 1977).

Element	Percent weight		
	ICRU muscle tissue	A-150 design composition	A-150 chemical analysis (std. dev. ^a)
Hydrogen	10.2	10.1	10.3 (0.1)
Nitrogen	3.5	3.5	3.5 (0.07)
Oxygen	72.9	5.2	6.4 (0.6)
Carbon	12.3	77.6	77.1 (0.3)
Calcium	0.007	1.8	NAC
Fluorine	—	1.7	NAC
Total	98.9 ^b	99.9	97.3

^aStandard deviation of 5 replicate samples analyzed.

^bThe 1.1 percent remainder consists of Na, Mg, P, S and K.

^cNA: no analysis.

mono-energetic relativistic neutron flux incident upon some material. Since absorbed dose occurs through ionization loss, one must consider the charged secondaries, which then cascade through the material, ionizing and perhaps producing additional tertiary particles of their own. The ionization loss at first increases with penetrating distance, but ultimately it reaches a maximum before falling off due to absorption of the primaries themselves.

The quantity of interest for radiation protection is the dose equivalent. Since this is related to the ionization loss, it is important that an instrument measure absorbed dose near the maximum ionizing region. This is often accomplished by adding absorbing material to the instrument walls or surroundings. Indeed, it is usually important to determine whether or not particle equilibrium is present; a standard technique for this determination is the addition of just such an absorber. As a helpful guide, Table 2 (from Ref. 3) shows the thickness of an ionization chamber wall required for establishment of electronic equilibrium for photons of various energies, while Table 3 is for complete equilibrium of neutrons traversing water.

Table 2. Thickness of ionization chamber walls required for establishment of electronic equilibrium (from ICRU, Ref. 3, with permission by International Commission on Radiation Units and Measurements © 1971).

Photon Energy (MeV)	Thickness ^a (g cm ⁻²)
0.02	0.0008
0.05	0.0042
0.1	0.014
0.2	0.044
0.5	0.17
1	0.43
2	0.96
5	2.5
10	4.9

^aThe thicknesses quoted are based on the range of electrons and protons in water. The values will be substantially correct for tissue-equivalent ionization chamber walls and also for air. Half of the above thickness will give an ionization current within a few percent of its equilibrium value.

In some cases (particularly behind low Z concrete shielding walls) particle equilibrium will already exist, but any conscientious survey should routinely include a check using various absorber thicknesses.

Table 3. Appropriate thickness of water required to establish complete equilibrium for neutrons of various energies (from ICRU, Ref. 3, with permission by International Commission on Radiation Units and Measurements © 1971).

Neutron Energy (MeV)	Thickness ^a (g cm ⁻²)
5	0.034
10	0.12
20	0.42
50	0.42
100	7.6

^aSee note on Table 1.

As a way of illustrating this point in another light, Fig. 1 (from Ref. 9) shows the neutron flux density required to produce the same dose equivalent (1 mrem/hr) at three depths of tissue, as a function of incident energy.

Fig. 1. Neutron flux density required to produce same dose equivalent (1 mrem/hr) at three depths of tissue as a function of incident energy (Ref. 9).

3. Radionuclide Decay Schemes

For information about detailed decay schemes, the reader is referred to the seventh edition of the Table of Isotopes (Ref. 10). In measurements of radioactive decay, the relationships between parents and daughters (and more generally among the many members of a long decay chain) are very important considerations. Often, measurements of a radionuclide are actually performed by using the properties of the chain in question. There are several examples among the radionuclides treated in this Survey. Among these are ⁹⁰Sr measure-

ments by ingrowth of ^{90}Y ; ^{226}Ra measurements by emanation of ^{222}Rn ; and the entire radon daughter measurement problem.

Here we shall outline a few of the considerations which must be taken into account when measurements of parent-daughter systems are made. We shall not attempt to give a general mathematical description, since that has been done well elsewhere. The reader is referred in particular to Evans' book The Atomic Nucleus (Ref. 11).

The basic exponential law governing the decay of one isolated ensemble of identical nuclei is:

$$N(t) = N^0 e^{-t/\tau} = N^0 e^{-\lambda t} = 2N^0 e^{-0.693t/T_{1/2}}$$

Here $N(t)$ = number of nuclei remaining at time t

$$\begin{aligned} N^0 &= \text{number at time } t = 0 \\ T_{1/2} &= \text{half-life} \\ \tau &= \text{mean life} = T_{1/2}/0.693 \\ \lambda &= \text{radioactive decay constant} = 1/\tau \\ &= 0.693 \left(\frac{1}{T_{1/2}} \right) \end{aligned}$$

The half-life is characteristic of the nuclear species. It can vary from billionths of a second to billions of years. The activity I , which is the rate of disintegrations per unit time, follows the identical decay law. It is equal to the number N present at any given time, multiplied by the decay constant λ :

$$I(t) = N(t)\lambda = I^0 e^{-t/\tau}$$

Note that the total number $N(t)$ of radioactive nuclei at any given time is the product of the mean life τ and the activity $I(t)$:

$$N(t) = I(t)\tau = N(t)\lambda\tau.$$

Next consider the case of the growth of a short-lived daughter (denoted by subscript 2) from a very-long-lived parent (subscript 1). Let the activity I_1 of the parent remain nearly constant. Then the rate of birth of the daughter is I_1 (equal to the rate of decay of the parent), and if the parent is the only nuclide present at time $t = 0$, the daughter activity at some later time t is:

$$I_2(t) = I_1^0 \left[1 - e^{-\lambda_2 t} \right].$$

Suppose there follows a grand-daughter (subscript 3). Then we find for the general case:

$$I_3(t) = \lambda_2 \lambda_3 I_1^0 \left[\frac{e^{-\lambda_1 t}}{(\lambda_2 - \lambda_1)(\lambda_3 - \lambda_1)} \right]$$

$$+ \frac{e^{-\lambda_2 t}}{(\lambda_1 - \lambda_2)(\lambda_3 - \lambda_2)} + \frac{e^{-\lambda_3 t}}{(\lambda_1 - \lambda_3)(\lambda_2 - \lambda_3)} \Bigg]$$

Unfortunately this mathematics yields very little direct insight. Consider a real example: In Fig. 2 (from Ref. 12) is shown the growth of activity of the three successive daughters RaA, RaB, and RaC from a unit activity of the parent ^{222}Rn . . . the half-lives of the three daughters are about 3, 27, and 20 minutes, respectively. The half-life of the parent ^{222}Rn is very much longer: 3.82 days or 5500 minutes.

Fig. 2. Growth of activity of the individual short-lived decay products in a constant source of radon having unit activity. If RaA, RaB, and RaC are in radioactive equilibrium with a constant source of Rn, and the Rn is removed at $t=0$, the subsequent decay produced is the up-to-down mirror image of these growth curves. Thus, for decay the vertical scale would read 0 to 1.0 from top to bottom, while for growth it reads 0 to 1.0 from bottom to top, as shown here and in Fig. 3 (from Evans, Ref. 12, with permission by McGraw Hill © 1969).

One crucial point to note is that when equilibrium is reached, all daughters have activities equal to that of the parent:

$$I_{\text{RaC}} = I_{\text{RaB}} = I_{\text{RaA}} = I_{\text{Rn}}$$

However, as we saw above, the number of atoms present in each species is proportional to the mean life, and thus in equilibrium, the number of atoms is in the ratio

$$\begin{aligned} N_{\text{Rn}} : N_{\text{RaA}} : N_{\text{RaB}} : N_{\text{RaC}} &= \tau_{\text{Rn}} : \tau_{\text{A}} : \tau_{\text{B}} : \tau_{\text{C}} \\ &= 5500 : 3 : 27 : 20. \end{aligned}$$

This is a very important concept to understand.

Next, suppose that, after equilibrium is established, the parent source is removed. Naturally, the daughter(s) will subsequently decay. It is easy to show (see Evans, Ref.12) that the decay is given by a direct inversion of the growth equations. Figure 2, it turns out, also shows the decay of RaA, RaB, and RaC after the parent Rn is removed . . . simply reverse the picture (upside-down mirror image) as described in the caption, and relabel the axes. This is done for the reader in Fig. 3.

Fig. 3. The inversion of Fig. 2 as described in its caption (from Evans, Ref. 12, with permission by McGraw-Hill © 1969).

Another useful insight which can be seen from simple considerations is the growth of activity on, say, an air filter. Assume a constant collection rate of radioactive dust from air containing constant activity. This case is identical to the case of activity growth of, say, RaA from a constant activity of its parent, Rn. The build-up follows a curve similar to the RaA plot in Fig. 2. If the pump is turned off, so that sampling stops, the decay of the activity on the filter follows Fig. 3, the upside-down version of Fig. 2, as previously discussed. This applies equally well to the ingrowth in a chemical vial of ^{90}Y (64 hours) from ^{90}Sr (28 years) after radiochemical purification of the strontium . . . only the time axis must be re-labelled to match the different half-lives.

4. Calibration of Instruments

Correct interpretation of monitoring results is based on the accurate calibration

of radiation measuring instruments. The reliability and accuracy of the measurements are also partly based on the procedures applied in calibration (Ref. 13). Discussions of the problems of instrument calibration can be found in the individual sections on instrumentation (e.g., Neutron Monitoring Instrumentation). As an introduction, we shall quote here from a discussion of general calibration problems in ICRU Report No. 20 (Ref. 3).

"The accuracy of a given instrument depends not only on the characteristics of the instrument but also on the care taken in its calibration and on the performance checks made during its use.

"Calibrations may be carried out in several ways:

- (1) By national standards laboratories using calibration standards of the highest possible accuracy (Refs.14, 15)
- (2) By major government, academic, research or industrial laboratories using institutional standards established by careful comparison with a national standard (Ref. 16).
- (3) By other groups using working standards (reference instruments) of lesser accuracy that are periodically calibrated by comparison with one of the above-mentioned national or institutional standards.

Only the last two will be discussed in this section.

"Calibration of instruments for health physics purposes involves the comparison of a given instrument with a reference instrument, or its irradiation in a field whose properties are either defined through use of a standard source, or which have been fully explored with a reference instrument. The calibration of the reference instrument may be determined by comparison with an institutional or national standard. (Ref. 14).

"The object of calibrating an instrument intended for routine use is to ensure that its accuracy is adequate under the conditions in which it is used. This requires that assessment be made of its response not only to the type and energy of radiation for which the instrument was designed, but also to other radiations which may be present in practice and contribute to the instrument reading. It also requires that mechanical and electrical integrity, and effects such as saturation, reproducibility, pulse pile-up, scale non-linearity, and range-change errors be examined. Since manufacturers may not have adequate facilities for calibrating instruments, it is desirable to check new instruments after delivery. If

the instrument has new design features, the check should be thorough, and include tests for the rejection of radiation which the instrument was not designed to measure, for pulse pile-up and for saturation. The effective center of the detector should be determined so that measurements of distance from the source of radiation can be made to this point; this is particularly important when the detector is situated within the instrument. This complete check need not be repeated for all instruments built to the same design. However, it is desirable to check every instrument for scale non-linearity and for range-change errors before putting it into service. The initial calibration may involve either a wide or narrow spectrum, or alternatively a complete energy response curve using a number of monochromatic sources may be required. The choice depends on the purpose for which the instrument is intended to be used and should test the claimed or advertised response characteristics.

"The frequency and scope of subsequent calibrations is governed by the rate at which components in the instrument age or become damaged. Unless there is evidence or experience to suggest that more frequent calibrations are desirable, yearly intervals are usually sufficient. Within these intervals simple constancy checks should be made to see that the instrument has not developed a calibration fault on any range. Such constancy checks, which are particularly important for instruments used only occasionally, can be made with jigs incorporating small sealed sources, or with leakage fields from a large installation (e.g., reactor or accelerator) operating under steady conditions. The most satisfactory form of constancy check is one in which the sensitive element is irradiated by the type of radiation for which the instrument is designed. It should be carried out initially just after an instrument has been calibrated.

"The characteristic of an instrument which is most likely to change over a period of time is sensitivity. Some instruments provide an adjustment so that the reading can be brought back to the correct value when changes occur. In general, however, such adjustments should be made only at a calibration center having the standards needed to re-check the instrument over its whole absorbed dose or exposure range; changing sensitivity may not always be linear over the entire range of the instrument. Some instruments are fitted with a switch position for testing the battery and thus high pressure gas ionization chambers should have their energy response accurately determined. This is because thin-walled ones tend to have significant over response in the low energy end of the spectrum. Low pressure, tissue equivalent chambers should be checked for unusual contamination from manufacturing processes."

The reader should be aware of the possible services that can be provided by the National Bureau of Standards (Ref. 20). Referral is also made to ANSI Standard N323 - 1978, "Radiation Protection Instrumentation Test and Calibration" (Ref. 18). ANSI N323 addresses several performance areas in a qualitative nature for the calibration and testing of portable radiation protection instruments. It is believed by some (Ref. 19) that compliance with the standard will be difficult to achieve.

5. Instrumental Response to Environmental Conditions

a. Temperature and Pressure Response.

The temperature response of detectors can be quite important particularly for scintillation counters. For example, with a sodium iodide crystal the pulse height decreases approximately 0.20%/°C over the temperature range from 30°C to 145°C; (see Fig. 4).

Fig. 4. Relative light output of NaI(Tl) as a function of temperature.

In addition to the changing of the sodium iodide phosphor itself, the photomultiplier tube may also show both a small gain change and the dark current noise will increase with increasing temperature. The gain change is somewhat dependent on the phototube type and, in fact, will vary from phototube to phototube of the same type. Because of the variation in phototube temperature gain change, one should always determine the temperature response of the scintillation detector.

Geiger tubes and proportional counters normally will show little effect when operated over normal ambient temperature range. Sealed

Table 4. Air density corrections. Instrument is calibrated in International Roentgens corrected to 0°C when used at 22°C and 760 mm mercury (Hg) barometric pressure. For temperatures other than 22°C and pressures other than 760 mm Hg., multiply the scale reading by the factor obtained from the following table.

Inches	Mm.	F. 60.8 C. 16	64.4 18	68.0 20	71.6 22	75.2 24	78.8 26	82.4 28	86.0 30	89.6 32	93.2 34	96.8 36	100.4 38	104.0 40
19.68	500	1.489	1.499	1.509	1.520	1.530	1.541	1.551	1.561	1.571	1.582	1.592	1.602	1.613
20.08	510	1.460	1.469	1.479	1.490	1.499	1.510	1.520	1.530	1.540	1.551	1.561	1.571	1.581
20.47	520	1.431	1.441	1.451	1.461	1.471	1.481	1.491	1.500	1.510	1.520	1.530	1.540	1.550
20.87	530	1.405	1.414	1.424	1.434	1.444	1.453	1.463	1.473	1.482	1.492	1.502	1.512	1.521
21.26	540	1.378	1.388	1.397	1.407	1.416	1.426	1.435	1.445	1.454	1.464	1.474	1.483	1.493
21.65	550	1.354	1.363	1.373	1.382	1.391	1.401	1.410	1.419	1.429	1.438	1.448	1.457	1.466
22.05	560	1.329	1.338	1.348	1.357	1.366	1.375	1.384	1.394	1.403	1.412	1.421	1.431	1.439
22.44	570	1.306	1.315	1.324	1.333	1.342	1.351	1.360	1.369	1.378	1.387	1.396	1.405	1.414
22.83	580	1.283	1.292	1.301	1.310	1.319	1.328	1.337	1.345	1.354	1.363	1.372	1.381	1.389
23.23	590	1.262	1.270	1.279	1.288	1.297	1.305	1.314	1.323	1.331	1.340	1.349	1.358	1.366
23.62	600	1.241	1.249	1.258	1.267	1.275	1.284	1.293	1.301	1.309	1.318	1.327	1.336	1.344
24.02	610	1.220	1.229	1.237	1.246	1.254	1.263	1.271	1.279	1.288	1.297	1.305	1.314	1.322
24.41	620	1.200	1.208	1.217	1.225	1.233	1.242	1.249	1.258	1.266	1.275	1.283	1.292	1.299
24.80	630	1.181	1.189	1.198	1.206	1.214	1.222	1.230	1.239	1.247	1.255	1.263	1.271	1.279
25.20	640	1.164	1.171	1.180	1.188	1.196	1.204	1.212	1.220	1.228	1.236	1.244	1.252	1.260
25.59	650	1.145	1.153	1.161	1.169	1.177	1.185	1.193	1.201	1.208	1.216	1.224	1.232	1.240
25.98	660	1.127	1.135	1.143	1.151	1.159	1.167	1.174	1.182	1.189	1.198	1.206	1.213	1.221
26.38	670	1.111	1.119	1.126	1.134	1.142	1.149	1.157	1.165	1.172	1.180	1.188	1.195	1.203
26.77	680	1.095	1.103	1.110	1.118	1.125	1.133	1.141	1.148	1.156	1.163	1.171	1.179	1.186
27.16	690	1.078	1.086	1.093	1.101	1.108	1.116	1.123	1.131	1.138	1.146	1.153	1.161	1.168
27.56	700	1.064	1.071	1.079	1.086	1.093	1.101	1.108	1.115	1.123	1.130	1.137	1.145	1.152
27.95	710	1.048	1.055	1.063	1.070	1.077	1.084	1.092	1.098	1.106	1.113	1.121	1.128	1.135
28.35	720	1.033	1.041	1.048	1.055	1.062	1.069	1.076	1.083	1.091	1.098	1.105	1.112	1.119
28.74	730	1.019	1.027	1.034	1.041	1.048	1.055	1.062	1.069	1.076	1.083	1.090	1.097	1.105
28.94	735	1.013	1.019	1.027	1.034	1.041	1.048	1.055	1.062	1.069	1.076	1.083	1.090	1.097
29.13	740	1.006	1.013	1.020	1.027	1.034	1.041	1.048	1.055	1.062	1.069	1.075	1.083	1.089
29.33	745	.999	1.006	1.013	1.020	1.027	1.034	1.040	1.048	1.054	1.061	1.068	1.075	1.082
29.53	750	.992	.999	1.006	1.013	1.020	1.027	1.033	1.040	1.047	1.054	1.061	1.068	1.075
29.72	755	.986	.993	1.000	1.007	1.014	1.021	1.027	1.034	1.041	1.048	1.055	1.062	1.068
29.92	760	.980	.986	.993	1.000	1.007	1.014	1.020	1.027	1.034	1.041	1.047	1.054	1.061
30.12	765	.972	.979	.986	.993	.999	1.006	1.013	1.020	1.026	1.033	1.040	1.047	1.054
30.31	770	.967	.973	.980	.987	.994	1.000	1.007	1.014	1.020	1.027	1.034	1.041	1.047
30.51	775	.961	.968	.974	.981	.987	.994	1.001	1.007	1.014	1.021	1.027	1.034	1.041
30.71	780	.954	.961	.967	.974	.980	.987	.994	1.000	1.007	1.014	1.020	1.027	1.033
30.90	785	.948	.955	.961	.968	.974	.981	.988	.994	1.001	1.007	1.014	1.021	1.027
31.10	790	.942	.949	.955	.962	.968	.975	.981	.988	.994	1.001	1.008	1.014	1.021

ion chambers will also show little change over the same range; however, an ion chamber open to the atmosphere will change according to the air density correction table (see Table 4). Under most conditions the temperature/pressure correction factor will be in the range $\pm 10\%$. This arises not so much from the temperature or pressure change at a given location, as from differences between local conditions and those where the instrument was calibrated. Most manufacturers who sell ion chamber instruments that are open to the atmosphere calibrate them to standard temperature and pressure. Under most conditions, this should not be any problem; however, at high elevations, such as Las Vegas, Denver, etc., there can be a considerable error because of the decreased barometric pressure. If the decreased barometric pressure is coupled with a sizable temperature change, there could be a -30% error.

b. Humidity Response. High humidity, although having little importance on results from individual measurements, can adversely affect instrument life. The Geiger counter, proportional counter and the scintillation counter show little reaction to humidity providing reasonable care is taken to keep their electrical connections dry. In general

one will obtain the best life from a survey meter if it is kept as dry as possible. When in an extremely high humidity area, it is wise to put several bags of silica gel inside of the survey meter.

6. Critical Pathway Analysis

The general steps of dose assessment are:

- The characterization of the radionuclides or radiation sources.
- Determination of radioactivity distributions.
- Determination of radiation incident on (and/or the radionuclide accumulation by) man.
- Determination of consequent dose to man.

By considering the emission of particular radionuclides from a nuclear facility, these elements can be illustrated. Table 5 shows the various steps in the inference of the dose from radionuclides deposited in man following transport (movement) in the terrestrial environment. The activity released per unit time

Table 5. Illustrative equilibrium pathway for internal radiation exposure for airborne radionuclide effluent (from NCRP, Ref. 17, with permission by National Council on Radiation Protection and Measurements © 1976).

Pathway Step	Variable	Product of Variables
Source	Source terms (Q)	Radioactive material released per unit time Q
Dispersion	Dispersion (χ/Q)	Radioactive material per unit volume $Q(\chi/Q)$
Deposition	Deposition velocity (V)	Radioactive material per unit area and unit time $V_d(\chi/Q)Q$
Cumulative Deposition	Accumulation time (τ)	Radioactive material per unit area $\tau V_d(\chi/Q)Q$
Environmental and Biological Transfer ^a	Transfer coefficient (T_r)	Radioactive material per unit mass of environmental or biological medium $T_r \tau V_d(\chi/Q)Q$
Human Intake	Intake rate (I)	Radioactive material intake per unit time $IT_r \tau V_d(\chi/Q)Q$
Human Absorbed Dose Rate	Intake rate to absorbed dose rate conversion factor (k)	Absorbed dose per unit time $kIT_r V_d(\chi/Q)Q$

^a T_r is simplified here; T_r is the product of a number of coefficients, e.g., soil to air, soil to grass, grass to animal, animal to animal, etc., and depends on time and on metabolic, physical and chemical variables.

to the environment is dispersed in the air, at least partially deposited on soil or vegetation, and partially taken in by man via ingestion.

This discussion emphasizes the pathways of practical interest in nuclear facilities' monitoring of liquid and airborne effluents. For pathways that could be critical, consideration is given to those points in the pathways that present opportunities for optimum measurement. What follows is based largely on NCRP Report 50 (Ref. 17).

a. External Irradiation

• Source → Man

Photon emitters are generally the most significant radiation sources because of their range, although betas may be important in

certain cases (e.g. ⁸⁵Kr release near a reactor fuel reprocessing plant). The proper evaluation of this type of pathway requires measurements of both the absorbed dose rate in free air and/or particle flux density as function of particle type and energy.

• Source → Water → Man

Radioactivity released as dissolved or suspended material in a liquid effluent into the environment can potentially cause direct radiation exposure to individuals who are either immersed in the water or near the water surface. Photon emitters are again of primary interest, although for total immersion the possibility of significant β -skin dose should be considered. Dose rates can be estimated either by radiation measurements or by calculation based on the radionuclide concentration in water.

- Source → Water → Sediments
→ Man

Since many radionuclides are preferentially deposited on exposed sediments on the shores of streams, rivers, and lakes or on tidally-washed estuaries or coastal areas this pathway differs from the preceding one. Some radionuclides for which this may be a significant pathway are ^{106}Ru , ^{95}Nb , ^{134}Cs , and ^{137}Cs (Ref. 17).

- Source → Air → Man

Immersion in air containing radioactive gases and aerosols will lead to external β and γ irradiation. If the cloud or plume is located nearby, then only γ irradiation occurs. "The determination of the integrated dose from such plumes by radiation measurements may require a long-term monitoring program because of the complex relationships between plume geometry and local meteorology." (Ref. 17). The NCRP indicates that this is a significant pathway for fuel reprocessing plants and boiling water reactors with short hold-up times (less than several hours). It is also one of the important pathways for large accidental releases.

- Source → Air → Soil → Man

Radioactivity released to the atmosphere can deposit on the ground adding significantly to the γ radiation field near the air-ground interface produced by naturally occurring sources. Either direct radiation measurement or a determination of the radionuclide distribution with soil depth may be used for dose assessment. This pathway could have great importance in a severe accident at a nuclear facility.

b. Internal Irradiation

- Source → Water → Man

If all age groups in a general population consume large quantities of local water this pathway could assume importance. The possible removal of radionuclides by water treatment plants should be investigated and measurements of radionuclide concentration in drinking water supplies should be carried out after any such treatment. Examples of this pathway are possible at sites near nuclear power reactors if the liquid effluent may reach drinking water supplies.

For an introduction to the vast literature concerning the import of radionuclides in the aquatic environment, the reader is referred to Ref. 21.

- Source → Water → Algae → Man

In some areas, people may consume food prepared from edible seaweed, so

this pathway may assume importance whenever radioactivity is discharged into bodies of water where such algae grow. Consideration should be given the possible concentration of certain radionuclides above the level in the surrounding water.

- Source → Water → Shellfish
→ Man

If radioactivity releases are made in the vicinity of shellfish farms, potential radiation exposure could result from the consumption of such shellfish.

- Source → Water → Fish → Man

"This may be the critical pathway in freshwater or whenever shellfish or algae in a marine or estuarine area are not consumed." (Ref. 17).

- Source → Water → Soil → Food
→ Man

If releases of radioactivity are made to bodies of water used for irrigation, this pathway is important. Radionuclides in water may also enter into man's food supply by means of uptake by animals foraging at the water's edge.

- Source → Air → Man

Inhaling radioactivity released to the atmosphere directly from a source or resuspended after deposition could result in its absorption in the lung or gastrointestinal tract. The physical state of the radionuclide determines absorption and subsequent distribution in the body for some airborne radionuclides. For example, tritium (HTO or HT) may enter the body by transpiration through the skin. This could be an important pathway during accidental releases from nuclear facilities.

- Source → Air → (Soil→)
Vegetation → Man

Airborne radionuclides attached to particulates are deposited on the surfaces of vegetation by dry deposition and by precipitation. "The ingestion of such vegetation is in general not an important dose pathway to man since the residence time is usually short." (Ref. 17) An exception would be a high deposition rate of radioiodine on leafy vegetables. Chemical and biological properties of the nuclide and of the soil determine uptake from the soil in the case of absorption of radionuclides into the plant system from the leaves or through the roots. Radionuclides accumulated in the plant are available for direct ingestion by man or for ingestion by grazing animals.

- Source → Air → (Soil→)
Vegetation → Meat or Milk
→ Man

This pathway is often important, notably for ^{90}Sr and ^{137}Cs in fallout from weapons tests and releases of ^{131}I at nuclear facilities. "The bulk of the dose from these internally-deposited radionuclides may be derived from milk and/or meat in the diet." (Ref. 17) Much information exists for this complex pathway about the various parameters and how radionuclides are transferred between the various media. Radionuclide concentration measurements in one medium are related to those in other media and then ultimately to absorbed dose via animal and then human ingestion. (Ref. 22).

c. Measurement Methodologies

- Dose from internally deposited radionuclides

A common feature of the internal radiation pathways discussed above is the transfer of radionuclides from medium to medium until they are inhaled or ingested by man. Studies have been performed to determine the extent to which this transfer takes place for particular radionuclides in certain pathways. Eisenbud (Ref. 23) has reviewed this subject.

The dose to a particular organ can be determined from models of radionuclide distribution in the body (Ref. 24a) once the intake of a radionuclide has been estimated. UNSCEAR (Ref. 24b) has given values for the important parameters in internal dose calculations. In the analysis of a dose pathway, the primary role of measurement is to provide an observation in the environment from which the identity of the responsible radionuclides and the size of the dose may be determined. The reliability of a dose estimate is best when measurements have been made of the radionuclide content in media along the pathway to man or in man himself since there are unavoidable uncertainties, including transfer and concentration factors in various media, dietary analyses, and calculations of organ dose per unit intake (Ref. 25).

NCRP (Ref. 17) gives the following suggested guidance for environmental measurement:

"a. The goal of environmental measurements directed toward assessing the dose from internally deposited radionuclides is to quantify their human intake rate and resulting organ concentration,

"b. Most measurements will be directed toward defining radionuclide concentrations in environmental media,

"c. In the absence of comprehensive dietary measurements, the experimental data

must be supplemented by a computational model of radionuclide transport from the environment to man,

"d. The identification of critical dose pathways is a prerequisite to the design of a rational measurement program and this identification requires a detailed specification of the radionuclide source,

"e. The choice of measurements to be made depends on the degree of accuracy desired in dose assessment, on practical limitations (e.g., sensitivity, discrimination, cost) of available techniques, and on the importance attached to the characterization of actual environmental radionuclide distributions,

"f. In the absence of direct measurements of man himself, the media consumed most directly by man should be measured to obtain optimal accuracy."

- Dose from externally-incident radiation

Measurements of environmental radiation fields or radioactivity distributions to infer dose to man involves the interpretation of measured quantities in terms of the radiation field and the application of appropriate dose models combining the physical parameters of the radiation field and the appropriate organ or whole body geometry. The proper interpretation of detector response involves discrimination among the various components of a radiation field. Although the characteristics of a particular environmental field (the composition, intensity, energy distribution, and angular distribution) cannot usually be directly measured, they generally can be inferred from other information. For example, measurements of the concentrations of natural radionuclides in near-surface soil at an outdoor location, coupled with the reasonable assumption of a homogeneous distribution with depth, is usually adequate to determine the natural γ -radiation absorbed dose rate in air above the soil. Furthermore, to infer the human organ dose from a knowledge of the environmental radiation field requires the application of appropriate conversion factors (e.g., ratio of absorbed dose in some organ to that in free air.) Using a detector whose response is directly proportional to a particular human organ dose would yield a more direct approach to dose assessment. Measurement inside a human phantom is an example; however, this approach is generally impractical and requires additional measurements in order to determine parameters of the radiation field.

A word of caution concerning realistic long-term estimates of exposure is that the absorbed doses to individuals, or to averages of populations, should take into account their movement and living habits. This is because indoor and outdoor radiation fields are gen-

erally different and may vary with time, location, and meteorology.

The NCRP (Ref. 17) suggests two feasible approaches to population absorbed dose assessment from external radiation. "In situ measurements of the radiation field can be made at representative indoor and outdoor locations, and the results weighted according to typical or average occupancy times. Alternatively, small sensitive dosimeters can be worn by a 'representative' sample of the population for sufficiently long periods of time to average the occupancy times in various radiation environments. This second method cannot be easily applied to the assessment of absorbed dose contributions from particular components of the total radiation environment, (e.g., from man-made radionuclides)."

Additional discussions of critical pathway analysis can be found in Refs. 25-27.

7. Reference Methods

The purpose of this brief section is primarily to alert the reader to those procedures recommended by organizations such as the NRC, EPA, DOE, IAEA, ASTM, APHA, etc., as reference methods for the determination of ionizing radiation.

The Nuclear Regulating Commission by issuing Regulatory Guides delineates techniques or provides guidance on pertinent matters under its jurisdiction. The guides that suggest appropriate radiation measurements and techniques are as follows:

- Regulatory Guide 1.21 - Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactivity in Liquid and Gaseous Effluents from Light Water Nuclear Power Plants (Ref. 37).
- Regulatory Guide 4.5 - Measurements of Radionuclides in the Environment -- Sampling and Analysis of Plutonium in Soil (Ref. 38).
- Regulatory Guide 4.6 - Measurements of Radionuclides in the Environment -- Strontium-89 and Strontium-90 Analysis (Ref. 39).
- Regulatory Guide 4.14 - Measuring, Evaluating, and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Airborne Effluent from Uranium Mills (Ref. 40).
- Regulatory Guide 4.16 - Measuring, Evaluating, and Reporting Radioactivity in Releases of Radioactive Material in Liquid and Airborne Effluents from Nuclear Fuel Proc-

essing and Fabrication Plants (Ref. 41).

Regulatory Guide 5.16 - Standard Methods for Chemical, Mass Spectrometric, Spectrochemical, Nuclear, and Radiochemical Analysis of Nuclear Grade Plutonium Nitrate Solutions and Plutonium Metal (Ref. 42).

The Department of Energy through its Environmental Measurements Laboratory (formerly HASL) issues a manual on measurement techniques that can be used for guidance (Ref. 29).

The Environmental Protection Agency in the Federal Register published Drinking Water Regulations for Radionuclides that suggests methods for analysis. Their recommendations are given in Table 6. In addition to the general guidance "Radiochemical Methodology for Drinking Water" (Ref. 30), the EPA also runs a quality control program on radiation measurements (Refs. 31, 32). In the future, they will put forth reference methods following a check of tentative methods (Ref. 33).

The U. S. Geological Survey also issues pertinent guidance to its laboratories concerning the measurement of gross α , gross β , ^{226}Ra , and ^{228}Ra measurements in water (Ref. 36).

At the international level, the IAEA has issued guidance (Ref. 34) on monitoring airborne and liquid radioactivity released from nuclear facilities to the environment. Its aim is to define the objectives of effluent monitoring programs and to provide guidance on the design of an adequate system and on the choice of appropriate monitoring methods and instruments for different types of nuclear fuel cycle facilities.

One good source of information for measurements in water and waste-water is Standard Methods for the Examination of Water and Wastewater by APHA (Ref. 28). The methods with "standard" status have been studied extensively and accepted as applicable within the limits of sensitivity, precision, and accuracy given. "Tentative" methods are those still under investigation that have not been evaluated fully or are not considered sufficiently tested at present to be designated "standard."

Another source of information on measurements is the American Society for Testing and Materials. They have published a set of ASTM Standards (Ref. 35).

The American National Standards Institute publishes voluntary standards such as "Method of Test for Tritium in Air" (Ref. 43), "Method for Measurement of Alpha Radioactivity of Water" (Ref. 44), and many more. Their most current catalog should be consulted (Ref. 45) for a complete listing.

Table 6. Procedures recommended by EPA.

- a) The methods specified in *Interim Radiochemical Methodology for Drinking Water*, Environmental Monitoring and Support Laboratory, EPA-600/4-75-008, USEPA, Cincinnati, Ohio 45268, or those listed below, are to be used to determine compliance with §§ 141.15 and 141.16 (radioactivity) except in cases where alternative methods have been approved in accordance with § 141.27.
- 1) Gross Alpha and Beta-Method 302 "Gross Alpha and Beta Radioactivity in Water" *Standard Methods for the Examination of Water and Wastewater*, 13th Edition, American Public Health Association, New York, N.Y., 1971.
 - 2) Total Radium-Method 304 "Radium in Water by Precipitation" *ibid*
 - 3) Radium-226 - Method 305 "Radium-226 by Radon in Water" *ibid*
 - 4) Strontium-89,90-Method 303 "Total Strontium and Strontium-90 in Water" *ibid*
 - 5) Tritium - Method 306 "Tritium in Water" *ibid*
 - 6) Cesium-134 - ASTM D-2459 "Gamma Spectrometry in Water," 1975 *Annual Book of ASTM Standards, Water and Atmospheric Analysis*, Part 31, American Society for Testing and Materials, Philadelphia, PA (1975)
 - 7) Uranium - ASTM D-2907 "Microquantities of Uranium in Water by Fluorometry," *ibid*
- b) When the identification and measurement of radionuclides other than those listed in paragraph (a) is required, the following references are to be used, except in cases where alternative methods have been approved in accordance with § 141.27.
- 1) *Procedures for Radiochemical Analysis of Nuclear Reactor Aqueous Solutions*, H. L. Krieger and S. Gold, EPA-R4-73-014. USEPA, Cincinnati, Ohio, May 1973.
 - 2) *HASL Procedure Manual*, Edited by John H. Harley. HASL 300, ERDA Health and Safety Laboratory, New York, N.Y., 1973.
- c) For the purpose of monitoring radioactivity concentrations in drinking water, the required sensitivity of the radioanalysis is defined in terms of a detection limit. The detection limit shall be that concentration which can be counted with a precision of plus or minus 100 percent at the 95 percent confidence level (1.96σ where σ is the standard deviation of the net counting rate of the sample).
- 1) To determine compliance with § 141.15(a) the detection limit shall not exceed 1 pCi/l. To determine compliance with § 141.15(b) the detection limits shall not exceed 3 pCi/l.
 - 2) To determine compliance with § 141.16 the detection limits shall not exceed the concentrations listed below:
Radionuclide	Detection limit
Tritium	1,000 pCi/l
Strontium-89	10 pCi/l
Strontium-90	2 pCi/l
Iodine-131	1 pCi/l
Cesium-134	10 pCi/l
Gross beta	4 pCi/l
Other radionuclides	1/10 of the applicable limit
- d) To judge compliance with the maximum contaminant levels listed in sections 141.15 and 141.16, averages of data shall be used and shall be rounded to the same number of significant figures as the maximum contaminant level for the substance in question.

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1. Introduction

This section on particle accelerators is intended to provide background information about the extent of radiation hazards due to increasing use of accelerators in modern technology. Included in the discussion are the types of radiation present and their measurement. For direct instrument comparisons the reader is referred to the appropriate sections (for example, "Neutron Monitoring", or "Beta Particle Instrumentation", or "X and Gamma Monitoring Instrumentation") elsewhere in this volume. In this spirit, the following section must be viewed as complementary to these instrumentation sections.

International and national conferences are regularly held on accelerator technology and applications. Several of the most important series are represented by the following proceedings: Ref. 1 from the 1979 Particle Accelerator Conference at San Francisco; Ref. 2 from the 1978 conference on Applications of Small Accelerators; and Ref. 3 from the 1974 International Conference on High Energy Accelerators, held at Stanford.

An international conference specifically on accelerator dosimetry was held at Stanford in 1969 (Ref. 4). A useful general reference on radiation protection at accelerators is the book by Patterson and Thomas (Ref. 5). Standards for radiation protection and monitoring are: ICRU Report No. 20, "Radiation Protection Instrumentation and Its Application" (Ref. 6); ANSI Standard N43.4 - 1969, "Radiological Safety in the Design and Operation of Particle Accelerators" (Ref. 7); and NCRP Report No. 51, "Radiation Protection Design Guidelines for 0.1 - 100 MeV Particle Accelerator Facilities" (Ref. 8).

Particle accelerators in the world today number in the thousands, with the great majority having maximum energies below 100 MeV. Figure 1 (from Ref. 9) indicates numerical growth until 1968. Since the early 1940's, the number had grown by about 10% annually. Approximately half of the accelerators are located in the United States. The 1968 breakdown of U. S. accelerators by type, energy range, and application is given in Fig. 2 (based on Ref. 9). In 1972, the U. S. number was found to be about 2600, as indicated in Fig. 3; however, the energy range covered is slightly different than in Fig. 2, ac-

Fig. 1. Worldwide particle accelerator population, 1930 - 1968. (Reproduced from Ref. 9).

counting for much of the numerical difference. Although no recent survey comparable to that of Ref. 9 is available, it is clear that the total number of accelerators has continued to increase, perhaps doubling during the 1970's.

As indicated in Fig. 2, accelerators are applied in various ways. Table 1 (from Ref. 9) gives a more detailed look at the expanding use of accelerators as of 1968. About 28% of the accelerators in the United States were used for nuclear science and engineering; about 35% in medical therapy and radiography; and 29% in the study of radiation effects. While the growth in nuclear science and engineering has been small in recent years, the medical usage is growing very rapidly, as is use in industry. Accelerator technology has just begun to be exploited in such diverse areas as hot-atom chemistry research; ion-beam polishing and surface-coating; energy-dispersive x-ray analysis; ion implantation; and chemical processing. The increasing portion of low energy accelerators used in industry has made systematic survey of numbers and application more difficult.

The accelerated particles are of two basic types: electrons and ions (including protons). Although there are a few machines (perhaps 2 or 3 dozen in this country) with energies in excess of 100 MeV, these are primarily of the research type, used in large, well-equipped laboratories. These machines range in energy all the way up

Fig. 2. U. S. particle accelerators, 1969 (see also Table 1).

Fig. 3. U. S. particle accelerators, estimate for 1972. (Reproduced from I. L. Morgan, Ref. 10, with permission of IEEE Editorial Department, © 1973.

to those of the Stanford linear accelerator (typically 22 GeV electrons, with a 40 GeV maximum), the Brookhaven Alternating-Gradient Synchrotron (33 GeV protons), and the Fermi National Accelerator Laboratory machine (500 GeV protons, soon to be increased to about 800 GeV).

The problems of radiation measurements around these few high-energy machines, which are both difficult and special, are being handled by teams of skilled experts; thus we shall concentrate here upon the lower energy machines. There are three reasons for this emphasis: (a) the numbers of low-energy machines are expanding rapidly; (b) in many cases they are being installed in medical or industrial sites where personnel have had little previous experience with high-energy ionizing radiation; and (c) with their special problems, the largest accelerators tend to have the best monitoring equipment in the hands of the most experienced personnel.

2. Sources of Radiation Exposure

The sources of exposure from accelerators can conveniently be divided as follows:

Table 1. U. S. Particle Accelerators, 1968 (adapted from Ref. 9).

CATEGORIES OF ACCELERATOR APPLICATION	1964	1968	% Incr.	% '68 Pop.
<u>Nuclear Science and Engineering</u>				
Nuclear structure and reaction research	108	108		
Neutron and photo-nuclear research	87	80		
Reactor physics, shielding, and engineering	23	41		
Injection into larger accelerators	21	22		
Teaching and training	43	46		
	282	297	5.	27.5
<u>X-Rays (Neutrons) for Therapy and Radiography</u>				
Radiation therapy (with x-rays, electrons, etc.).	101	198		
Industrial radiography (with x-rays)	133	176		
Neutrons for therapy or radiography	-	2		
	234	376	60.	34.7
<u>Radiation Effects</u>				
Research in chemistry, biology, foods technology	41	52		
Activation or scattered-particle analysis	58	104		
Radioisotope production	-	7		
Space-radiation simulation and effects	31	46		
Other research in radiation effects	95	106		
	225	315	40.	29.1
<u>Atomic and Solid-State Physics</u>				
Beam-foil optical spectroscopy	-	3		
Ion implantation and channeling	-	17		
Solid-state research and development (miscellaneous)	5	10		
Atom-ion interactions; hot-atom chemistry/physics	-	5		
	5	35	600.	3.2
<u>Radiation Processing, Industrial</u>				
Processing of chemicals	21	46		
Processing for biological effect	1	-		
Processing of solid-state devices and materials	4	2		
Radiation service facility	10	12		
	36	60	66.	5.5
Totals	782	1083	39% Av.	100.0%
Electron Accel.		598		55.0
Ion Accel.		485		45.0
		1083		100.0%

a. exposure from the direct beam or its halo or scatter

b. exposure from leakage of radiation through shielding walls

c. exposure to induced radioactivity from the activation of items such as the beam pipes, targets, and shielding walls

d. exposure from respiration of air containing radioactive particulates, or unintended ingestion of radioactivity from various sources.

In this section we emphasize the first three of the above sources. Except for a brief treatment in connection with the environmental impact of accelerators, the ingestion and respiration of radioactive nuclides is discussed primarily in Part IV. Thus, we shall deal primarily with external exposure. Although such exposure is primarily occupational in nature, some installations do produce small external levels in the general environment, as discussed at the end of this section. Another principal non-occupational exposure, not discussed explicitly here,

is that received intentionally by a medical or dental patient from accelerator-beam diagnostics or therapy. Section II.D discusses sources of x-radiation; which contribute the major dose from medical diagnostics and in many cases from accelerator therapy. However, therapy can take various forms, and only generic considerations that pertain to occupational exposure are stressed here. Patient exposures, on the other hand, are a medical consideration. Ref. 11 discusses medical electron linear accelerators and associated radiation safety considerations and guidelines.

We note here that at the large research accelerators, at which both interlock systems and shielding are well designed and maintained, the most important source of exposure is usually the induced radioactivity around beam pipes and targeting areas, rather than the beam or its secondaries (Ref. 12).

In the case of occupational exposure, the relevant Maximum Permissible Exposure guidelines (discussed in more detail in Part I of this volume) are basically a limit of 5 rem/year, with no more than 3 rem to be received in any 13-week quarter. As always, exposures are to be kept as low as is reasonably achievable.

3. Safety Procedures - Licensing and Compliance Practices

Accelerators fall into three broad categories: first those operated by the U. S. Department of Energy (DOE) and its prime contractors and licensees; second, those operated in "agreement states" under state supervision; and third, those in non-agreement states. An "agreement state" is one where the state has agreed to certain Nuclear Regulatory Commission stipulations (Ref. 13) on the licensing and supervision of non-governmental radiation usage, and has undertaken this supervisory task. There were 25 such states in 1979. In "non-agreement" states, the state has either not requested the authority or has been unable to obtain such NRC approval (the latter is rare); even in these states, associated use of radioactivity is typically overseen to some degree by the state. Reference 14 surveys state programs for particle accelerators as of 1976.

Agreement states typically have programs of registration, licensing and compliance, but - as discussed in Ref.

14 - procedures vary from state to state, as may the state agencies involved. For example, a state may require licensing only for those accelerators from which significant activation occurs, thereby excluding many of the smaller installations, such as electron accelerators below about 8 MeV. The responsibility for supervising compliance may be divided according to application, e.g., medical, instructional, industrial. And finally, registration of new accelerators may be required of the owner upon delivery, independently of a license requirement. After registration (or upon licensing), an appropriate state agency may conduct an inspection to ensure that proposed shielding is adequate, that appropriate monitoring will exist, that records will be kept properly, and that the responsibility for operation rests with well-informed personnel. Subsequently, routine visits at fixed intervals may be made to ascertain that compliance with licensing requirements is continuing.

The responsibility for safe installation of an accelerator rests ultimately with the employer or user, not with the manufacturer. However, the manufacturer may be liable for product defects and for design inadequacies and is responsible for meeting applicable local, state, and federal rules, regulations, and design standards.

The adequacy of the shielding and the resulting radiation levels in various places on and near the installation are often determined with the aid of a consulting health physicist. The consultant will usually make quite complete survey measurements with sophisticated instrumentation and offer advice as well on the choice of monitoring instrumentation for later (routine) survey work. The day-to-day monitoring is then performed with less complex instrumentation by the appropriate on-site personnel, relying heavily upon the basic measurements of the original consultant.

The arrangement described above (which is probably typical of much practice in this field nation-wide) is the result of a compromise among a number of factors. First, a complete survey by a qualified health physicist is essential; it is required for initial licensing and it is not done by the state itself due to the lack of financial resources. Second, the typical safety officer at a hospital or industrial plant is often an industrial hy-

gientist by training, whose experience with ionizing radiation may not be adequate or up-to-date for the initial licensing requirements, but may suffice for routine monitoring. Third, the budget for monitoring instrumentation for such radiation is sometimes much less than would be desirable. Finally, the types of radiation fields which exist in these situations often change very little with time, so that frequent extensive re-evaluations of the levels are often unnecessary.

The exceptions are in applications where the operating conditions change frequently. Thus, research accelerators use a variety of beam configurations and a variety of targeting and detection equipment, in contrast to a typical medical therapy installation where there may be few important changes in operation over a period of several years.

Perhaps the most compelling reason for engaging a consulting health physicist is the complexity of the considerations in making meaningful measurements of the radiation levels and associated hazards. It is therefore important for every installation to retain the services of a consulting health physicist. The measurement problems will be discussed in the next section.

4. Special Measurement Aspects

a. Duty Cycle. "Duty cycle", defined as the fraction of the time in which the beam is on, is the term introduced because many accelerators only deliver fully accelerated beams during a fraction of the available time. One of the "worst" duty cycles is that typical of the high-energy electron linear accelerators (linacs): thus at the 22 GeV Stanford Linear Accelerator the beam occurs in bursts of 1 to 2 microseconds or at a repetition rate of 360 bursts/sec. The beam is present for less than one-thousandth of the time. Another typical accelerator type is the high-energy proton synchrotron, with useful beams for perhaps 5 to 10% of the time. At the other extreme are the low-energy Van de Graaff and Cockcroft-Walton accelerators, in which the beam is on essentially continuously. Superimposed on the beam-pulse structure just described is the "rf-structure" present in essentially all of the machines whose operation relies upon the synchronous principle. The rf accelerating voltage imposes its own characteristic periodic structure upon the accelerated beam, which can range

in frequency from a few to a few hundred Megahertz. The current in each rf packet may occupy only a small fraction of the available time in an rf cycle; 5% to 30% is typical.

The time structure of the beam is a crucial consideration in the choice and operation of monitoring instrumentation. Typical radiation fields during beam operation, even outside of shielding walls, are very closely correlated in time with the pulse structure of the primary beam, i.e., within a few nanoseconds. The main exception to this rule is thermal neutrons; depending upon geometrical and shielding variations, as much as several milliseconds can elapse before the thermal neutron flux has died away. Counters (for example gas proportional, scintillation, G-M counters) are particularly susceptible to incorrect measurements of the dose with low duty cycle, since in some cases they can have dead-times of many microseconds. Ionization chambers are somewhat less sensitive if operated so that the voltage is high enough to overcome much of the columnar recombination (recombination of positive ions and electrons within a single track before they are separated sufficiently by the applied voltage). In using ionization chambers, it is important to determine the extent of this latter problem and of volume recombination by calibrating in an actual pulsed source situation and varying the collector voltage to insure saturation. Instruments with no sensitivity with regard to duty cycle are, of course, activation/counting systems, such as counting the radioactivity resulting from the activation of foils containing gold, indium, carbon, etc.

There are two methods for circumventing the problem of poor duty cycles. The first is to use a measurement technique that is intrinsically an integrating method (such as a photographic film, a TLD chip, or perhaps an ion chamber or other electronic instrument operated in current-integration mode rather than pulsed mode). The second is to alter the source appropriately: a sufficient reduction in intensity or an artificial lengthening of the beam pulse structure might be feasible, although these factors are usually governed by considerations other than radiation protection monitoring.

b. Mixed Fields. Another key concept in the discussion of measurement problems around accelerators is the idea of mixed fields. External radia-

tion may consist of a number of particle species over a wide range of energies, with fluxes of variable spatial distribution, and sometimes with unusual duty cycle. All of these considerations immensely complicate the job of measuring dose equivalent levels.

Fortunately, most shielding is sufficiently thick that exposure to the direct beam or its secondaries does not present a problem. A proper interlock system should prevent access to the beam areas while the beam is on, and shielding is usually sufficient so that charged particle leakage is extremely small. Thus, outside the shielding walls of well shielded machines, the main radiation hazard during beam-on operations is usually from neutral radiation; the two dominant species are neutrons and photons (x or gamma-rays). It is also generally true that photons dominate the problem at electron accelerators, while at proton and ion machines the neutron flux is the dominant feature of the radiation field (Ref. 15). The difficulty in monitoring neutral particles is primarily due to their behavior when they interact with matter; indeed the differences between neutrons and photons as classes may often be less troublesome than the differences as a function of energy of either one of the species by itself.

This leads to the following statement: the most important task in the monitoring of external radiation levels around an operating accelerator is the evaluation of the dose equivalent rate due to neutrons and photons by the quantitative measurement of their flux densities as a function of energy. To this must be added the task, usually both easier and of less importance, of measuring the flux density of charged particles.

With regard to accuracy, a useful viewpoint has been expressed by ICRU Report No. 20 (Ref. 6): "Because the dose limits recommended by ICRP have been conservatively derived, great accuracy in radiation protection measurements appears unwarranted. It is suggested that when the MADE (maximum dose equivalent) is comparable to the Maximum Permissible Dose (MPD), an accuracy of $\pm 30\%$ be achieved."

The ideal detector (of photons and neutrons in a mixed field) would:

1. separate the two species and determine their flux densities independently;

2. weight the fluxes by the appropriate quality factors (QF), which have an energy dependence in the case of neutrons, and

3. provide a direct read-out of the flux densities (particles/cm²), the absorbed dose (rad), and the dose equivalent (rem) for each species separately.

Unfortunately, no such ideal instrument exists. The closest approximation is the use of several instruments, which are sensitive to one or more of the particles or energy regions but are reliably insensitive to others. In combination, these can be used to give a reasonable picture of the radiation field.

There are two basic pieces of information that one wants to know: (a) the particle constituents of the radiation field as a function of energy (for example, neutrons per cm. sec.- MeV); (b) the total dose equivalent (rem). While it is true that sufficiently accurate knowledge of (a) can yield numbers for (b), the reverse is not true (Ref. 6). However, considerable effort has been devoted in recent years toward the development of instrumentation that reads directly in rem, especially for the fast neutron fields which are so troublesome around most proton accelerators. It should be emphasized that, in our opinion, a rem-meter is useful in only a limited class of circumstances and is not a substitute for more detailed knowledge of the constituents contributing to the total rem being measured. There are several reasons for this; the most obvious is that if radiation levels need to be reduced, one has no indication of the predominant constituents. Is increased shielding necessary? If so, what kind and where? Is it primarily thermal neutrons? Is low-energy photon radiation properly accounted for? One real danger is that the composition of the radiation might change without a correspondingly significant change in the rem value (for example, the thermalization of fast neutrons by some change in operating conditions. With this in mind we shall deal separately with neutron and photon monitoring instrumentation, and then follow with discussions of other particles and of induced radioactivity.

c. Neutrons. One key design problem for neutron-monitoring in the vicinity of accelerators is to make an instrument which is insensitive to pho-

tons, but with great neutron sensitivity. As mentioned above, we shall defer detailed discussion of such instruments to the section on Neutron Monitoring. However, a few comments are pertinent here. For example, there are several types of accelerators whose radiation fields outside of the shielding walls normally contain very few neutrons of any energy. This is almost always the case around electron machines of low energy (~ 10 MeV) and is even the case at the few very high energy (GeV-range) machines (Ref. 15). The reason for this is that at electron machines the dominant process of primary-beam energy loss is cascade Bremsstrahlung, in which electromagnetic showers build up in the shielding. Shielding is usually designed primarily to cope with this problem, but even so the dominant remaining backgrounds are photons. At low energies, neutrons result primarily from electro or photo-disintegration of nuclei, but the thresholds are high (several MeV to several tens of MeV) for the nuclei which comprise most shielding walls, and the disintegration cross sections are relatively small. However, this general statement must be applied with some caution. For example, the use of a deuterium or tritium target (binding energies 2.3 and 8.5 MeV, respectively) is a guaranteed way to increase the neutron component of the radiation field, and if measurements with other targets show little or no neutron component, this will not necessarily remain true when a target of deuterium or tritium is placed in the beam. Likewise, change of collimation material can be a problem; thus while lead is commonly used in collimators, it is important to realize that substitution of uranium for lead can increase neutron backgrounds many-fold, due to beam-induced fission.

When changes are made in operating procedures and one requires a rapid check on the possible increase in thermal neutron levels, a well-known method is the indium-foil technique, because indium has an extremely large thermal-neutron capture cross-section. Also, intermediate-energy and fast components can be rapidly assessed by surrounding the indium with a moderator, such as polyethylene. Both of these techniques are described in "Neutron Monitoring".

Finally, a quick method for ascertaining an "upper limit" estimate of dose-equivalent is the use of a tissue-equivalent ion chamber (with various absorbers to study the problem of particle equilibrium) to measure total ab-

sorbed dose (rad). This can then be assigned a QF of 10 (essentially always an over-estimate) to determine dose-equivalent (rem) in a rough way.

d. Photons. As in the neutron case, we refer for more detailed information on instrumentation to the separate sections devoted to photons, i.e., x and gamma rays. Here, we shall discuss only some aspects of the problem which are unique or relevant to the case of accelerators.

For photon measurements one does not have the problem of quality factor dependence, since $QF = 1$ for all photon energies. However, the possible energy spectrum of gamma-rays and x-rays has to be considered in the choice of instruments. Thus, while the ion chamber is the most common x and gamma-ray measuring instrument in the energy region from about 50 keV to about 3 MeV, it is not usually an appropriate instrument outside of this range. On the high-energy end it is limited where it becomes impractical to make chamber walls of a thickness sufficient to establish particle equilibrium. On the low-energy end, the opposite is true: the limitation occurs when the wall cannot be made thin enough for the establishment of electronic equilibrium. However, ion chambers are often used where photon energies extend from well above 3 MeV down even into the 10 to 15 keV range, and there is not usually much special need around MeV-sized accelerators to measure the low-energy x-radiation separately.

Although the response of most ion chambers is not proportional to absorbed dose for energies above a few MeV, they are still the instrument usually used where higher-energy photons are present or suspected. When very high-energy (even GeV level) electromagnetic showers develop in shielding, the primary energy is successively degraded into more and more particles of lower and lower energy and eventually the process reaches a peak with regard to the number of particles in the shower. The absorption processes of photons in shielding are such that most low-energy photons are absorbed in rather short distances after their production. However, for high-energy photons, i.e., above about 5 or 10 MeV, the absorption coefficients are both small and relatively constant. When one of these higher-energy photons does interact, energy is then lost quite quickly due to production of lower energy secondaries of shorter range.

Thus, once equilibrium is reached, the fall-off of intensity with increasing penetration is slow, and governed by the high-energy attenuation probabilities, despite the fact that at any point the dominant particle flux (and absorbed dose) are from low-energy particles. Thus the low-energy flux, the one of most interest from the radiological point of view, remains proportional to the high energy photon progenitors, and it is often not necessary to measure the latter directly. This is why ionization chambers, whose sensitivity is poor above a few MeV, are nevertheless adequate instruments for absorbed dose in equilibrium photon radiation fields.

If it is suspected that non-equilibrium high-energy components are present, the best check is to add absorber around the ion chamber in stages, up to about 30 g/cm². In equilibrium fields, there should be a small, monotonic decrease in instrument response, while in non-equilibrium situations an actual build-up of response may occur as mass is added.

e. Charged Particles. For all well-shielded accelerators, the charge-particle fluxes are almost insignificant compared to the neutral fluxes. We exclude from consideration those low energy charged particles that are in equilibrium with the dominant flux of neutral particles and shall discuss here only those few situations in which charged-particle backgrounds can be important.

In the very high-energy regime, there is a significant yield of muons. These come mainly from decay of pions (at proton accelerators) or from pair production (at electron accelerators). Muons are a problem because they interact with shielding wall materials almost exclusively through ionization loss. Below a few GeV, muons present little problem, since the shielding necessary for protection against nuclear cascade leakage is sufficient to stop muons by range. The higher the machine intensity below a few GeV, the more nearly true this statement becomes since the shielding requirements become more stringent. At energies above a few GeV (this includes only about 5 machines in this country, as well as several abroad) muons have such long ranges that their effective removal by shielding is difficult and expensive. Thus at the Brookhaven AGS (33 GeV proton accelerator) there were some initial operational difficulties due to

muon backgrounds (Ref. 16). At Fermilab, a penetrating muon beam is one of the more significant sources of radiation off site (Ref. 17).

Another source of charged particles is possible pin-holes or cracks in the shielding. These are especially difficult to monitor, and in common practice the only way to determine charged particle fluxes (above, say, a few MeV) is through use of a scintillation-counter coincidence telescope. This is very seldom done, even in large, sophisticated laboratories, because it is seldom a significant problem. Even in these cases, the telescopes are essentially always assembled from instruments at hand. Integral countertelescopes are not commercially available.

To summarize, there is very little instrumentation presently available for the measurement of charged particle fluxes; this is due to the rather low level of typical backgrounds (except in a few rare cases). It would appear that despite this lack in instrumentation, there is probably no need to assign high priority to its development.

5. Universal Dose-Equivalent Instruments

The measurement of dose-equivalent becomes difficult in mixed fields. This has led to the design of so-called "universal dose-equivalent instruments", three of which have been described in ICRU Report No. 20 (Ref. 6). Each of these instruments measures LET or quality factor, so that it must be used in conjunction with a separate measurement of absorbed dose in order to determine dose-equivalent. The following has been taken directly from Report No. 20. The references cited in the quotation are listed as Ref. 18 to 29 of this text. Even as of 1978, instruments of this type were not available commercially.

"For accelerator applications, an instrument giving dose equivalent for all types of radiations would be very useful. Such instruments are not commercially available but have been developed and used at a number of laboratories. Three types have been developed. The first is a spherical linear energy transfer (LET) spectrometer, the second is based on columnar recombination in a gas, and the third on the same phenomenon in a liquid. Although these methods are presently difficult to apply, improvements resulting in practical field instruments are to be

anticipated.

"The distribution of dose with LET may be determined by means of the spherical LET spectrometer (Rossi and Rosenzweig, 1955 and 1955a; Rossi, et al., 1962). The detector employed is a spherical proportional counter of tissue equivalent plastic containing tissue equivalent gas at low pressure. The energy deposited by individual charged particles traversing the interior of the counter depends on the product of LET and the length of the track intercepted. The resulting spectrum of pulse heights can be converted to a spectrum of LET by appropriate mathematical treatment. A limitation is imposed on the operation of the instrument by the requirement that particles traversing the cavity incur only small changes in LET and that the particle trajectories be essentially straight lines. This limitation and the fact that the proportional counter has a finite recovery time, which limits the particle fluence rate which can be measured, restrict the applicability of the device. Since the instrument is functional for essentially all charged particles with a quality factor of 1 or more, and since the total tissue absorbed dose may be determined with a tissue equivalent ionization chamber, a combination of these two instruments may be used to determine dose equivalent for any radiation. This method falls short of being ideal in two respects: first, it is quite complicated in application and, second, it does not give values of dose equivalent directly. Considerable progress, however, has been made towards overcoming these difficulties (Baum, 1967; Baum et al., 1970).

"The phenomenon of columnar recombination has been utilized to obtain an indication of effective quality factor for an unknown mixture of radiations (Zielczynski, 1963; Sullivan and Baarli, 1963). The method uses a simple parallel plate ionization chamber which is operated at 6 atmospheres of pressure of tissue equivalent gas and field strengths of up to 2,000 V cm⁻¹. The response of such a chamber can be deduced from existing theories of columnar ionization. The corrected ionization current is given by the equation $I = KV^n$, where K is a function of absorbed dose rate and type of radiation, V is the electric field strength and n is an index depending on the LET of the radiation used. The index n, within limits acceptable for practical application, is independent

of dose rate and the direction of the radiation, and directly proportional to quality factor. In mixed radiation fields the measured value of n would be an average value for the various components, if the collected ionization current were directly proportional to the rate of energy absorbed. This, however, is not the case because the collection efficiency varies with the LET of the ionizing particles. However, the error due to this effect is thought to be no more than 20% for unknown fields of mixed radiations. More recently, values of quality factor and dose equivalent have been obtained by comparison of the current in two such ionization chambers operated with widely different collecting potentials (Sullivan, 1964; Distenfeld and Markoe, 1965); the use of multiplate chambers has also been tested (Zielczynski, et al., 1964). While these developments are still experimental and not commercially available, they are promising and, like the LET spectrometer approach discussed above, may lead to a practical dose equivalent meter applicable to any type of radiation.

"Work on the basic properties of liquid filled ionization chambers (Blanc, et al., 1963; Ladu, et al., 1967), indicated that, by use of a property of liquid dielectrics, it may be possible to determine the effective quality factor of a mixed radiation field. For this type of detector, the ionization current (I) is said to be a linearly increasing function of the collecting voltage V for sufficiently high values of the applied electric field i.e., $I = R(hV + k)$, where R is the intensity of the radiation and h and k are two constants whose values depend upon the nature of the radiation, the chamber geometry and the dielectric. Plots of I vs V, when extrapolated to zero current, will intersect the V axis at $-(k/h)$ and this point of interception has been shown to depend on quality factor. Dielectrics that have been used are n-pentane, and iso-octane, the latter being some three times more sensitive than other dielectrics. The system has potentialities but suffers from the inherent instability experienced with current measurements in liquids. Also, further development is required to study the consequences of different energy spectra and dose ranges."

6. Induced Radioactivity

The largest source of personnel exposure around accelerators is often

from radioactivity induced in such materials as beam pipes, targets, and shielding walls. This exposure problem usually requires portable monitoring instrumentation: the task is to determine the level of activity and the dosage received by personnel at work on and around these "hot" areas. The particles principally involved in this case are the familiar products of nuclide decay, i.e., beta, gamma, and x rays of low energy and well-defined character. Heavily-ionizing alphas have such small range that, if not ingested, they are only a possible hazard to the skin upon contact; moreover, except for nuclear chemistry experiments, there is typically little alpha activity at accelerators. The typical nuclear betas also have short range. The gammas, of course, have penetrating power but are not difficult to measure. There are normally no neutron emitters among these sources. There exist many types of instruments for this measurement problem, and a discussion of these can be found elsewhere in this volume.

A few comments are in order, however. First, there are no duty cycle considerations when measuring induced activity; this simplifies the problem considerably. Second, care must be taken to measure activity exactly where the personnel might be exposed to it. Thus, if "hot" beam pipes or targets are being handled, one needs to measure activity at the surface. Third, radiation levels are highest immediately after beam turn-off, so that it may be wise to delay access to the area in question to allow some of the high-activity but short-lived components to decay away. In this regard, one ought to carry a portable meter whenever access is made to an area hot enough to be even close to maximum permissible levels, inasmuch as previous measurements may have missed a particular area. Alternatively, an accident (e.g., a small piece of material may have slipped into a beam line) can produce dangerously high levels not previously present. Finally, as mentioned earlier, care must be taken whenever a new or different material is exposed to the beam for the first time since unsuspected high levels of activity may occur from new and different activation processes.

7. Environmental Impact

To the extent that radiation or radionuclides can escape from an accelerator facility, these machines can

have a radiological impact on the environment. For very low energy machines, shielding can relatively easily control the amount of prompt radiation reaching the environment. Moreover, radionuclides produced in the course of operation have little potential for reaching the environment if ordinary caution is exercised.

High energy accelerators, on the other hand, produce very intense radiation fields, so that even with massive, carefully designed shielding, sufficient radiation can escape to require monitoring. Moreover, these fields can produce radioactivity either in the environment or in a position to escape thereto. From Ref. 30:

"In summary, the four possible radiological environmental impacts due to high-energy accelerators, in order of apparent importance are;

1. The production of "prompt" radiation fields during accelerator operation;

2. The production of radionuclides and noxious chemicals (such as ozone and nitric oxide) in the air in the accelerator vault and their subsequent release;

3. The production of radionuclides in the soil and ground water near the accelerator, with the possibility of infiltration in the underlying ground water systems and subsequent migration from the accelerator site into wells or other sources of potable water;

4. The induction of long-lived radionuclides in accelerator components which may subsequently be "recycled" with the consequent transmission of radionuclides into the general environment; this would be particularly true in the case of valuable metals such as copper, brass, iron, or aluminum.

"Thus, we see that the environmental impact of high-energy accelerators is different in character from most types of nuclear installations. The predominant source of population exposure is due to the radiation field produced during operation rather than due to the leakage of radionuclides into the environment.

"It is only in the past 5 or 6 years that the relative importance of radiological impact has been placed in this order. For many years, it was believed that the production of radionu-

clides in the soil and ground water would be the second most important source of impact. The fact that we now understand that the potential for radioactive contamination of ground water is very small is an important step forward in our understanding of accelerator-produced radiation."

High energy accelerators are intense neutron sources, so that associated environmental monitoring programs are usually devoted largely to neutron monitoring (discussed in Section 7II.F). The dose to surrounding populations is dominated by prompt radiation, usually consisting largely of neutrons, although in some cases photons or muons may contribute significantly. Reference 30 notes that, for the large accelerator; facilities mentioned in the introduction, a population dose of 50 to 1000 person-rem is incurred for each rem of dose equivalent at the fence post. This typically implies on the order of 10 person-rem/year due to a large accelerator, about the same population dose as is typically ascribed to routine operation of a new large nuclear power plant (see Section II.B). Like nuclear power plants, environmental impact statements are required for new high-energy accelerators in the United States. In any case, existing U. S. high-energy accelerators, which are operated for the Department of Energy, file annual environmental monitoring reports (see, for example, Ref. 31).

8. Summary

The purpose of this section on particle accelerators has been to outline (1) the extent of their use today in research, medicine, and industry; (2) the various classes of accelerators

and the typical radiation fields associated with them; (3) the types of measurement problems in a typical mixed field situation; and (4) some (brief) discussion of the relevant instrumentation for monitoring external radiation levels. For information about accelerator design and principles of operation, there are several excellent reviews in the literature (Ref. 32). The reader is referred for detailed discussions about instrumentation to the relevant separate sections.

The overall picture in terms of low-energy accelerator instrumentation may be summarized as follows: While instrumentation generally exists for the monitoring task, it is usually complicated enough to require an experienced health physicist for complete analysis of a given radiation situation. In many smaller installations this means that an expert must be retained on a consulting basis. Once a detailed survey of the radiation field's components has been made, less expensive and simpler survey devices are usually sufficient for routine monitoring.

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B. Nuclear Reactors

1. Introduction

The purpose of this section is to provide background information on the types and levels of radioactivity concentrations and radiation fields associated with the use of nuclear reactors and to describe the instrumentation available for measurement of these concentrations and fields. For detailed information on instruments, the reader is referred to the appropriate sections (such as "Neutron Monitoring") elsewhere in this volume; the following discussion must be viewed as complementary to the instrumentation sections.

The present discussion is intended to acquaint the reader with the radiation levels associated with the operation of nuclear power plants, to put their hazards in perspective, and to serve as an introduction to instrumentation suitable for making measurements to determine the effects of reactor operation on radiation doses or radioactivity levels. This discussion depends directly on the vast amount of literature which is now available on the subject of radioactivity from nuclear power plants. A representative selection of this literature is cited.

Certainly the most important sources of information on emissions from power plants are the utilities that operate the plants, and the federal agencies involved in their monitoring and licensing. The primary agency in this respect is the U. S. Nuclear Regulatory Commission (NRC), which is responsible for the licensing of commercial nuclear facilities and which issues an Environmental Statement for each proposed nuclear power plant. All such statements are available at NRC libraries and from the National Technical Information Service. The NRC and the U. S. Environmental Protection Agency (EPA) have also issued Environmental Statements for important actions connected with the operation of nuclear power plants. Of particular interest are the EPA statement related to their standard on emissions from the uranium fuel cycle¹ and the NRC statement on the use of "mixed oxide" (i.e., mixtures of uranium and plutonium) fuel in light water reactors.²

Figure 1 shows the locations of commercial nuclear power plants in the United States as of December 1977.³ The total operable nuclear electrical generating capacity given for that date was 46 gigawatts, with another 163 gigawatts under construction or on order. It is expected that the total U. S. electrical

Fig. 1. United States commercial nuclear power plants. (Reproduced from Ref. 3).

generating capacity will rise substantially in the near future and that nuclear power plants will constitute an important part of the increase. Making precise predictions is not possible, however, both because of alterations in the perceived need for new electrical capacity and because of the various possible responses to this need. However, planning for the future is important because of the long time between the decision to build a power plant and the completion of the plant. For nuclear power plants, this time is now about ten years.

Virtually all of the large power reactors in this country are cooled with ordinary ("light") water. Light water reactors are made in two varieties: the boiling water reactor (BWR) and the pressurized water reactor (PWR). Other possible types of reactors include heavy water reactors, gas-cooled reactors, liquid metal fast breeder reactors, and a few others. Various types of power reactors are discussed briefly in the next section.

In addition to large power reactors, which now tend to be designed for about 1 gigawatt output, there are hundreds of smaller reactors in the United States, plus the large number of operational military reactors (such as used in submarines). These smaller reactors are typically used for experimental purposes, including research and teaching not necessarily connected with the nuclear power program. In addition, many of the small early power reactors served as pilot and demonstration plants for the large scale reactors now being built. The considerations set forth below on reactor emissions and monitoring requirements pertain primarily to large commercial nuclear power plants. However, it will be obvious that many of the same considerations apply to these smaller reactors.

Nuclear electric generation obviously involves operation of nuclear power plants, but with this are associated many auxiliary activities, including mining and milling (see Section II.E), transportation, and fuel reprocessing/waste management (see Section II.C). Each facility must comply with both radiation protection standards and more general standards on conventional emissions, heat, noise, occupational protection, etc. A principal force behind careful analysis of the overall impacts of proposed facilities is the National Environmental Policy Act, which requires the preparation of environmental statements in support of major federal actions, including the licensing of nuclear facilities.

Various regulatory agencies have responsibilities where nuclear power is concerned. As has been pointed out, federal agencies regulating the operation of domestic nuclear facilities in the United States are the U. S. Nuclear Regulatory Commission (NRC) and the U.S. Environmental Protection Agency (EPA). The

NRC has responsibility for licensing and regulating individual facilities, but the EPA has responsibility for limiting environmental impacts. As a result, the emission and public exposure limits imposed by the NRC are expected to be consistent with EPA regulations. The NRC also regulates exposure of workers at nuclear facilities. In many occupational matters, the Occupational Safety and Health Administration and state industrial safety departments also have jurisdiction. State and local agencies together with the above mentioned organizations exercise authority over the siting of nuclear facilities. Many of the radiation protection standards applicable to nuclear power facilities, including power plants, were discussed in Section I.D. on radiation protection.

The manner in which the NRC regulates domestic nuclear facilities is specified in Title 10 of the Code of Federal Regulations (CFR). The parts of major interest in connection with nuclear power plants are Part 20 (Standards for Protection Against Radiation), Part 50 (Licensing of Production and Utilization Facilities), and Part 100 (Reactor Site Criteria). In addition to 10 CFR, the NRC review process depends heavily on NRC Regulatory Guides, which specify how the applicant for a license may comply with the requirements of 10 CFR. Appendix A indicates the contents of 10 CFR 20, 50, and 100. Appendix B indicates the contents of the NRC Regulatory Guides. Regulatory Guide 4.1 (Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants) is reproduced in its entirety.

The EPA is specifically charged with controlling the impact of emissions, including radioactive, in order to limit their impact on the environment, with a view to protecting humans from undue harm. In connection with this responsibility, the EPA monitors concentrations and levels of pollutants in the environment. However, the primary monitoring responsibilities reside with other agencies (including state and local) and with the operators of facilities. The EPA has published a guide for monitoring environmental radiation, parts of which are reproduced in Appendix C. In addition, the EPA has promulgated, in 40 CFR 190, limitations on emissions and exposures from uranium fuel cycle activities (see discussion below and in Section I.D). This regulation, reproduced in Appendix D, places limitations on the overall fuel cycle, rather than on any particular facility (which would be the responsibility of the NRC). It would, however, require more stringent control of gaseous emissions from reprocessing plants than has been anticipated in the past (see reprocessing/waste management, Section II.C).

The character of a nuclear power plant imposes rather specific requirements on moni-

toring equipment, as regards both radionuclides of interest and monitoring conditions. This survey considers both monitoring at the source and environmental radiation monitoring.

Source monitors, in this context, are distributed within the reactor site and include: 1) devices to monitor the ambient radiation fields to which employees are exposed; and 2) fixed release monitors, designed for monitoring gaseous, liquid, or particulate radioactivity released via gaseous stack effluents or liquid discharges during normal operation, refueling, or accidents.

Environmental radiation monitors are designed to measure radioactive concentrations or radiation fields in the general environment, particularly along pathways to human exposure. These may include fencepost monitors surrounding a reactor site, instruments for sampling water or soil, and others.

(A third class of instruments, not fully covered in this survey, are "process" monitors, which keep track of the flow of materials within a facility. To the extent that these instruments may be used for site surveys and environmental monitoring, they are included in the notes and discussion.)

Although the radioactive content and emissions of a nuclear power plant may be characterized on a generic basis, the specific instrumentation required for monitoring around a particular site is usually selected after careful consideration of the features of that site. The environmental monitoring program is designed to ascertain the amount of radioactivity released from the site. The release modes and, certainly, the ultimate hazards associated with any releases depend strongly on the site characteristics. The on-site monitoring program, on the other hand, may not vary as substantially from one plant to another. This program may include stack and liquid effluent monitors, personnel dosimeters, area monitors, and special exercises (such as radiation emergency drills).

It is also important that natural background radiation levels be evaluated, particularly as a baseline for monitoring man-made environmental radiation and radioactivity. The total average natural dose to individuals in the United States is about 100 mrem/year (see Section II.F), most of which is contributed by cosmic ray and other external sources (including a small contribution from radioactive fallout from weapons testing). This corresponds to an irreducible background dose of about 10 μ rem/hour, which can cause substantial difficulty in making dose measurements below a few μ rem/hour. A significant problem, in fact, is that the natural levels may have diurnal and seasonal changes, masking the presence of small man-made contributions to the

total radiation environment. Similar difficulties may arise in determining radionuclide concentrations (rather than the overall radiation field). Measurements of man-made radionuclide concentration in environmental samples, such as air, water, or soil, can be significantly affected by natural background activities from such radionuclides as potassium-40 and the heavy natural alpha-emitting chains. As discussed in Section I.D., the regulations applicable to nuclear power operations only limit the increase in radiation levels that is caused by human activities.

The discussion that follows characterizes reactors as sources of radiation, discusses routine and accidental exposure modes, and notes requirements on monitoring instrumentation. Specific instruments are described elsewhere in this volume. A brief discussion of the nuclear fuel cycle serves only as an introduction to separate sections on fuel processing and uranium extraction.

2. Nuclear Power Plants as Sources of Radiation

Commercial nuclear power plants in the United States utilize a specific class of nuclear reactors, i.e., those cooled and moderated with ordinary water. Not surprisingly, the regulatory apparatus and associated monitoring requirements are directed primarily to light-water reactor (LWR) power plants and their associated fuel cycles. There are, however, numerous other reactor types, some of which may assume importance by the end of this century. Fortunately from the point of view of environmental monitoring, the radioactive emissions associated with nuclear power do not often depend drastically on the type of reactor used, except for specific differences in any recycled materials (see Section II.C).

Figure 2 provides a simplified diagram of the essential elements of nuclear-powered electrical generating plants which have been commercially available in the United States. The portion on the right shows the elements common to all of these plants. Steam produced in the reactor or in a heat exchanger associated with the reactor drives a turbine-generator to produce electricity. The turbine exhaust is condensed and the water pumped back to the reactor system, where it is vaporized once again. Typically about one third of this energy generated by the heat source is converted to electricity and two thirds is rejected to the environment as waste heat. This heat is dissipated in the atmosphere (via cooling towers) or in a body of water, or both.

It is a common feature of nuclear power plants that the primary coolant is recirculated to the reactor. Because of this, radioactive contaminants escaping from the reactor fuel into the coolant can be effectively confined.

Fig. 2. The three drawings on the left represent three important types of power reactors built in the United States: (a) the PWR (Pressurized Water Reactor); (b) the BWR (Boiling Water Reactor); and (c) the HTGR (High Temperature Gas-Cooled Reactor). They all boil water to generate steam. Following the generation of steam, the nuclear power plants are essentially the same, as indicated by the turbogenerator, condenser, and cooling tower on the right. Holding ponds are used sometimes to cool the water before releasing it to the environment. An important point: the cooling water from the environment does not enter the reactor. (From Seaborg and Corliss, Ref. 4, with permission by E. P. Hutton © 1971.)

As a result, the routine emissions from a power plant are predominantly gaseous species; even these, however, can be rather effectively contained. Nevertheless, there are opportunities for radioactive releases and, independently of this, workers at a nuclear power plant can receive exposures from radioactive materials which are within the plant.

a. Nuclear Power Plant Types. The pressurized water reactor (PWR) and the boiling water reactor (BWR), represented in Fig. 2 contribute the bulk of nuclear power generation in the United States and a substantial portion of that abroad. The high-temperature gas-cooled reactor (HTGR), also shown, has been withdrawn from the commercial market after construction of only one demonstration plant, but interest in this reactor type continues. A heavy-water-moderated reactor (CANDU) has been successfully marketed in Canada and elsewhere, with moderator and coolant of "heavy" water (wherein the hydrogen is the mass 2 isotope, deuterium) and a pressurized coolant system, as in a PWR.

Figure 2a shows a PWR. The energy generated in the core as a result of fission heats the water passing through the reactor vessel. The coolant is maintained at sufficiently high

pressure that gross boiling does not occur in the primary loop. Instead, the coolant is pumped through a heat exchanger or "steam generator" in which heat is transferred to a secondary fluid, also water, which is permitted to boil. The resulting steam drives the turbogenerator.

In a BWR, represented in Fig. 2b, the coolant is recirculated within the reactor vessel and is permitted to boil. The steam that is formed within the vessel is used directly to drive the turbogenerator. Because of the differences in the cooling systems, the radioactive effluents associated with reactor operation differ between a PWR and a BWR.

Both PWRs and BWRs initially have cores consisting of uranium oxide pellets encased in a metal cladding made of zirconium alloy. This is characteristic of most commercial power plants, including the Canadian CANDU reactor. However, LWR fuel consists of "enriched" uranium, wherein the percentage concentration of ^{235}U has been elevated to 2 to 4%, slightly above the natural 0.7%. The CANDU, on the other hand, uses natural uranium, a choice which is made possible by the use of heavy water, which absorbs fewer neutrons than

ordinary water, and by on-line, rather than batch, refueling.

Gas-cooled reactors, such as the HTGR, can use a different fuel type and moderator. The HTGR uses small fuel pellets of thorium and highly enriched uranium embedded in large blocks of carbon moderator. (Neutron capture by ^{232}Th leads to formation of ^{233}U , a fuel that is even superior to ^{235}U .) Functionally, though, the HTGR is similar to a PWR (see Fig. 2c).

All of the above are thermal reactors, in which neutrons are moderated to low ("thermal") energies in order to increase the probability of nuclear fission. On the other hand, fast breeder reactors, in which the neutrons are not moderated, have undergone substantial development in the United States and abroad. As ordinarily envisioned, these reactors would use ^{239}Pu as the principal material to be fissioned, releasing enough neutrons that, not only is one available to split another nucleus (thereby continuing a chain reaction), but enough neutrons are captured by ^{238}U , leading to ^{239}Pu , that more ^{239}Pu is produced than is consumed. As a result, there would be little demand on uranium resources, even if the nuclear electric generating system were growing relatively rapidly. The "breeder" that is most highly developed is the liquid metal fast breeder reactor (LMFBR).

Other types of reactor/fuel systems can be built. In many cases these rely on the thorium- ^{233}U fuel cycle, indicated above for the HTGR, in order to achieve effective uranium utilization in a thermal reactor. Depending on the design of the reactor and fuel cycle, as much ^{233}U may be produced as is destroyed, so that little demand would be made on uranium resources, except to build up an inventory of ^{233}U . See Section II.C (Nuclear Fuel Processing) for further discussion.

See Ref. 5 for a more detailed discussion of nuclear reactor types and principal features. Reference 6 treats associated questions of fuel reprocessing and waste disposal for conventional, as well as alternative, fuel cycles.

b. Sources and Types of Radiation.

Nuclear power plants, as well as other parts of the fuel cycle, utilize or produce materials with a range of chemical, physical, and radioactive characteristics. Together these properties strongly affect both pathways to humans and the resulting radiation doses. As discussed in the introduction (Section I.F.7), the chemical and physical characteristics determine the form and flow of materials in the environment and in the human body; the radioactive decay modes determine the type of exposures that may ensue. In some circumstances, the materials from nuclear power may expose humans externally, particularly if their de-

cay yields x-rays or gamma rays. The effects of less penetrating forms of radiation may be mitigated by small amounts of shielding; in many cases, even the air or the external layer of skin may suffice. However, many such radionuclides may be taken up by the body, so that the decay radiation may expose the body internally. The primary hazard of most alphas and many beta emitters, as well as many of the x-ray and gamma emitters, occurs because of this possibility (Ref. 7).

Not all of the radiation doses associated with operations arise from radioactivity generated in reactors. Nor are reactors the only points in the nuclear fuel cycle where releases of reactor-produced material may occur. The next section summarizes other types of releases and indicates where some of these are discussed in this volume.

The amounts of radioactivity released from nuclear power plants and/or the radiation doses to workers or members of the general population depend both on the reactor type and on the manner in which it is constructed and operated. In spite of this variation, some general statements can be made. This arises both because the fundamental nuclear processes occurring inside nuclear reactors do not vary greatly from one reactor to another and because regulatory practice has some degree of uniformity.

Light water power reactors are by far the most prevalent type today. Table 1 (from Ref. 8) gives estimates of the radioactive inventory (based on discharge fuel composition) of an LWR that has reached equilibrium; it also indicates some important physical and health considerations for the various elemental classes. In addition, the table specifies how the inventory changes for two time periods (see Nuclear Fuel Processing Section.)

The total radioactive inventory of a large operating reactor amounts to billions of curies. However, only an extremely small portion of this can escape into the environment, even under very unusual circumstances. Reactors are ordinarily designed so that several barriers serve to confine this large amount of radioactivity. Moreover, the internal shielding of a power plant is designed to limit exposures of plant workers to acceptable levels.

Several "intrinsic" barriers limit the amount of radioactivity that escapes. The first is the fuel material itself, wherein most of a reactor's radioactive inventory is generated. For conventional reactors, this material is a solid ceramic, uranium dioxide (or, in reactors that use plutonium, plutonium dioxide), formed into pellets. Most of the products of the chain reaction taking place in this material remain in this solid

Table 1. Radioactivity in reactor and fuel cycle for a 1000-MWe uranium-fueled Light-Water Reactor
(From Nero, Ref. 5, with permission by UC Press © 1979.)

Radionuclides	Half-Life	Reactor Inventory ^b (10 ⁶ Curies)	In Discharge Fuel ^c (10 ⁶ Curies/yr)			Elemental Boiling Temperature (°F)	Health Considerations and Principal Radiation
			At Discharge	150-Day Decay	10-Yr Decay		
Fission Products							
Tritium ^d	³ H	12.26 yr	0.0723	0.0241	0.0239	0.0139	212°F (as HTO) Internal hazard, β (0.019 MeV max)
Krypton	83m	1.86 hr	5.71	1.90	0	0	-243°F (Gaseous) External irradiation, β (0.67 MeV max for ⁸⁵ Kr)
	85m	4.4 hr	17.2	5.70	0	0	
	85	10.76 yr	1.16	0.383	0.373	0.201	
	87	76 min	34.0	11.3	0	0	
	88	2.8 hr	49.0	16.5	0	0	
	89	3.18 min	61.8	20.3	0	0	
90	33 sec	58.5	22.7	0	0		
Total ^e		325	107.7	0.373	0.201		
Strontium	89	52.7 days	71.6	23.8	3.22	0	2,490°F Nonvolatile Internal hazard to bone and lung, β (0.55 MeV max for ⁹⁰ Sr)
	90	27.7 yr	7.80	2.58	2.56	2.02	
Total ^e		526	174	5.78	2.02		
Iodine	129	1.7×10 ⁷ yr	3.03×10 ⁻⁶	1.00×10 ⁻⁶	1.02×10 ⁻⁶	1.03×10 ⁻⁶	333°F (Highly volatile) Internal hazard to thyroid, β (0.8 MeV max for ¹³¹ I) and γ
	131	8.05 days	71.9	23.9	6.01×10 ⁻⁵	0	
	132	2.26 hr	103	34.2	0	0	
	133	20.3 hr	137	45.6	0	0	
	134	52.2 min	156	51.8	0	0	
	135	6.68 hr	123	40.7	0	0	
	136	83 sec	54.0	17.9	0	0	
Total ^e		1,017	337	6.11×10 ⁻⁵	1.03×10 ⁻⁶		
Xenon	131m	11.8 days	0.582	0.193	9.05×10 ⁻⁵	0	-162°F (Gaseous) External irradiation, β (0.35 MeV max for ¹³³ Xe) and γ
	133m	2.26 days	3.29	1.09	0	0	
	133	5.27 days	137	45.6	0	0	
	135m	15.6 min	36.8	12.2	0	0	
	135	9.14 hr	25.7	8.52	0	0	
	137	3.9 min	132	43.8	0	0	
	138	17.5 min	128	42.5	0	0	
	139	43 sec	107	35.6	0	0	
Total ^e		680	225	9.05×10 ⁻⁵	0		
Cesium	134	2.046 yr	19.0	6.32	5.49	0.215	1238°F (Volatile) Internal hazard to muscle, β (1.176 MeV max for ¹³⁷ Cs) and γ
	137	30.0 yr	9.92	3.29	3.26	2.61	
Total ^e		595	198	8.75	2.83		
Zr, Nb, Mo, Tc, Ru, Rh, Pd, Ag, Cd, In, Sn, Sb			1,880	625	54.1	0.0450	Nonvolatile External and internal hazard
Rare Earths: La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Dy, Ho			4,140	1,374	48.8	0.416	Nonvolatile External and internal hazard
Total fission products			11,970	3970 (1.067 Te/yr)	130	9.98	
Actinides							
Uranium	237	6.75 day	108	35.9	7.42×10 ⁻⁶	0	Nonvolatile External and internal hazard, β (1.29 MeV max for ²³⁷ U) and γ
	239	23.5 min	1,708	566	0	0	
Total ^e		1,816 (99.7 Te)	601 (33.0 Te/yr)	2.72×10 ⁻⁵	2.15×10 ⁻⁵		
Plutonium	238	86.4 yr	0.138	0.0459	0.0483	0.0463	Nonvolatile Internal hazard to bone, liver, lung, lymph, α (about 5 MeV for ^{238,239,240,242} Pu) and β (for ²⁴¹ Pu)
	239	24,390 yr	0.0318	0.0105	0.0107	0.0107	
	240	6,580 yr	0.0500	0.0166	0.0166	0.0166	
	241	13.2 yr	12.4	4.11	4.04	2.42	
	242	3.79×10 ⁵ yr	1.24×10 ⁻⁴	4.14×10 ⁻⁵	4.14×10 ⁻⁵	4.14×10 ⁻⁵	
	243	4.98 yr	22.2	7.35	0	0	
Total ^e		34.9 (0.893 Te)	11.57 (0.296 Te/yr)	4.11	2.49		
Fissionable Pu (²³⁹ Pu + ²⁴¹ Pu)			(0.627 Te)	(0.207 Te/yr)			
Americium (241, 242m, 242, 243) and Curium (242, 243, 244)			1.14	3.79	0.340	0.0611	Nonvolatile Internal hazard, α (5.5 MeV for ²⁴¹ Am)
Total Actinides Th, Pa, U, Np, Pu, Am, Cm			3,614	1,198	4.45	2.55

Table 1. (Continued)

Radionuclides	Half-Life	Reactor Inventory ^b (10 ⁶ Curies)	In Discharge Fuel ^c (10 ⁶ Curies/yr)			Elemental Boiling Temperature (° F)	Health Considerations and Principal Radiation
			At Discharge	150-Day Decay	10-Yr Decay		
Cladding							
Activated Zircaloy cladding and Inconel spacers (Cr, Mn, Fe, Co, Ni, Zr, Nb, Sb)		12.9 (28.2 Te)	4.28 (9.36 Te/yr)	0.967	0.109	Nonvolatile	
TOTAL FISSION PROD- UCTS, ACTINIDES, AND CLADDING ^f		15,600 (132 Te)	5,170 (43.9 Te/yr)	135	12.6		

a. Overall thermal efficiency = 32%.

b. Calculated on the basis that all fuel has the composition of discharge fuel. For nonsaturable species, whose half-lives are large compared with the typical fuel life of 3 years, the inventory within the reactor will be about one-half to two-thirds of the value listed, depending upon the fuel management scheme. The inventories listed in the table are accurate for saturable (short-half-life) species. Material quantities in metric tonnes (Te) are listed in parentheses.

c. Fuel cycle quantities are calculated on the basis of a fuel exposure of 33,000 MWd/Te 100% load factor. Material quantities are listed in parentheses.

d. Quantities listed are for fission-product tritium only and do not include tritium formed by neutron-activation reactions in boron controls and contaminants.

e. Totals include all isotopes of the element, including short-lived species not listed in the table.

f. Does not include ¹⁴C, which is produced in the fuel and cladding (the order of 50 Ci/yr), as well as the coolant (roughly 10 Ci/yr).

Source: Adapted from T. H. Pigford et al., "Fuel Cycles for Electric Power Generation," Teknekron report EEED 101 (January 1973, rev. March 1975).

matrix. This includes both the fission products and actinides (see Table 1). However, some portion of this radioactivity can migrate out of the fuel pellets. This is particularly true of the gaseous fission products, including tritium, krypton, xenon, and iodine. However, the fuel pellets are contained in long, thin metal tubes (cladding), which provide both structure to the nuclear core and serve as a second barrier. (Bundles or "assemblies" of these rods are the basic fuel handling unit.) Other possible sources of fission products - apparently minor - are fuel that contaminates the surface of new fuel elements and fuel that passes into reactor coolant water from failed fuel elements. Some radioactivity is also generated in the cladding itself, primarily from neutron activation (see Table 1), but this remains in the cladding except for minor corrosion.

Some small percentage of the time, the cladding develops leaks, permitting the more volatile fission products to escape from the fuel rods into the surrounding coolant. In addition, other radionuclides, including tritium, nitrogen 16 and 13, and carbon 14, are generated as activation products in the coolant itself. These activation products in reactor coolant water are formed by neutron irradiation of the water and its contents (including gases and dissolved or suspended solids) and materials in contact with the coolant (container and structural surfaces, fuel and control rod cladding) that subsequently corrode or erode. Additional radionuclides may be formed in other parts of the reactor than the core. However, the fact that the coolant is confined to a closed recirculating system serves largely to confine to the system both these radionuclides and leaked fission products. Some of

these are so short lived that relatively brief retention in the primary coolant system permits enough decay to effectively reduce their concentration.

However, not all of the remaining radioactivity is permitted to accumulate in the coolant. Every power plant has a system that controls the chemistry of this coolant by routinely bleeding off and treating a portion of the coolant inventory. Some of the radioactivity is collected in residue from this chemistry control system, later to be packaged for disposal. Some is discharged to the environment, often after storage in holdup tanks, permitting further decay. And a portion may be returned to the coolant system. As discussed in the next section, routine releases to the environment are strongly affected by the form of various radioactive control systems.

3. Routine and Accidental Releases and Exposures from the Nuclear Fuel Cycle

The main focus of this section is the nuclear power plant itself. However, although such a power plant is the central object of a commercial nuclear power system, the facilities that support this plant by supplying fresh fuel and disposing of spent fuel are also of interest from an environmental point of view and are discussed briefly at the end of this section and more thoroughly elsewhere. Figure 3 presents a nominal material and environment release flow sheet (from Ref. 8) for an LWR fuel cycle that reprocesses spent fuel and stores plutonium. Reference 2 estimates radioactive releases and exposures that may be associated with various portions of LWR fuel cycles, with and without reprocessing or plutonium recycling.

Fig. 3. Material and environmental release flow sheet for a typical 1000 MWe Light Water Reactor nuclear power plant (Ref. 8).

a. Routine Power Plant Emissions. Turning first to routine emissions from reactor power plants, several radionuclides - alluded to in the last section - are of primary interest: tritium, krypton, xenon, iodine, and carbon. Except for carbon and tritium, these releases arise almost solely from radionuclides produced in the fuel. A substantial portion of the ^{14}C may be produced in the coolant. The major sources of released tritium are 1) ternary fission in the fuel, 2) activation of deuterium in the water, and 3) neutron-boron reactions in the boron control curtain (see table on page 16 of Ref. 10). These nuclides are able to escape from the power plant largely because they are gaseous or volatile in elemental form or enter into compounds which are. However, the amounts which are released depend strongly on the form of the liquid and gaseous radioactivity control systems. Questions concerning the fraction of a ^{14}C release to be in CO_2 form as opposed to other chemical species (e.g., CH_4) remain to be answered. (See Fig. 4 for examples of these systems in a recently completed nuclear power plant). Table 2 presents some estimates of annual releases from nuclear power plants.

For a 1 gigawatt (1000 MWe) power plant, some tens to hundreds of curies of tritium in compound as water, are released annually to air in the form of water vapor or to the body of water being used for cooling. Most of the tritium produced remains confined in the fuel and would only be released were the fuel to be reprocessed. (See Section IV.A and II.C). Tritium, with a half life of 12.3 years, decays by emitting a very soft beta ray ($E_{\text{max}} = 18.6 \text{ keV}$), so that, for small releases, its principal hazard arises from internal doses.

Tens of thousands of curies of the noble gases krypton and xenon are routinely released per gigawatt-year of operation. The releases would be substantially higher, but for holdup of these gases to permit the short-lived radioactivity to decay. The most important releases are ^{85}Kr , which has an 11 year half-life and a 670 keV (max) beta ray, and ^{133}Xe , with a 5 day half-life and a 546 keV (max) beta ray. Much greater amounts of these materials are left in the fuel. Of the noble gas radionuclides, it is ^{85}Kr that, like tritium, can cause significant population exposures if released at fuel reprocessing plants (see Section II.D).

Releases of iodine are very small, less than one curie per gigawatt-year, but are controlled particularly carefully because deposited iodine can be concentrated by an important food chain (grass-cow-milk-human) and - furthermore - it tends to collect in the thyroid. See Section IV.D. For short-term considerations, ^{131}I (8 days, 364 keV gamma)

has the greatest significance, but in terms of longer commitment, ^{129}I (17 million years, 150 keV max beta) can assume greater importance.

From the point of view of dose commitment, the several curies of ^{14}C released yearly from a large power plant can contribute the largest population dose over the medium term (hundreds of years). This radionuclide (5730 year half-life, 155 keV max. beta) escapes in the form of CO_2 or other compounds and - over centuries - can contribute hundreds of person-rem per gigawatt-year. It is difficult to decide over what period to estimate such dose commitments or to understand the significance, compared with other risks, of such long-term population exposures. The doses accumulated during the operation of the plant are numerically much smaller, a few person-rem per gigawatt year. These population doses arise primarily from tritium and noble gas releases.

The doses to individuals in the general population are ordinarily much smaller than the generally applicable limit of 500 mrem/year. The numerical guidelines used in examining the adequacy of proposed radioactivity control systems for nuclear power plants specify doses in the vicinity of 10 mrem/year as design goals (see Table 3). These guidelines were chosen by comparing the availability and cost of control systems with the benefits of reduced radiation exposures (Ref. 14). Similarly, EPA regulations on overall fuel cycle emissions (Ref. 1) were set on the basis of similar considerations (see also Section II.C). These annual doses are much less, by at least an order of magnitude, than the generally applicable standards (500 mrem/yr for an individual member of the general public) and substantially less than normal background dose levels (roughly 100 mrem/year). Measuring reactor-caused increases against a background of this size can be a challenge for a monitoring system, particularly when the natural background varies as a function of both time and place. Several reports^{15,16} are available that summarize radioactive wastes released from, or shipped offsite by, nuclear power plants in the United States; these summarize measurements of specific radionuclides both in gaseous and liquid effluents and in solid waste shipments.

b. Accidental Releases from Nuclear Power Plants. The ceramic fuel pellets, the sealed cladding, and the closed reactor coolant system typical of LWRs and most other reactor types serve as barriers to releases of radioactivity under both routine and accident conditions. However, accidents might occur that would breach one or more of these intrinsic barriers, possibly leading to increased, even massive, releases of radioactivity. To cope with the spectrum of failures that might occur, these intrinsic features are supplemented by several "engineered" safety systems. These

a.

b.

Fig. 4. Radioactivity control systems.
a. Gaseous. b. Liquid.
(From Ref. 11.)

Table 2. Projected Yearly Routine Gaseous Emissions from LWR Power Plants
(Ci/reactor-year, uranium fuel cycle).

Radionuclides	PWR (BWR) ^b	Rancho ^c Seco (PWR)	Diablo ^d Canyon (PWR)	Kosh. ^e (PWR)	GESMO ^f (PWR)	Processing ^g
³ H ^a	10 to 50(same)	900 (not given)		580	1100	2.1×10 ⁴
¹³¹ I	0.016(0.016)	0.011 (0.28)		0.009	0.025	0.06(¹²⁹ ¹³¹ I)
Krypton, Xenon	7000(50,000)	12,000 (3700)		330	13,000	1.8×10 ⁵ (⁸⁵ Kr)
¹⁴ C	Not given	Not given		9	8	15
(transuranics)						(0.004)

^aIt should be noted that a trade-off can occur between tritium discharges into air versus water, so that these numbers can be highly variable, even aside from normal considerations of control technology.

^bfrom Pigford et al., reference 8.

^cfrom the 1973 Final Environmental Statement for the 900 MWe PWR unit at Rancho Seco, reference 11.

^dfrom the 1973 Final Environmental Statement for each 1060 MWe PWR unit at Diablo Canyon, reference 12.

^efrom the 1976 draft environmental statement for each 994 MWe PWR unit at Koshkonong, reference 13.

^fPWR releases assumed by GESMO² for radwaste systems of the current type; re-leases calculated in accordance with Regulatory Guide 1.112.

^gApproximate numbers for releases in Curie per gigawatt-year from reprocessing plants; taken from reference 8, except for ¹⁴C, which is taken from reference 2. For radionuclides other than ¹⁴C, the two references broadly agree, except that the numbers are more difficult to extract from reference 2.

systems do not prevent failures, but are designed to limit the course of the accident, either by preventing substantial release of radioactivity bound in the fuel or by containing radioactivity that succeeds in escaping from the primary coolant system.

Under ordinary conditions, almost all of the plant radioactivity is contained in the coolant system or in auxiliary systems, such as the radioactive effluent control systems. Releases from these auxiliary systems or even from the primary system may be regarded as relatively minor, provided the fuel does not overheat to such an extent that contained volatile fission products are released from the cladding or, worse yet, the fuel melts.

Reducing the possibility of such damage to the core is a major goal of engineered

safety systems. Such damage, however, can occur - once a reactor has been operating at power for a time - merely by losing the coolant from the reactor vessel. Although for an LWR such loss of coolant terminates the chain reaction (due to lack of neutron moderation), large amounts of thermal energy can remain in the fuel, and fission products continue to decay, releasing energy at a substantial rate: immediately after shutdown, the decay power is about 7% of the full power level. This reduced power level is sufficient to melt the fuel cladding and/or fuel very rapidly if cooling is not available. To cope with the possibility that coolant may be lost, power reactors are equipped with emergency core cooling systems designed to replenish the coolant inventory under a wide range of conditions. The primary and emergency cooling systems are designed with a high degree of

Table 3. Nuclear Power Plant Off-Site Dose Limiting Objectives (10 CFR 50, Appendix I)

Type of Dose	Design Objective (per unit)
<u>Liquid effluents</u>	
dose to total body from all pathways	3 mrem/yr
dose to any organ from all pathways	10 mrem/yr
<u>Gaseous effluents (only for noble gases)</u>	
gamma dose in air	10 mrad/yr
beta dose in air	20 mrad/yr
dose to total body	5 mrem/yr
dose to skin	15 mrem/yr
<u>Radioiodines and particulates released to the atmosphere</u>	
dose to any organ from all pathways	15 mrem/yr

redundancy and independence to prevent major damage to the core.

In the United States, the primary system is surrounded by a containment building designed to withstand overpressures that might be expected on breach of the primary system. This containment serves to confine radioactivity releases, whether large or small, from the reactor system. The containment includes cooling systems to reduce the internal pressure, filter systems to collect radioactivity, and spray systems, which may do both.

Considering the variety of failures that may occur in the basic reactor systems and the degrees of success that are possible in effectiveness of safety systems, a wide range of radioactive releases might result from possible accidents. These releases would vary as to size, radionuclide mixtures, and dispersion conditions (such as rate of release, amount of heat released, meteorological conditions). Finally, the population at risk varies markedly from one power plant site to another. Large uncertainties and substantial controversy over the probability and consequences of reactor accidents, arising directly from imprecise information on failure rates, effectiveness of safety systems, the details of accident sequences and subsequent exposures. Reference 17 examines the state of LWR safety. In 1975, a major AEC/NRC study^{18*} of the risk from nuclear power accidents was completed. Many of

its results are uncertain and controversial, as indicated in a review^{19*} of the study, but it is useful to note its estimate that the average population dose from accidents from power plant operation is in the range of hundreds of person-rem per gigawatt-year, much greater than the yearly population doses in the vicinity of a nuclear power plant and comparable to the dose commitment estimated from ¹⁴C. Reference 19 reviews studies of reactor safety.

Extremely high doses could result from nuclear accidents, although no major accidents have so far occurred. These doses would arise from substantial releases of a wider range of radionuclides than are given off under normal conditions. In particular, radionuclides such as ¹³⁷Cs and ⁹⁰Sr that ordinarily remain in the fuel may be released. Moreover, much greater amounts of iodine and other volatile radionuclides can escape than normal. However, in addition to causing high doses to individuals, an accident can release radioactivity that persists, subjecting large numbers of people to chronic low levels of radiation over a large area. Either of these cases presents a serious monitoring problem, the first because ordinary environmental monitoring instruments are not designed to measure high radiation levels; at the other extreme, low levels of radiation may be difficult to measure, merely because of the presence of ordinary background radiation.

*Sometimes the reference numbers will appear as superscripts.

The present regulatory approach to nuclear accidents concentrates on assuring that doses to members of the general public do not exceed certain limits. The nominal accident dose of interest in examining a proposed site during licensing proceedings is 25 rem. (See 10 CFR 100.11.) On the other hand, EPA protective action guides (Ref. 20), specify 1 to 5 rem as the dose at which emergency actions should begin.

c. Fuel Handling and Other Occupational Exposures. The operation of a nuclear power plant raises the ambient radiation levels nearby only slightly but this occurs to a much larger degree inside the plant. As a result, plant workers receive radiation doses that are larger than those imposed on the general public. However, the bulk of these increased exposures arise, not when the reactor is operating, but during refueling and maintenance.

Radioactivity collects at several points in the plant systems. Most resides in the fuel and, not surprisingly, the bulk of worker doses is associated with x-ray and gamma doses from handling fuel during refueling operations. This is true even though precautions are taken that control these doses, such as handling the fuel under water. In addition, radioactivity can collect in the primary system and certainly in the radioactive effluent control systems. Maintenance associated with these systems can lead to substantial x-ray and gamma exposures. Plant operators take care that occupational exposure limits (such as the 5 rem/yr whole body dose) are not exceeded, but often workers are permitted to receive doses close to this amount.

d. Transportation and the Balance of the Fuel Cycle. Of the few person-rem per gigawatt-year general public exposures that are directly associated with power plant operations, about half is estimated to arise from transport of the spent fuel (Refs. 11-13). This fuel is transferred in large shipping casks, designed to provide physical protection, cooling, and radioactive shielding. Nevertheless care must be taken that this shielding is adequate and particularly that it remain so after even serious accidents.

In the full nuclear fuel cycle, there are of course other possibilities for radiation exposures than the ones we have so far discussed. Release points were indicated in summary form in Fig. 3. Other parts of this volume discuss some of the principal fuel cycle operations: uranium mining and milling; fuel reprocessing and waste disposal. Although their average impact, too, is uncertain, each of these could cause population exposures comparable to those discussed above for nuclear power plants.

4. Measurement Considerations

To assure that nuclear power plants meet applicable regulations and specifications for emissions and exposures, the Nuclear Regulatory Commission has formulated regulatory guides that specify monitoring practice within the plant and in its environs. Appendix B indicates the regulatory guides relevant to monitoring both inside and outside the plant. Of particular interest are: RG 1.21 ("Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants", Rev. 1, 6/74); RG 4.1 ("Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants", Rev. 1, 4/75), which is reproduced in its entirety; and various guides from Division 8 ("Occupational Health"). These guides, and the requirements that they are intended to fulfill, set forth programs for dose monitoring within the plant, effluent release monitoring, and environmental monitoring (with particular attention to critical pathways leading to human exposures); these programs must serve in a variety of situations, including normal operation, operational transients, and severe accidents. Both the level and type of radiation to be measured can depend on the situation at hand. For example, most of the radionuclides released during normal operation emit β and γ radiation; during accidents, significant amounts of α emitters could also be released.

For the purposes of this discussion, measurements are regarded to be applied within the plant itself ("plant" monitoring) or outside the plant ("environment"). "Plant" monitoring is taken to include programs for measuring radiation fields within the plant, as well as those for measuring releases at the stack or through other effluent streams. However, it does not include "process radiation monitoring", even though process monitors may provide a reference for correlating possible sources of environmental releases; the purpose of and operating conditions for such sensors (especially those operating near the core) may result in substantially different designs than those for use in making low-level measurements.

This section briefly discusses the two applications of interest, "plant" and "environmental" monitoring, then notes some of the considerations in formulating a monitoring program, a subject that is also treated in Division I. More detailed information on specific instrumentation is contained on Division III and IV of this volume.

a. Plant Monitoring. One general requirement is that sufficient capabilities be present to monitor radiation fields within

the plant. Instrumentation of various types is available for such measurement. Measurements of gross beta and gamma activity provide an overall reference, although more specialized instruments may be employed to acquire more detailed information. In addition, (thermal) neutron detectors are placed to indicate large fluxes. Specific attention to the various components of radiation due to individual sources is not always necessary, although it is certainly useful information when attempting to diagnose a leak and document its history.

Even within the plant, instruments may be fixed in position, used for special surveys, or designed for personnel dosimetry. Wall-mounted β - γ /neutron counters require sufficient sensitivity (down to about 1 μ R/hr) to indicate ordinary daily variations. A distinct set of instruments is required for measuring accidental releases; ranges up to about 10,000 R/hr are appropriate. For either type, fail-safe audible and visual alarms, as well as provision for recording measurements, are ordinarily available.

Portable instruments operating on similar principles as fixed units may also be used in surveys. On the other hand, surveys may be conducted by securing samples that are then analyzed in fixed instruments; as an example, surface radioactive contamination may be monitored by taking surface wipes for analysis by such instruments.

Dosimeters for measuring employee exposures may be both pocket ion chambers, worn outside of heavy clothing, and integrating dosimeters (of film or thermoluminescent type), changed periodically. These provide information, respectively, on short-term doses and on long-term (integral) doses received by employees.

A second requirement of plant monitoring systems is that they provide adequate information on radioactivity leaving the plant, whether as packaged waste material, or as liquid or gaseous effluents. Stack effluent monitors provide a temporal record of emissions, in addition to integrating the total amount released. They can also serve as a warning of abnormalities. However, for "operational occurrences" and "accidents", instruments must be designed for unusually high radiation levels (Ref.21).

A principal requirement for the instruments discussed above, as well as those indicated under environmental monitoring, is that periodic calibration and maintenance procedures be performed. For example, remotely controlled or portable radioactive sources are used routinely to check the calibration and operation of permanently installed monitoring instrumentation. In addition, a

calibrated electronic test pulse may be used for rapid identification of instrument failures. A part of this inspection and testing program is also normally devoted to assuring that sampling, such as in the stack,^{22*} is occurring properly.

As suggested by such considerations, monitoring instruments are not to be viewed in isolation, but rather as part of a monitoring system that is designed with certain objectives and must operate within certain specifications, all of which are reviewed during the licensing process. Furthermore, with recent advances in data handling equipment, individual monitoring units may be incorporated into a system that not only assures proper functioning of the units, but reads data out of these instruments, performs necessary conversions (e.g., from count rate to dose rate), and organizes the resulting information for examination and, if necessary, response (see, for example, Ref. 23). Such systems would, of course, be designed to meet requirements specified in applicable regulations.

Information on satisfying NRC requirements for waste and effluent monitoring is given in Regulatory Guide 1.21. See also the American National Standards Institute recommendation on continuously monitoring effluents.^{24*} Requirements for in-plant radiation monitoring are also given in other regulatory guides, including Division 8 (Occupational Health). Appendix B contains lists of relevant regulatory guides. Detailed descriptions of instrumentation are contained in other portions of this volume.

Other important considerations for plant monitoring, including program design, quality assurance, and inspection are discussed briefly in "monitoring programs" below.

b. Environmental Monitoring. The instruments for monitoring in the environs of a power plant are based on the same principles as those indicated above. However, the requirements on sensitivity may be more stringent because of the relatively small increases in levels typically caused by nuclear power plant effluents. For the same reason, variations of background level with place and time, whether hourly, daily, or seasonally are an important monitoring consideration. As discussed in Section II.F, local background radiation exposures are typically about 100 mrem, with $\pm 50\%$ variation from place to place and with daily variations as much as $\pm 50\%$. It may therefore be difficult to measure the additional dose from a properly operating reactor, which is ordinarily expected to contribute less than 5 mrem/yr at the site fence. Another primary consideration for environmental monitors is that accuracy be maintained over a wide range of environmental conditions,

including large variations in temperature, humidity, pressure, wind speed, precipitation and dust.

Simple passive monitors, TLDs and film, serve as the backbone of environmental monitoring systems. They are easily used to provide a record of penetrating radiation at numerous points in the area to be monitored, and they are conveniently used for archival storage. However, more complex instruments or techniques may be required for specific monitoring purposes, including observation of particular radionuclides, monitoring of critical pathways, and emergency radiation monitoring. The choice of instruments depends on the purpose and the design of the monitoring program, but may include ion chambers and semiconductor detectors, both in survey and in stationary devices.

The NRC regulatory guide on environmental monitoring programs, RG 4.1, is given in Appendix B. However, other guides cited there are relevant. In addition, the EPA has published an Environmental Radioactivity Surveillance Guide (see Appendix D). References 25 and 26 are useful guides to surveillance around nuclear facilities. As illustrative of the types of measurements that may be made around nuclear facilities, Appendices E and F excerpt portions of reports on monitoring in the vicinity of a PWR^{27*} and a BWR,^{10*} respectively. For a discussion of environmental impact monitoring, see Ref. 28. A guide to general environmental monitoring has recently been published by the NCRP.^{29*} Questions of program design are the subject of the next section.

c. Monitoring Programs. In the discussion above, we have alluded to several generic questions that affect the design of a monitoring program. These include choice of instruments, sampling method and rate, data handling, and testing maintenance. These matters are discussed in moderate detail in the references cited above, but it is useful to summarize them briefly.

Instrument type and sampling method or rate depend strongly on the purpose of the monitoring program. Much of the discussion above emphasized in situ measurements, but grab sampling techniques, where a sample is taken at one location, then analyzed elsewhere, are important, particularly for sensitive monitoring techniques. The in situ techniques may employ fixed instruments of various sorts, in the plant, at emission points, or in the environment. Also in use are various types of survey instruments and dosimeters. All of these are discussed in detail elsewhere in this volume.

In many cases, such as at the stack or liquid effluent stream, fixed instruments monitor continuously. This provides both an

alarm capability and a record of emissions, as mentioned in the discussion of plant monitoring. Based on meteorological models and local meteorological data, emission data may be used to estimate doses resulting from plant releases.

On the other hand, a substantial part of the monitoring program - particularly in the environs of the plant - may depend on taking samples for subsequent analysis. Samples may be taken of environmental media and of plants and animals: air, water, fish, birds, corn, cows, milk, thyroid, etc. Depending on the particular monitoring program, the sampling frequency may vary (daily to even yearly), and the radionuclides chosen may vary (³H, ⁸⁵Kr, ⁹⁰Sr, ¹³¹I). The purpose of such a program would be both to check the plant emission data and - more importantly - to evaluate radionuclide transport in the environment, particularly along critical pathways. Both for these pathways and for the more general environmental monitoring program, the site, type, and rate of sampling should be carefully chosen to support the monitoring objectives. The techniques applicable in the vicinity of a nuclear facility will differ in some cases from those for monitoring far from such facilities, since the mix of radionuclides may vary with proximity and the radionuclides of interest depend on the monitoring objectives. Table 4 is a summary of the advantages and disadvantages of a variety of sampling methods for the principal types of environmental media (Ref. 25).

Although samples may be examined only for gross activity or for overall α , β , or γ activity, they may also be analyzed for radionuclide content. Techniques for specific radionuclide monitoring are discussed in Division IV. For gas and liquid phase measurements, as well as soil and biological samples, chemical separation techniques are employed, which may be followed by a variety of instrumental measurements. Airborne particulates are typically collected by air filters for subsequent analysis, perhaps using similar instruments. These instruments may include spectrum analyzers. In addition to Division III and IV, the reader is referred to Volumes 1 and 2 on Air and Water, which contain sections on neutron activation analysis, x-ray fluorescence, etc.

Designing a monitoring program also requires choosing an approach to data handling and use. As noted under plant monitoring, the instruments themselves may be incorporated into a monitoring system that collects the data, performs necessary conversions, and displays or otherwise transfers the results for designated purposes. In addition, the system may partially check instruments to ensure proper functioning, an important requirement, as discussed below. To some extent, systems for environmental monitoring may also be inte-

Table 4. Advantages and Disadvantages of Sampling Methods for Various Samples

Type of sample and/or method of collection	Advantages	Disadvantages
ATMOSPHERIC		
Particulates		
Constant air monitors	Real-time readout, chart record	High cost, special maintenance requirements, lack of sensitivity, interference (e.g., radon)
Pump and filter	Sensitivity, low cost	Time delay in obtaining results
Electrostatic precipitators	Sensitivity	High cost, variable efficiency
Halogens		
Charcoal	All forms collected	Not nuclide specific
Molecular sieves		High cost, low capacity, not nuclide specific
Bubblers	Sensitivity, low cost	Special maintenance requirements, not all forms collected
Noble gases		
Evacuated chamber	Portability	Lack of sensitivity, not continuous
Compressed air	Sensitivity	Not continuous
Cryogenic	Sensitivity	High cost, not portable
Direct radiation (thermoluminescent dosimeter)	Low cost, dose integration, direct dose measurement	Lack of sensitivity, high background, not nuclide specific
Ion chamber	Real-time readout, chart record, sensitivity, direct dose-rate measurement	High cost, special maintenance requirements
Tritium		
Cryogenic	No sample preparation	High cost, not portable, HT not collected, collection not ^3H -specific
Bubblers	Low cost, no sample preparation	Special maintenance requirements. Lack of sensitivity, HT not collected, collection not ^3H -specific
Molecular sieve	Portability, ^3H -specific (with distillation)	Sample preparation needed. HT not collected, capacity varies with relative humidity
Drierite	Portability, ^3H -specific (with distillation)	Poor water capacity, capacity varies with relative humidity
Silica gel	Portability, ^3H -specific (with distillation)	Sample preparation needed. HT not collected, capacity varies with relative humidity
TERRESTRIAL		
Milk	Direct dose vector, ease of data interpretation	Not always available
Foodstuffs	Direct dose vector, ease of data interpretation	Not always available, weathering effects

(continued)

Table 4. (Cont.)

Type of sample and/or method of collection	Advantages	Disadvantages
Wildlife	Direct dose vector	High mobility, not always available, data difficult to interpret
Vegetation	Availability, multiple accumulation modes	Not nuclide specific, data difficult to interpret, weathering effects
Direct instrument reading	Low cost, speed	Lack of sensitivity, not nuclide specific
Deposition collection	Defined time span for sampling period, area and collection factors known	Weathering effects
Soil sampling	Good time integrator	High cost, lack of representativeness, long time delay in obtaining results
AQUATIC		
Surface water	Availability	Not dose related
Groundwater	Indicator of waste management	Not always available, data difficult to interpret
Drinking water	Direct dose vector	Lack of sensitivity
Aquatic plants	Sensitivity	Data difficult to interpret
Sediment	Sensitivity, integration	Data difficult to interpret
Fish, shellfish	Direct dose vector, sensitivity	Frequent unavailability, high mobility
Waterfowl	Direct dose vector	Frequent unavailability, high mobility, data difficult to interpret

grated in a similar way. In reducing data, monitoring systems - whether for plant or environment - may even perform dispersion calculations (based on meteorological models and data supplied), as well as pathway analysis (based on appropriate pathway models).

Even if it is not appropriate to devise such a monitoring system, it is important that the overall program be organized to meet specified objectives. This must include assurance that the program operates properly. In Regulatory Guide 4.15 (see list, Appendix B), the NRC specifies "Quality Assurance for Radiological Monitoring Programs (Normal Operation) - Effluent Streams and the Environment". A quality assurance program should include:²⁶*acceptance testing of instrumentation; routine calibration, routine replicate sampling; laboratory cross-checking; procedural audits; and program documentation.

5. Summary

The commercial nuclear power program in the United States is currently based on use of

light-water reactors that use low-enriched uranium fuel, although other types of power plants and fuels are feasible and may later be used. The principal federal entities that regulate operation of these power plants for the protection of both workers and the general public are the Nuclear Regulatory Commission and the Environmental Protection Agency. The design of a monitoring program to assure compliance with applicable requirements must consider both the limitations imposed on emissions or exposures and the possible sources of radiation or radioactivity at the particular facility. Substantial monitoring efforts are required both at the facility, internally and at emission points, and in the environment, which may be affected by plant emissions. Appropriate instrumentation has been designed for the important emissions from nuclear power plants, and the reader is referred to more detailed discussions of instruments designed for particular types of radiation (Division III) or particular radionuclides (Division IV).

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APPENDIX A

PART 20—STANDARDS FOR PROTECTION AGAINST RADIATION

GENERAL PROVISIONS

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20.3 Definitions.
20.4 Units of radiation dose.
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PERMISSIBLE DOSES, LEVELS AND CONCENTRATIONS

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in restricted areas.
20.102 Determination of accumulated dose.
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trations of radioactive materials in air in
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APPENDIX I—NUMERICAL GUIDES FOR DESIGN OBJECTIVES AND LIMITING CONDITIONS FOR OPERATION TO MEET THE CRITERION “AS LOW AS PRACTICABLE” FOR RADIOACTIVE MATERIAL IN LIGHT-WATER-COOLED NUCLEAR POWER REACTOR EFFLUENTS

SECTION I. Introduction. Section 50.34a provides that an application for a permit to construct a nuclear power reactor shall include a description of the preliminary design of equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced

during normal reactor operations, including expected operational occurrences. In the case of an application filed on or after January 2, 1971, the application must also identify the design objectives, and the means to be employed, for keeping levels of radioactive material in effluents to unrestricted areas as low as practicable.

Section 50.36a contains provisions designed to assure that releases of radioactive material from nuclear power reactors to unrestricted areas during normal reactor operations, including expected operational occurrences, are kept as low as practicable.

This Appendix provides numerical guides for design objectives and limiting conditions for operation to assist applicants for, and holders of, licenses for light-water-cooled nuclear power reactors in meeting the requirements of §§ 50.34a and 50.36a that radioactive material in effluents released from these facilities to unrestricted areas be kept as low as is reasonably achievable. Design objectives and limiting conditions for operation conforming to the guidelines of this Appendix shall be deemed a conclusive showing of compliance with the “as low as is reasonably achievable” requirements of 10 CFR 50.34a and 50.36a. Design objectives and limiting conditions for operation differing from the guidelines may also be used, subject to a case-by-case showing of a sufficient basis for the findings of “as low as is reasonably achievable” required by §§ 50.34a and 50.36a. The guides presented in this Appendix are appropriate only for light-water-cooled nuclear power reactors and not for other types of nuclear facilities.

SEC. II. Guides on design objectives for light-water-cooled nuclear power reactors licensed under 10 CFR Part 50. The guides on design objectives set forth in this section may be used by an applicant for a permit to construct a light-water-cooled nuclear power reactor as guidance in meeting the requirements of § 50.34a(a). The applicant shall provide reasonable assurance that the following design objectives will be met.

A. The calculated annual total quantity of all radioactive material above background¹ to be released from each light-water-cooled nuclear power reactor to unrestricted areas will not result in an estimated annual dose or dose commitment from liquid effluents

¹Here and elsewhere in this Appendix background means radioactive materials in the environment and in the effluents from light-water-cooled power reactors not generated in, or attributable to, the reactors of which specific account is required in determining design objectives.

for any individual in an unrestricted area from all pathways of exposure in excess of 3 millirems to the total body or 10 millirems to any organ.

B.1. The calculated annual total quantity of all radioactive material above background to be released from each light-water-cooled nuclear power reactor to the atmosphere will not result in an estimated annual air dose from gaseous effluents at any location near ground level which could be occupied by individuals in unrestricted areas in excess of 10 millirads for gamma radiation or 20 millirads for beta radiation.

2. Notwithstanding the guidance of paragraph B.1:

(a) The Commission may specify, as guidance on design objectives, a lower quantity of radioactive material above background to be released to the atmosphere if it appears that the use of the design objectives in paragraph B.1 is likely to result in an estimated annual external dose from gaseous effluents to any individual in an unrestricted area in excess of 5 millirems to the total body; and

(b) Design objectives based upon a higher quantity of radioactive material above background to be released to the atmosphere than the quantity specified in paragraph B.1 will be deemed to meet the requirements for keeping levels of radioactive material in gaseous effluents as low as is reasonably achievable if the applicant provides reasonable assurance that the proposed higher quantity will not result in an estimated annual external dose from gaseous effluents to any individual in unrestricted areas in excess of 5 millirems to the total body or 15 millirems to the skin.

C. The calculated annual total quantity of all radioactive iodine and radioactive material in particulate form above background to be released from each light-water-cooled nuclear power reactor in effluents to the atmosphere will not result in an estimated annual dose or dose commitment from such radioactive iodine and radioactive material in particulate form for any individual in an unrestricted area from all pathways of exposure in excess of 15 millirems to any organ.

D. In addition to the provisions of paragraphs A, B, and C above, the applicant shall include in the radwaste system all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. As an interim measure and until establishment and adoption of

better values (or other appropriate criteria), the values \$1000 per total body man-rem and \$1000 per man-thyroid-rem (or such lesser values as may be demonstrated to be suitable in a particular case) shall be used in this cost-benefit analysis. The requirements of this paragraph D need not be complied with by persons who have filed applications for construction permits which were docketed on or after January 2, 1971, and prior to June 4, 1976, if the radwaste systems and equipment described in the preliminary or final safety analysis report and amendments thereto satisfy the Guides on Design Objectives for Light-Water-Cooled Nuclear Power Reactors proposed in the Concluding Statement of Position of the Regulatory Staff in Docket-RM-50-2 dated February 20, 1974, pp. 25-30, reproduced in the Annex to this Appendix I.

SEC. III. *Implementation.* A.1. Conformity with the guides on design objectives of Section II shall be demonstrated by calculational procedures based upon models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated, all uncertainties being considered together. Account shall be taken of the cumulative effect of all sources and pathways within the plant contributing to the particular type of effluent being considered. For determination of design objectives in accordance with the guides of Section II, the estimations of exposure shall be made with respect to such potential land and water usage and food pathways as could actually exist during the term of plant operation: *Provided, That*, if the requirements of paragraph B of Section III are fulfilled, the applicant shall be deemed to have complied with the requirements of paragraph C of Section II with respect to radioactive iodine if estimations of exposure are made on the basis of such food pathways and individual receptors as actually exist at the time the plant is licensed.

2. The characteristics attributed to a hypothetical receptor for the purpose of estimating internal dose commitment shall take into account reasonable deviations of individual habits from the average. The applicant may take account of any real phenomenon or factors actually affecting the estimate of radiation exposure, including the characteristics of the plant, modes of discharge of radioactive materials, physical processes tending to attenuate the quantity of radioactive material to which an individual would be exposed, and the effects of averaging exposures over times during which determining factors may fluctuate.

B. If the applicant determines design objectives with respect to radioactive iodine on

the basis of existing conditions and if potential changes in land and water usage and food pathways could result in exposures in excess of the guideline values of paragraph C of Section II, the applicant shall provide reasonable assurance that a monitoring and surveillance program will be performed to determine:

1. The quantities of radioactive iodine actually released to the atmosphere and deposited relative to those estimated in the determination of design objectives;
2. Whether changes in land and water usage and food pathways which would result in individual exposures greater than originally estimated have occurred; and
3. The content of radioactive iodine and foods involved in the changes, if and when they occur.

SEC. IV. *Guides on technical specifications for limiting conditions for operation for light-water-cooled nuclear power reactors licensed under 10 CFR Part 50.* The guides on limiting conditions for operation for light-water-cooled nuclear power reactors set forth below may be used by an applicant for a license to operate a light-water-cooled nuclear power reactor as guidance in developing technical specifications under § 50.36a(a) to keep levels of radioactive materials in effluents to unrestricted areas as low as is reasonably achievable.

Section 50.36a(b) provides that licensees shall be guided by certain considerations in establishing and implementing operating procedures specified in technical specifications that take into account the need for operating flexibility and at the same time assure that the licensee will exert his best effort to keep levels of radioactive material in effluents as low as is reasonably achievable. The guidance set forth below provides additional and more specific guidance to licensees in this respect.

Through the use of the guides set forth in this Section it is expected that the annual releases of radioactive material in effluents from light-water-cooled nuclear power reactors can generally be maintained within the levels set forth as numerical guides for design objectives in Section II.

At the same time, the licensee is permitted the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power even under unusual operating conditions which may temporarily result in releases higher than such numerical guides for design objectives but still within levels that assure that the average population exposure is equivalent to small fractions of doses from natural background

radiation. It is expected that in using this operational flexibility under unusual operating conditions, the licensee will exert his best efforts to keep levels of radioactive material in effluents within the numerical guides for design objectives.

A. If the quantity of radioactive material actually released in effluents to unrestricted areas from a light-water-cooled nuclear power reactor during any calendar quarter is such that the resulting radiation exposure, calculated on the same basis as the respective design objective exposure, would exceed one-half the design objective annual exposure derived pursuant to Sections II and III, the licensee shall:²

1. Make an investigation to identify the causes for such release rates;

2. Define and initiate a program of corrective action; and

3. Report these actions to the appropriate NRC Regional Office shown in Appendix D of Part 20 of this Chapter with a copy to the Director of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, within 30 days from the end of the quarter during which the release occurred.

B. The licensee shall establish an appropriate surveillance and monitoring program to:

1. Provide data on quantities of radioactive material released in liquid and gaseous effluents to assure that the provisions of paragraph A of this section are met;

2. Provide data on measurable levels of radiation and radioactive materials in the environment to evaluate the relationship between quantities of radioactive material released in effluents and resultant radiation doses to individuals from principal pathways of exposure; and

3. Identify changes in the use of unrestricted areas (e.g., for agricultural purposes) to permit modifications in monitoring programs for evaluating doses to individuals from principal pathways of exposure.

C. If the data developed in the surveillance and monitoring program described in

²Section 50.36a(a)(2) requires the licensee to submit certain reports to the Commission with regard to the quantities of the principal radionuclides released to unrestricted areas. It also provides that, on the basis of such reports and any additional information the Commission may obtain from the licensee and others, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate.

paragraph B of this section and in paragraph B of Section III or from other monitoring programs show that the relationship between the quantities of radioactive material released in liquid and gaseous effluents and the dose to individuals in unrestricted areas is significantly different from that assumed in the calculations used to determine design objectives pursuant to Sections II and III, the Commission may modify the quantities in the technical specifications defining the limiting conditions for operation in a license authorizing operation of a light-water-cooled nuclear power reactor.

SEC. V. *Effective dates.* A. The guides for limiting conditions for operation set forth in this Appendix shall be applicable in any case in which an application was filed on or after January 2, 1971, for a permit to construct a light-water-cooled nuclear power reactor.

B. For each light-water-cooled nuclear power reactor constructed pursuant to a permit for which application was filed prior to January 2, 1971, the holder of the permit or a license, authorizing operation of the reactor shall, within a period of twelve months from June 4, 1975, file with the Commission:

1. Such information as is necessary to evaluate the means employed for keeping levels of radioactivity in effluents to unrestricted areas as low as is reasonably achievable, including all such information as is required by § 50.34a (b) and (c) not already contained in his application; and

2. Plans and proposed technical specifications developed for the purpose of keeping releases of radioactive materials to unrestricted areas during normal reactor operations, including expected operational occurrences, as low as is reasonably achievable.

CONCLUDING STATEMENT OF POSITION OF THE REGULATORY STAFF (DOCKET-RM-50-2)

GUIDES ON DESIGN OBJECTIVES FOR LIGHT- WATER-COOLED NUCLEAR POWER REACTORS

A. For radioactive material above background¹ in liquid effluents to be released to unrestricted areas:

1. The calculated annual total quantity of all radioactive material from all light-water-cooled nuclear power reactors at a site should not result in an annual dose or dose commitment to the total body or to any organ of an individual in an unrestricted

area from all pathways of exposure in excess of 5 millirems; and

2. The calculated annual total quantity of radioactive material, except tritium and dissolved gases, should not exceed 5 curies for each light-water-cooled reactor at a site.

3. Notwithstanding the guidance in paragraph A.2, for a particular site, if an applicant for a permit to construct a light-water-cooled nuclear power reactor has proposed baseline in-plant control measures² to reduce the possible sources of radioactive material in liquid effluent releases and the calculated quantity exceeds the quantity set forth in paragraph A.2, the requirements for design objectives for radioactive material in liquid effluents may be deemed to have been met provided:

a. The applicant submits an evaluation of the potential for effects from long-term buildup in the environment in the vicinity of the site of radioactive material, with a radioactive half-life greater than one year, to be released; and

b. The provisions of paragraph A.1 are met.

B. For radioactive material above background in gaseous effluents the annual total quantity of radioactive material to be released to the atmosphere by all light-water-cooled nuclear power reactors at a site:

1. The calculated annual air dose due to gamma radiation at any location near ground level which could be occupied by individuals at or beyond the boundary of the site should not exceed 10 millirads; and

2. The calculated annual air dose due to beta radiation at any location near ground level which could be occupied by individuals at or beyond the boundary of the site should not exceed 20 millirads.

3. Notwithstanding the guidance in paragraphs B.1 and B.2, for a particular site:

a. The Commission may specify, as guidance on design objectives, a lower quantity of radioactive material above background in gaseous effluents to be released to the atmosphere if it appears that the use of the design objectives described in paragraphs

²Such measures may include treatment of clear liquid waste streams (normally tritiated, nonaerated, low conductivity equipment drains and pump seal leakoff), dirty liquid waste streams (normally nontritiated, aerated, high conductivity building sumps, floor and sample station drains), steam generator blowdown streams, chemical waste streams, low purity and high purity liquid streams (resin regenerate and laboratory wastes), as appropriate for the type of reactor.

¹"Background," means the quantity of radioactive material in the effluent from light-water-cooled nuclear power reactors at a site that did not originate in the reactors.

B.1 and B.2 is likely to result in an annual dose to an individual in an unrestricted area in excess of 5 millirems to the total body or 15 millirems to the skin; or

b. Design objectives based on a higher quantity of radioactive material above background in gaseous effluents to be released to the atmosphere than the quantity specified in paragraphs B.1 and B.2 may be deemed to meet the requirements for keeping levels of radioactive material in gaseous effluents as low as practicable if the applicant provides reasonable assurance that the proposed higher quantity will not result in annual doses to an individual in an unrestricted area in excess of 5 millirems to the total body or 15 millirems to the skin.

C. For radioactive iodine and radioactive material in particulate form above background released to the atmosphere:

1. The calculated annual total quantity of all radioactive iodine and radioactive material in particulate form from all light-water-cooled nuclear power reactors at a site should not result in an annual dose or dose commitment to any organ of an individual in an unrestricted area from all pathways of exposure in excess of 15 millirems. In determining the dose or dose commitment the portion thereof due to intake of radioactive material via the food pathways may be evaluated at the locations where the food pathways actually exist; and

2. The calculated annual total quantity of iodine-131 in gaseous effluents should not exceed 1 curie for each light-water-cooled nuclear power reactor at a site.

3. Notwithstanding the guidance in paragraphs C.1 and C.2 for a particular site, if an applicant for a permit to construct a light-water-cooled nuclear power reactor

has proposed baseline in-plant control measures³ to reduce the possible sources of radioactive iodine releases, and the calculated annual quantities taking into account such control measures exceed the design objective quantities set forth in paragraphs C.1 and C.2, the requirements for design objectives for radioactive iodine and radioactive material in particulate form in gaseous effluents may be deemed to have been met provided the calculated annual total quantity of all radioactive iodine and radioactive material in particulate form that may be released in gaseous effluents does not exceed four times the quantity calculated pursuant to paragraph C.1.

(Sec. 201, Pub. L. 93-438, 88 Stat. 1242 (42 U.S.C. 5841))

[40 FR 19442, May 5, 1975, as amended at 40 FR 40818, Sept. 4, 1975; 40 FR 58847, Dec. 17, 1975; 41 FR 16447, Apr. 19, 1976; 42 FR 20139, Apr. 18, 1977]

PART 100—REACTOR SITE CRITERIA

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100.1 Purpose.

100.2 Scope.

100.3 Definitions.

SITE EVALUATION FACTORS

100.10 Factors to be considered when evaluating sites.

100.11 Determination of exclusion area, low population zone, and population center distance.

APPENDIX A—Seismic and Geologic Siting Criteria for Nuclear Power Plants

APPENDIX B

U. S. Nuclear Regulatory Commission Regulatory Guides

Regulatory Guides are issued to describe and make available to the public methods acceptable to the NRC staff of implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in evaluating specific problems of postulated accidents, or to provide guidance to applicants. Regulatory Guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.

Comments and suggestions for improvements in these guides are encouraged at all times, and guides will be revised, as appropriate, to accommodate comments and to reflect new information or experience. However, the staff's consideration of comments received during the initial public comment period for this guide has resulted in the determination that there is no need for a revision at this time.

Comments should be sent to the Secretary of the Commission, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Docketing and Service Branch.

The guides are issued in the following ten broad divisions:

1. Power Reactors
2. Research and Test Reactors
3. Fuels and Materials Facilities
4. Environmental and Siting
5. Materials and Plant Protection
6. Products
7. Transportation
8. Occupational Health
9. Antitrust Review
10. General.

Requests for single copies of issued guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future guides in specific divisions should be made in writing to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Document Control.

DIVISION 1 REGULATORY GUIDES POWER REACTORS (selected)

<u>Number</u>	<u>Title</u>
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors (Revision 2, 6/74).
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors (Revision 2, 6/74).
1.5	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors (Safety Guide 5, 3/10/71).
1.21	Measuring Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants (Revision 1, 6/74).
1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25, 3/23/72).
1.52	Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants (Revision 1, 7.76).
1.98	Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor (3.76).
1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I (3/76).
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors (Revision O-R, 3/76; reissued 5/77).
1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I (Revision 1, 4/77).

* Design Basis Guidance for Radioactive Waste Management Systems Installed in Light-Water-Cooled Power Plants

DIVISION 4 REGULATORY GUIDES ENVIRONMENTAL AND SITING (Selected)

<u>Number</u>	<u>Title</u>
4.1	Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants
4.5	Measurements of Radionuclides in the Environment - Sampling and Analysis of Plutonium in Soil
4.6	Measurements of Radionuclides in the Environment - Strontium-89 and Strontium-90 Analyses
4.11	Terrestrial Environmental Studies for Nuclear Power Stations
4.13	Performance, Testing, and Procedural Specifications for Thermoluminescence Dosimetry: Environmental Applications
4.15	Quality Assurance for Radiological Monitoring Programs (Normal Operations) - Effluent Streams and the Environment (For Comment).

* Under development.

U.S. NUCLEAR REGULATORY COMMISSION
REGULATORY GUIDE
OFFICE OF STANDARDS DEVELOPMENT

REGULATORY GUIDE 4.1

PROGRAMS FOR MONITORING RADIOACTIVITY IN THE ENVIRONS
OF NUCLEAR POWER PLANTS

A. INTRODUCTION

General Design Criterion 64, "Monitoring Radioactivity Releases," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that licensees provide means for monitoring the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

Paragraph 20.106(e) of 10 CFR Part 20, "Standards for Protection Against Radiation," provides that the Commission may limit the quantities of radioactive materials released in air or water during a specified period of time to ensure that the daily intake of radioactive materials from air, water, or food by a suitable sample of an exposed population group, averaged over a time period not exceeding one year, would not exceed specified quantities. Section 20.201 of 10 CFR Part 20 further requires that a licensee conduct surveys of levels of radiation or concentrations of radioactive material as necessary to show compliance with Commission regulations.

This guide describes a basis acceptable to the NRC staff for the design of programs for monitoring levels of radiation and radioactivity in the plant environs.

B. DISCUSSION

Present requirements to keep levels of radioactive material in effluents as low as practicable (as specified in 10 CFR §50.34a) should ensure that radiation doses to the public resulting from effluent releases will continue to remain minimal. The principles presented in this guide are considered an acceptable basis for establishing preoperational and operational monitoring programs to provide information needed to determine whether exposures in the environment are within established limits

and to ensure that long-term buildup of specific radionuclides in the environment will not become significant.

A preoperational program should be conducted in the environs of each proposed nuclear power plant site to (1) measure background levels and their variations in environmental media in the area surrounding the plant, (2) evaluate procedures, equipment, and techniques, and (3) provide experience to personnel.

Years of experience at various Atomic Energy Commission facilities have demonstrated that specific radionuclides behave in known ways under given environmental conditions. Therefore, analyses of "indicator media" can be used to define radionuclide levels in the environment. The "indicator medium" (or organism) concept of environmental surveillance involves the sampling of organisms and media which are sensitive and reliable measures of the quantities of individual radionuclides cycling through pathways. For example, where the plant-cow-milk-man food chain is determined to be an important pathway, it may not be necessary to extensively sample and measure grazing plants and fodder to keep track of iodine-131 cycling in the food chain, since sampling and measuring the milk produced by dairy cows in surrounding areas may be adequate.

After the plant is in operation, a program for measuring radiation levels and radioactivity in the plant environs must be maintained on a continuing basis to assist in verifying anticipated radioactivity concentrations and related public exposures. The initial preoperational and operational monitoring programs should be designed in accordance with the following criteria:

1. They should be based on the analysis of important pathways for the anticipated types and quantities of radionuclides released from the plant into the surrounding environment;

USNRC REGULATORY GUIDES

Regulatory Guides are issued to describe and make available to the public methods acceptable to the NRC staff of implementing specific parts of the Commission's regulations to delineate techniques used by the staff in evaluating specific problems or postulated accidents or to provide guidance to applicants. Regulatory Guides are not substitutes for regulations and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.

Comments and suggestions for improvements in these guides are encouraged at all times and guides will be revised as appropriate to accommodate comments and to reflect new information or experience. However, comments on this guide if received within about two months after its issuance will be particularly useful in evaluating the need for an early revision.

Comments should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Section.

The guides are issued in the following ten broad divisions:

1. Power Reactors	6. Products
2. Research and Test Reactors	7. Transportation
3. Fuels and Materials Facilities	8. Occupational Health
4. Environmental and Siting	9. Antitrust Review
5. Materials and Plant Protection	10. General

Copies of published guides may be obtained by written request indicating the divisions desired to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Office of Standards Development.

2. They should consider the possibility of buildup of long-lived radionuclides in the environment and identify physical and biological sites of accumulation that may contribute to human exposures;

3. They should consider the potential radiation exposure to important plants and animals;¹

4. They should be designed to establish correlations between levels of radiation and radioactivity in the environment and radioactive releases from plant operation. (A variety of techniques, including measurements at control locations, preoperational surveys, and comparisons of operating versus shutdown levels of radioactivity in the environs may all be useful for this purpose.) Information of this nature will be of considerable help in modifying the initial environmental measurements program.

Data obtained from this program should be used by the licensee to show that environmental levels are consistent with those determined from the effluent radiation monitoring and sampling program. The latter program is used together with dispersion estimates to ensure that plant releases to the environment and radiation doses to the public are maintained within the numerical design objectives determined to be as low as practicable.

The Commission's staff has evaluated the types of information needed to provide supporting evidence for assessing the performance of the plant with respect to keeping population exposures as low as practicable and to verify predictions of concentrations of specific radionuclides in the environment based on effluent measurements at the plant. Based on this evaluation and on a review and assessment of existing licensees' monitoring and reporting programs, the staff has developed the regulatory position set forth below.

C. REGULATORY POSITION

The program for monitoring radioactivity in the environs of nuclear power plants should provide suitable information from which levels of radiation and radioactivity in the environs of each plant can be estimated.

¹ A species, whether animal or plant, is "important" if a specific causal link can be identified or hypothesized between the nuclear power plant and the species and if one or more of the following criteria applies: (1) the species is commercially or recreationally valuable, (2) the species is threatened or endangered, (3) the species affects the well-being of some important species within criteria (1) or (2), or (4) the species is critical to the structure and function of the ecological system or is a biological indicator of radionuclides in the environment. A threatened or endangered species is any species officially designated as such by the Secretary of the Interior or the Secretary of Commerce.

This information also may provide supporting evidence in evaluating the performance of systems and equipment installed to control releases of radioactive material to the environment.

The basic principles set forth in this guide constitute an acceptable basis for use in establishing an environmental monitoring program. These same principles will also be used as bases in developing the licensee's corresponding technical specifications. Guidance on the format, content, and preparation of Appendix B environmental technical specifications for operating licenses is presented in Regulatory Guide 4.8, "Environmental Technical Specifications for Nuclear Power Plants."

1. Preoperational Program

A preoperational monitoring program should be initiated two years prior to operation to provide a sufficient data base for comparison with operational data and to provide experience that will improve the efficiency of the operational program. It may not be necessary for all media in the preoperational program to be sampled for the entire two-year period; i.e., for those media sampled frequently, a six-month or one-year period is usually sufficient. However, if the sampling period is not representative, because of abnormal conditions, the period may require extension until representative samples are obtained.

2. Operational Program

Generally, the operational program should be an extension of the preoperational program to ensure that data from the two programs are compatible and that a smooth transition between programs is achieved.

a. Sample Media

Where practical, a suitable indicator medium or organism in each important pathway should be sampled and analyzed for the plant-contributed radionuclides released to the environment. When sampling organisms, an abundant, readily available species with known habits should be selected.

Careful attention should be given to sampling programs to avoid inducing serious stress on populations of important species. Sampling of large numbers of organisms could result in a temporary decline or permanent loss of desirable populations. In such cases, other indigenous but abundant species should be substituted as indicator organisms to provide an estimate of the radionuclides available to man through natural food chains. In some instances, properly selected and sampled vegetation may also provide a good measure of the radionuclides in a pathway.

Where use of a single indicator medium is impractical, samples of several media from each pathway should be collected and analyzed. The Commission recognizes that some pathways do not have more than one environmental medium; e.g., external radiation exposures from clouds of gamma-ray-emitting radionuclides involve only one pathway — the atmosphere. The actual number of media to be sampled in each pathway should be determined on a case-by-case basis for each site. In some cases, field measurements may be preferable to collecting samples for laboratory analysis.

The program should include sampling of environmental media to estimate radionuclide concentrations in important biota (see B.3 above). Radiation exposures (external) and internal doses from short-lived nuclides may be estimated by calculations (using effluent measurements and appropriate dispersion and bioaccumulation factors) rather than by routine collection of samples of environmental media. In some cases, field measurements at certain locations to establish concentrations of specific radionuclides may be necessary, initially, to confirm predictions.

b. Sampling Frequency

When a radionuclide has a short half-life (minutes to days), it may be necessary to evaluate concentrations or radiation exposure by making measurements in the field (e.g., by use of thermoluminescent dosimeters, pressurized ion chambers, or in situ gamma spectrometers).

When a radionuclide with an intermediate half-life (weeks to months) is released continuously or frequently, sampling and analysis of environmental media in the associated pathway should generally be carried out at intervals no greater than two or three half-lives of the nuclide. For long-half-life radionuclides (years), measurements should generally be made at least once per year. Where seasonal or other temporal variations may be evident, the frequency of sampling should be planned to allow resolution of any important effects.

In some cases, sampling on a continuous basis may be necessary (e.g., air sampling and continuous measurement of cumulative external radiation exposure). Composites of some selected sample types such as air filters may be appropriate for measurement of long-lived radionuclides.

c. Program Scope

During the initial three years of commercial power operation (or other period corresponding to maximum fuel burnup in the initial core cycle), the measurement program should be relatively comprehensive in an attempt to verify any projected correlations between

radioactive effluents and levels in environmental media. The extent of measurement of environmental media should be based on the type, quantity, and concentration of radionuclides expected from the plant as well as the results obtained from previous measurements.

If, after this period, the licensee is able to demonstrate from levels in environmental media or calculations (using measured effluents and appropriate dispersion and bioaccumulation factors) that the doses and concentrations associated with a particular pathway are sufficiently small, the number of media sampled in the pathway and the frequency of sampling may be reduced. An adequate program with emphasis on indicator organisms and selected media should still be continued in order to confirm that the levels of radioactivity in environmental media remain small.

Results from all individual measurements should be retained by the licensee along with information on sampling location and date, sample size (e.g., wet/dry weight), sampling and analytical procedures, units of data presentation, and precision and accuracy associated with individual measurements. Explanations of anomalous measurements should be provided.

d. Analyses

Samples should be analyzed for plant-contributed radionuclides released to the environment. Gross radioactivity measurements alone are generally not adequate for radiological monitoring. However, gross radioactivity measurements may be useful to indicate the concentration of a specific radionuclide when such measurements are shown to be truly indicative of the actual quantity or concentration of that radionuclide.

e. Quality Control

Control checks and tests should be applied to the analytical process by the use of blind duplicate analyses of selected samples and by cross-check analysis of selected samples with an independent laboratory. Quality controls should also be applied to the entire sample-collection procedure to ensure that representative samples are obtained and that samples are not changed, cross-contaminated, or otherwise affected prior to their analysis because of handling or because of their storage environment.

3. Detection Capabilities

The detection capabilities associated with measuring and analyzing radioactivity levels should be established primarily on the basis of potential human dose. These detection capabilities will vary depending on the instrumentation and analytical techniques used. Because of the need for a preoperational monitoring program,

detection capabilities for a particular program should be determined during an early stage of licensing. Every reasonable effort should be made to achieve detection capabilities that will detect radiation levels or radioactivity concentrations in pathways that could result in radiation doses corresponding to a few percent of the Federal Radiation Council's radiation protection guides (i.e., a few percent of 170 mrem/yr for whole body dose to a suitable sample of the exposed population).² Similarly, the detection capability of environmental measurements should be the most sensitive that is practicably achievable for measuring plant-contributed radionuclides in the environment.

²Federal Radiation Council Report No. 1, *Background Material for the Development of Radiation Protection Standards*, May 13, 1960.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for utilizing this regulatory guide.

This guide reflects current regulatory practice. Therefore, except in those cases in which the applicant proposes an alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used in the evaluation of submittals for operating license or construction permit applications docketed after the date of issue of this guide.

If an applicant wishes to use this regulatory guide in developing submittals for applications docketed on or prior to the date of issue of this guide, the pertinent portions of the application will be evaluated on the basis of this guide.

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- * Radiation Protection Training at Uranium Fuel Fabrication Plants

APPENDIX C

CHAPTER 1

INTRODUCTION

This Environmental Radioactivity Surveillance Guide recommends methods for conducting a minimum level of environmental radiation surveillance outside the plant site boundary of light-water-cooled nuclear power facilities. An environmental surveillance program is presented to achieve uniformity so that the data will be compatible and subject to singular interpretation relative to the estimated population radiation dose. The basic concepts presented may also apply to surveillance around other nuclear facilities such as gas-cooled and liquid-metal-cooled nuclear power facilities and nuclear fuel reprocessing plants. However, as additional nuclear facilities of these types are licensed and operated, additional guides may be needed. This Guide recommends procedures but does not establish the requirements for any particular organization for conducting environmental surveillance.

Radionuclides released with the effluents from nuclear power facilities become dispersed in the environment and contribute some radiation dose to the population. Environmental radiation surveillance programs conducted around nuclear power facilities should as a minimum provide data which may be used (1) for population dose calculations which can be compared with Federal and State standards, (2) for the evaluation of buildup of environmental radioactivity, and (3) for public information purposes.

Technical information for development of this Guide was obtained from radiological surveillance studies conducted by the Environmental Protection Agency (EPA) at an operating boiling water reactor (1) and an operating pressurized water reactor (2). These studies provide information on quantities and characteristics of radioactive material released to the environment and on critical pathways by which the public may potentially be exposed as a result of the releases.

Appendix C: Excerpts from "Environmental Radioactivity Surveillance Guide," Report ORP/SID 72-2, U.S. Environmental Protection Agency, Office of Radiation Programs, Washington, D.C. 20460 (1972).

ENVIRONMENTAL RADIOACTIVITY SURVEILLANCE GUIDE

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Office of Radiation Programs
Surveillance and Inspection Division
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Figure 1. Pathways between radioactive materials released to the atmosphere and man (5)

Figure 2 . Pathways between radioactive materials released to ground and surface waters (including oceans) and man. (5)

Figure 3. Estimated distance of maximum ground level concentration as a function of Pasquill atmospheric stability conditions and stack height in meters (15)

Wind rose data plotted as a function of
Pasquill atmospheric stability conditions.

(a) Facility site	(d) Site at 1.5 km based on
condition 'C'
(b) Population center sample site	(e) Site at 3 km based on
condition 'D'
(c) Site at 0.6 km based on
condition 'B'	(f) Control site at >20 km
based on annual
average wind rose

Figure 4. Air particulate sample sites around a nuclear power facility
based on Pasquill atmospheric stability conditions.

Table 1. Offsite surveillance of operating light-water-cooled nuclear power facilities

Operation or sample type	Approximate number of samples and their locations	Collection frequency	Analysis type ^a and frequency
Air particulates	1 sample from the 3 locations of the highest offsite ground level concentrations 1 sample from 1-3 communities within a 10-mile radius of the facility 1 sample from a location greater than a 20-mile radius in the least prevalent annual wind direction ^d	Continuous collection—filter change as required	Gross long-lived β at filter change ^b Composite for gamma isotopic analysis and radiostrontium analysis ^c quarterly
Air iodine	Same sites as for air particulates	Continuous collection—canister changes as required	Analyze weekly unless absence of radioiodine can be demonstrated
Direct radiation	2 or more dosimeters placed at each of the locations of the air particulate samples which are located at the 3 highest offsite ground level concentrations 2 or more dosimeters placed at each of 3 other locations for which the highest annual offsite dose at ground level is predicted ^e 2 or more dosimeters placed at each of 1-3 communities within a 10-mile radius of the facility ^f 2 or more dosimeters placed at a location greater than a 20-mile radius in the least prevalent annual wind direction ^d	Quarterly	Gamma dose quarterly
Surface water ^g	1 upstream 1 downstream after dilution (e.g., 1 mile)	Monthly (Record status of discharge operations at time of sampling)	Gross β , gamma isotopic analysis ^b monthly. Composite for tritium and radiostrontium analysis ^c quarterly
Ground water	1 or 2 from sources most likely to be affected	Quarterly	Gross β , gamma isotopic analysis ^b and tritium quarterly
Drinking water	Any supplies obtained within 10 miles of the facility which could be affected by its discharges or the first supply within 100 miles if none exists within 10 miles	Continuous proportional samples ¹	Gross β , gamma isotopic analysis ^b monthly. Composite for tritium and radiostrontium analysis quarterly ^c
Sediment, benthic organisms and aquatic plants	1 directly downstream of outfall ¹ 1 upstream of outfall ¹ 1 at dam site downstream or in impoundments ¹	Semiannually	Gamma isotopic analysis semiannually
Milk	1 sample at nearest offsite dairy farm in the prevailing downwind direction 1 sample of milk from local dairy representative of milkshed for the area	Monthly	Gamma isotopic analysis and radiostrontium analysis monthly ^c
Fish and shellfish	1 of each of principal edible types from vicinity of outfall 1 of each of the sample types from area not influenced by the discharges	Semiannually	Gamma isotopic analysis semiannually on edible portions
Fruits and vegetables	1 each of principal food products grown near the point of maximum predicted annual ground concentration from stack releases and from any area which is irrigated by water in which liquid plant wastes have been discharged 1 each of the same foods grown at greater than 20 miles distance in the least prevalent wind direction	Annually (At harvest)	Gamma isotopic analysis annually on edible portions
Meat and poultry	Meat, poultry, and eggs from animals fed on crops grown within 10 miles of the facility at the prevailing downwind direction or where drinking water is supplied from a downstream source	Annually during or immediately following grazing season	Gamma isotopic analysis annually on edible portions
Quality control ^k	Samples as required for accurate sampling and analysis		Minimum frequency—annually

^a Gamma isotopic analysis means identification of gamma emitters plus quantitative results for radionuclides that may be attributable to the facility.

^b Particulate sample filters should be analyzed for gross beta after at least 24 hours to allow for radon and thoron daughter decay.

^c Radiostrontium analysis is to be done only if gamma isotopic analysis indicates presence of cesium-137 associated with nuclear power facility discharges.

^d The purpose of this sample is to obtain background information. If it is not practical to locate a site in accordance with the criterion, another site which provides valid background data should be used.

^e These sites based on estimated dose levels, as opposed to ground level concentrations where the dose may be affected by sky shine, high plumes, or direct radiation from the facility being monitored.

^f These locations will normally coincide with the air particulate samplers used in the monitored communities.

^g For facilities not located on a stream, the upstream sample should be a sample taken at a distance beyond significant influence of the discharges. The downstream sample should be taken in an area beyond the outfall which would allow for mixing and dilution. Upstream samples taken in a tidal area must be taken far enough upstream to be beyond the plant influence when the effluent is actually flowing upstream during incoming tides.

^h If gross beta exceed 30 pCi/liter.

¹ Drinking water samples should be taken continuously at the surface water intake to municipal water supplies. Alternatively, if a reservoir is used, drinking water samples should be taken from the reservoir monthly. If the holding time for the reservoir is less than 1 month, then the sampling frequency should equal this holdup time. Increases in concentration of activation and/or fission products at these sources necessitate the analysis of tap water for the purpose of dose calculations. Additional analyses of tap water may be necessary to satisfy public demand.

² See figure 6 for locations on a stream. For facilities located on large bodies of water, sampling sites should be located at the discharge point and in both directions along the shore line.

^k The Analytical Quality Control Service of the Surveillance and Inspection Division (SID) provides low-level radiochemical standards and interlaboratory services to State and local health departments, Federal and international agencies, and nuclear power facilities and their contractors. The Service operates several types of cross-check programs for the analysis of radionuclide in environmental media, such as milk, food, water, air, and soil. The samples are submitted on a routine schedule designed to fit the needs of each laboratory. Technical experiments are undertaken to permit detailed analyses of the accuracy and precision obtained by participating laboratories. In addition, low-level radioactivity standards are provided to the agencies participating in the various programs. Primary and secondary standardization is also performed as needed on those radionuclides not used on a routine basis.

Table 2. Detection capabilities associated with analytical methods of environmental radioactivity surveillance

Media and isotope	Analytical method from table 3	Sample size	Minimum detectable levels ^a	Annual dose associated with MDL (mrem/yr) ^b	Critical organ	Assumption for dose model	Annual intake ^c
Air particulates:							
Gross beta	E	300 m ³	3×10 ⁻³ pCi/m ³		Bone		7,300 m ³
⁹⁰ Sr	F	1,200 m ³	5×10 ⁻³ pCi/m ³	0.025	Bone		7,300 m ³
⁹⁰ Sr	F	1,200 m ³	1×10 ⁻³ pCi/m ³	.05	Bone		7,300 m ³
¹³⁴ Cs	A	1,200 m ³	1×10 ⁻² pCi/m ³	.005	Total body		7,300 m ³
¹³⁷ Cs	A	1,200 m ³	1×10 ⁻² pCi/m ³	.0025	Total body		7,300 m ³
¹⁴⁰ Ba-La	A	1,200 m ³	1×10 ⁻² pCi/m ³	.0038	GI (LLI)		7,300 m ³
Air gases:							
¹³¹ I	A	300 m ³	4×10 ⁻² pCi/m ³	1.0	Thyroid		^d 1,710 m ³
Short-lived gases	B	Not applicable	^e 20 mrem/yr	20	Total body		Not applicable
⁸⁶ Kr	C	1 m ³	1 pCi/m ³	.002	Skin		Not applicable
³ H (HTO)	D	10-15 ml of condensate	15×10 ⁻³ pCi/m ³	.000013	Body Tissue ^g		7,300 m ³
Water:							
⁵⁸ ⁶⁰ Co	A	100 ml	20 pCi/liter	.32	GI (LLI)		440 liters
⁵⁸ Co	B	3.5 liters	10 pCi/liter	.082	GI (LLI)		440 liters
⁶⁰ Co	B	3.5 liters	10 pCi/liter	.16	GI (LLI)		440 liters
¹⁴⁰ Ba-La	B	3.5 liters	10 pCi/liter	.41	GI (LLI)		440 liters
	H	1 liter	1.0 pCi/liter	.041			
	I	1 liter	1.0 pCi/liter	.041			
³ H	J	4-5 ml	200 pCi/liter	.018	Body tissue		440 liters
	K	10-15 ml	200 pCi/liter	.018			
	L	10-50 ml	400 pCi/liter	.036			
¹⁴ C	M	200 ml	30 pCi/liter	.031	Fat		440 liters
	S	500 ml	.6 pCi/liter	.0006			
⁹⁰ Sr	N	1 liter	5 pCi/liter	1.4	Bone		440 liters
	O	1 liter	5 pCi/liter	1.4			
	P	1 liter	5 pCi/liter	1.4			
	R	1 liter	5 pCi/liter	1.4			
⁹⁰ Sr	N	1 liter	1.0 pCi/liter	2.7	Bone		440 liters
	O	1 liter	1.0 pCi/liter	2.7			
	P	1 liter	1.0 pCi/liter	2.7			
	Q	1 liter	1.0 pCi/liter	2.7			
¹³⁴ ¹³⁷ Cs	B	1 liter	1.0 pCi/liter	.03	Total body		440 liters
¹³⁴ Cs	C	3.5 liters	10 pCi/liter	.30	Total body		440 liters
¹³⁷ Cs	C	3.5 liters	10 pCi/liter	.14	Total body		440 liters
⁶⁵ Zn	B	3.5 liters	20 pCi/liter	.054	Total body		440 liters
⁵⁴ Mn	B	3.5 liters	10 pCi/liter	.082	GI (LLI)		440 liters
	D	400 ml	40 pCi/liter	.32			
⁵⁵ Fe	E	100 ml	20 pCi/liter	.02	Spleen		440 liters
⁵⁹ Fe	B	3.5 liters	20 pCi/liter	.27	GI (LLI)		440 liters
	E	100 ml	100 pCi/liter	1.4			
¹³¹ I	B	3.5 liters	10 pCi/liter	27	Thyroid		440 liters
	F	100 ml	10 pCi/liter	27			
	T	10 liters	.04 pCi/liter	.1			
⁹⁰ Zr-Nb	B	3.5 liters	5 pCi/liter	.071	GI (LLI)		440 liters
⁹⁰ Zr	G	200 ml	15 pCi/liter	.21			
⁹⁰ Nb	G	200 ml	25 pCi/liter	.21	GI (LLI)		440 liters
Milk:							
⁹⁰ Sr	B	1 liter	5 pCi/liter	1.2	Bone		365 liters
	C	1 liter					
	D	1 liter					
⁹⁰ Sr	A	1 liter	1 pCi/liter	2.3	Bone		365 liters
	B	1 liter					
	C	1 liter					
	D	1 liter					
¹³¹ I	E	3.5 liters	10 pCi/liter	50	Thyroid		365 liters
¹³⁴ Cs	E	3.5 liters	10 pCi/liter	.25	Total body		365 liters
¹³⁷ Cs	E	3.5 liters	10 pCi/liter	.11	Total body		365 liters

See footnotes at end of table.

Table 2. Detection capabilities associated with analytical methods of environmental radioactivity surveillance continued

Media and isotope	Analytical method from table 3	Sample size	Minimum detectable levels ^a	Annual dose associated with MDL (mrem/yr) ^b	Critical organ	Assumption for dose model	Annual intake ^c
Shellfish (fish)							
⁶⁰ Co	A	200 grams	80 pCi/kg	027	GI (LLI)		18 25 kg
⁶⁰ Co	A	200 grams	80 pCi/kg	054	GI (LLI)		18 25 kg
¹³⁴ Cs	A	200 grams	80 pCi/kg	1	Total body		18 25 kg
¹³⁷ Cs	A	200 grams	80 pCi/kg	046	Total body		18 25 kg
⁶⁵ Zn	A	200 grams	160 pCi/kg	018	Total body		18 25 kg
⁵⁴ Mn	A	200 grams	80 pCi/kg	027	GI (LLI)		18 25 kg
⁵⁹ Fe	A	200 grams	160 pCi/kg	091	GI (LLI)		18 25 kg
⁵⁹ Fe	B ^d	100 grams	20 pCi/kg	00028	Spleen		18 25 kg
⁸⁹ Sr	C	200 grams	25 pCi/kg	28	Bone		18 25 kg
	D	200 grams	25 pCi/kg	28			
⁹⁰ Sr	C	200 grams	5 0 pCi/kg	55	Bone		18 25 kg
	D	200 grams	5 0 pCi/kg	55			
	E	200 grams	5 0 pCi/kg	55			

^a The minimum detectable levels (MDL) are practical detection levels, rather than theoretical detection levels. These levels are characteristic of the analytical procedure and the counting instrumentation in use. The MDLs listed assume the following instrumentation: (1) low background beta counter, (2) standard gamma scan—400 to 512 multichannel analyzer—4 by 4 inch NaI(Tl) detector, and (3) tritium—liquid scintillation counter. The detection limit for a specific radionuclide by gamma spectrometry is dependent upon the quantities of other radionuclides present in the sample. The detection limits tested are those practically obtained with the concentrations and mixtures of radionuclides normally encountered with environmental samples. If only a single radionuclide is present in a sample to be analyzed by gamma spectrometry, then the detection limits listed could probably be reduced by a factor of 2. The detection limits for specific nuclides would be considerably greater than those listed when complicated mixtures are encountered and in particular when certain constituents are present in relatively high concentrations.

^b These values were obtained by a simple ratio relating Radiation Protection Guides of the Federal Radiation Council (34) to the dose associated with these Guides. Actual dose calculations resulting from specific environmental levels should take into consideration additional factors relating to pathways intake and other environmental factors as appropriate.

^c Intake values assume standard man quantities or other referenced values as follows:

- 1 1 liter of milk per day for a 1-year-old child (17)
- 2 1 2 liters of water per day, adult (18)
- 3 20 cubic meters of air breathed per day for an adult (18)
- 4 4 7 cubic meters of air breathed per day for a 1-year-old child (17)

- 5 1 87 kilograms of food consumed per day for total diet of a teenager (19)
- 6 50 grams per day of shellfish (20)

^d The annual intake of air is for a child (age 1 year). In the case of ¹³¹I, the child thyroid is the limiting factor.

^e CaF₂ Mn dosimeter encapsulated in ⁴K free glass or equivalent.

^f Assuming temperature of 75° Fahrenheit and 90 percent relative humidity.

^g The critical organ for ³H gas may be the skin, depending upon the state of the ³H (gaseous or oxide). The body tissue is used as the most conservative case.

^h Assumed worst case mixture of 100 percent of ⁶⁰Co.

ⁱ Procedure B under shellfish is for aqueous solutions so that preliminary sample preparation is necessary prior to entering this procedure.

Table 3. Analytical methods for routine environmental radioactivity surveillance

Media	Code	Analytical method	References
Air	A	Gamma Spectrometry for Iodine-131 of Air Filters or Cartridge Samples (6 2 2)	(8)
	B	Dosimeter-External Exposure	(13)
	C	Cryogenic Separation and Liquid Scintillation Counting	(21)
	D	Determination of Tritium in Water	(9, 14)
	E	Gross Beta Counting of Air Filters (6 2 1)	(8)
	F	Oxalate Precipitation (6 2 6)	(8)
Water	A	Cobalt and Nickel	(10)
	B	Gamma Analysis in Water (5 2 6)	(8)
	C	Cesium—Phosphomolybdate—Chloroplatinate Method	(10)
	D	Radioactive Manganese (ASTM D2039-69)	(10)
	E	Radioactive Iron (ASTM D2461-69)	(10)
	F	Radioactive Iodine Distillation (ASTM D2334-68)	(10)
	G	Zirconium-Niobium-95	(10)
	H	Basic Carbonate Method (5 2 3)	(8)
	I	Radioactive Barium (ASTM D2038-68)	(10)
	J	Radioactive Tritium (ASTM D2476-69)	(10)
	K	Determination of Tritium in Water	(9, 14)
	L	Distillation Method—Tritium (5 2 5)	(8)
	M	Carbon-14—Distillation to CaCO ₃	(10)
	N	Basic Carbonate Method for Saline Water (5 2 2)	(8)
	O	Radiostromium in Saline Water (5 2 2)	(8)
	P	Oxalate Precipitation Method (5 2 2)	(8)
	Q	TBP Extraction (Strontium-90) (5 2 2)	(8)
	R	Difference Method (Strontium-89) (5 2 2)	(8)
Milk	S	Carbonate Carbon-14 by Liquid Scintillation Counting	(22)
	T	Iodine-131 by Ion Exchange and Precipitation	(22)
	A	Reference Method—TBP Extraction (3 2 1)	(8)
	B	TCA Precipitation Method—Nitric Acid Separation (3 2 2)	(8, 9)
	C	Ion-Exchange Procedure—TBP Extraction (3 2 3)	(8, 9)
Shellfish	D	Batch Ion-Exchange Procedure—TBP Extraction (3 2 3)	(8)
	E	Gamma Spectrometry on Milk (3 2 4)	(8)
	A	Gamma Spectrometry of Soft Tissue (7 2 6)	(8)
	B	Radioactive Iron (ASTM D2461-69)	(10)
	C	TTA Extraction Method (4 2 2)	(8)
	D	TBP Extraction Method (4 2 3)	(8)
	E	HDEHP Extraction Method (4 2 1)	(8)

APPENDIX D

SUBCHAPTER F—RADIATION PROTECTION PROGRAMS

PART 190—ENVIRONMENTAL RADIATION PROTECTION STANDARDS FOR NUCLEAR POWER OPERATIONS

Subpart A—General Provisions

Sec.

190.01 Applicability.

190.02 Definitions.

Subpart B—Environmental Standards for the Uranium Fuel Cycle

190.10 Standards for normal operations.

190.11 Variances for unusual operations.

190.12 Effective date.

AUTHORITY: Atomic Energy Act of 1954, as amended; Reorganization Plan No. 3, of 1970.

SOURCE: 42 FR 2860, Jan. 13, 1977, unless otherwise noted.

Subpart A—General Provisions

§ 190.01 Applicability.

The provisions of this Part apply to radiation doses received by members of the public in the general environment and to radioactive materials introduced into the general environment as the result of operations which are part of a nuclear fuel cycle.

§ 190.02 Definitions.

(a) "Nuclear fuel cycle" means the operations defined to be associated with the production of electrical power for public use by any fuel cycle through utilization of nuclear energy.

(b) "Uranium fuel cycle" means the operations of milling of uranium ore, chemical conversion of uranium, isotopic enrichment of uranium, fabrication of uranium fuel, generation of electricity by a light-water-cooled nuclear power plant using uranium fuel, and reprocessing of spent uranium fuel, to the extent that these directly

support the production of electrical power for public use utilizing nuclear energy, but excludes mining operations, operations at waste disposal sites, transportation of any radioactive material in support of these operations, and the reuse of recovered non-uranium special nuclear and by-product materials from the cycle.

(c) "General environment" means the total terrestrial, atmospheric and aquatic environments outside sites upon which any operation which is part of a nuclear fuel cycle is conducted.

(d) "Site" means the area contained within the boundary of a location under the control of persons possessing or using radioactive material on which is conducted one or more operations covered by this Part.

(e) "Radiation" means any or all of the following: alpha, beta, gamma, or X-rays; neutrons; and high-energy electrons, protons, or other atomic particles; but not sound or radio waves, nor visible, infrared, or ultraviolet light.

(f) "Radioactive material" means any material which spontaneously emits radiation.

(g) "Curie" (Ci) means that quantity of radioactive material producing 37 billion nuclear transformations per second. (One millicurie (mCi)=0.001 Ci.)

(h) "Dose equivalent" means the product of absorbed dose and appropriate factors to account for differences in biological effectiveness due to the quality of radiation and its spatial distribution in the body. The unit of dose equivalent is the "rem." (One millirem (mrem)= 0.001 rem.)

(i) "Organ" means any human organ exclusive of the dermis, the epidermis, or the cornea.

(j) "Gigawatt-year" refers to the quantity of electrical energy produced at the busbar of a generating station. A gigawatt is equal to one billion watts. A gigawatt-year is equivalent to the amount of energy output represented by an average electric power level of one gigawatt sustained for one year.

(k) "Member of the public" means any individual that can receive a radiation dose in the general environment, whether he may or may not also be exposed to radiation in an occupation associated with a nuclear fuel cycle. However, an individual is not considered a member of the public during any period in which he is engaged in carrying out any operation which is part of a nuclear fuel cycle.

(l) "Regulatory agency" means the government agency responsible for issuing regulations governing the use of sources of radiation or radioactive materials or emissions therefrom and carrying out inspection and enforcement activities to assure compliance with such regulations.

Subpart B—Environmental Standards for the Uranium Fuel Cycle

§ 190.10 Standards for normal operations.

Operations covered by this Subpart shall be conducted in such a manner as to provide reasonable assurance that:

(a) The annual dose equivalent does not exceed 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public as the result of exposures to planned discharges of radioactive materials, radon and its daughters excepted, to the general environment from uranium fuel cycle op-

erations and to radiation from these operations.

(b) The total quantity of radioactive materials entering the general environment from the entire uranium fuel cycle, per gigawatt-year of electrical energy produced by the fuel cycle, contains less than 50,000 curies of krypton-85, 5 millicuries of iodine-129, and 0.5 millicuries combined of plutonium-239 and other alpha-emitting transuranic radionuclides with half-lives greater than one year.

§ 190.11 Variances for unusual operations.

The standards specified in § 190.10 may be exceeded if:

(a) The regulatory agency has granted a variance based upon its determination that a temporary and unusual operating condition exists and continued operation is in the public interest, and

(b) Information is promptly made a matter of public record delineating the nature of unusual operating conditions, the degree to which this operation is expected to result in levels in excess of the standards, the basis of the variance, and the schedule for achieving conformance with the standards.

§ 190.12 Effective date.

(a) The standards in § 190.10(a) shall be effective December 1, 1979, except that for doses arising from operations associated with the milling of uranium ore the effective date shall be December 1, 1980.

(b) The standards in § 190.10(b) shall be effective December 1, 1979, except that the standards for krypton-85 and iodine-129 shall be effective January 1, 1983, for any such radioactive materials generated by the fission process after these dates.

APPENDIX E

PRESSURIZED WATER REACTOR

To further illustrate the required sensitivity and other measurement considerations we also quote from "Radiological Surveillance Study at the Haddam Neck PWR Nuclear Power Station," Section 8, Summary and Conclusions (Reference 27). This was a study performed around the Haddam Neck reactor in Haddam Neck, Connecticut.

8.1 Radionuclides in Effluents from the Haddam Neck Station

Radionuclides were discharged by numerous pathways in small amounts relative to effluent limits. The largest constituents among radioactive effluents were ^3H , mostly in liquid waste, and ^{133}Xe , mostly in airborne waste. These observations appear to be generally applicable to large PWR nuclear power stations, except that less ^3H is discharged when the fuel is clad in Zircaloy instead of stainless steel. Lesser discharges of many radionuclides, including ^{133}Xe , have been predicted when additional waste treatment is

discharges reported by the station operator and the discharge estimates presented in the Environmental Statement for the station; any differences in individual values are discussed in Sections 3 and 4. The largest discrepancy concerns calculations based on a 'model' plant: the amounts of discharged gaseous radionuclides are similar to those predicted for 0.25 percent of fuel elements releasing radionuclides to the coolant, although this value was only 0.02 percent at the Haddam Neck station.

The estimated amounts of radionuclides in airborne effluents during the second half of 1970 and the first half of 1971 are as follows:

Radionuclide	Radionuclides in airborne effluents, Ci/yr						
	(1) Waste gas surge sphere	(2) Vapor container air	(3) Primary auxiliary bldg air	(4) Steam jet air ejector	(5) Fuel bldg air	(6) Primary vent stack	(7) Turbine bldg air
^3H	0.007	0.18	4.3	0.081	2.3	12.	150.
^{14}C	0.032	0.18	< 2.	0.051	0.3	< 3.	< 10.
$^{85\text{m}}\text{Kr}$	na	na	na	6.6	na	na	na
^{86}Kr	29.	77.	9.1	12.	0.8	26.	43.
^{87}Kr	na	na	na	10.	na	na	na
^{88}Kr	na	na	na	17.	na	na	na
$^{135\text{m}}\text{Xe}$	0.4	0.3	na	18.	na	na	na
^{133}Xe	130.	120.	650.	1,100.	< 12.	1,500.	< 400.
^{135}Xe	0.50	na	na	58.	na	na	na
^{131}I	0.0007	na	na	na	na	0.013	na
Long-lived particulate	0.0003	na	na	na	na	0.003	na

note: na = not analyzed

applied to meet 'as low as practicable' criteria for design objectives.

Results of the effluent measurements in this study are summarized below, based on the information in Sections 3 and 4. For simplicity, they are given as annual releases. Because these values were obtained by occasional sampling, they should be considered only indications of the magnitude of radionuclide discharges. Exact values must be derived from frequent or continuous measurements at the many waste streams or discharge locations. The totals (see Sections 3.3.13 and 4.3.3) are comparable to the annual

The values in data column 6 include the same wastes as those in columns 3, 4, and 5, and also discharges from the blowdown flash tank, which could not be measured separately. The ^3H and ^{14}C values are for all forms of the radionuclides, but distinctions between tritiated water and gases, and between ^{14}C in CO_2 and other gases, are made in Section 3 for several of the waste streams. A small amount of ^{41}Ar was also observed in one of the gaseous wastes. The presence of $^{83\text{m}}\text{Kr}$, ^{89}Kr , $^{131\text{m}}\text{Xe}$, $^{135\text{m}}\text{Xe}$, ^{137}Xe , and ^{138}Xe in curie amounts was inferred, although these radionuclides could not be measured. Short-lived progeny of noble

gases, such as ^{86}Rb , and relatively short-lived iodine isotopes, such as ^{133}I and ^{135}I , were also indicated to be present. Some of the listed values in ventilating air were noted to be uncertain because too few samples were collected. A few totals are incomplete because a contributing waste stream could not be sampled.

The tabulation suggests that the usual program of radioactive gas measurements at the air ejector and in waste tanks can account for a large fraction of the discharges. Discharged ventilating air, however, also carries radioactivity by a variety of pathways. The radionuclide compositions of the various effluents differ widely.

The estimated amounts of radionuclides in liquid effluents during the same period are as follows:

Radionuclides in liquid effluents, Ci/yr

Radionuclide	Reactor system	Secondary system waste	
	waste	blowdown	leakage
^3H	8×10^3	6×10^1	2×10^2
^{14}C	3×10^2	6×10^4	
^{24}Na	ND	2×10^2	
^{32}P	5×10^4	7×10^3	
^{51}Cr	3×10^3	2×10^3	
^{54}Mn	1×10^1	2×10^3	
^{55}Fe	4×10^1	4×10^2	
^{59}Fe	8×10^3	(6×10^4)	
^{57}Co	6×10^3	(2×10^4)	
^{58}Co	4×10^1	2×10^2	
^{60}Co	2×10^1	1×10^3	
^{63}Ni	6×10^3	(2×10^3)	
^{90}Sr	4×10^4	1×10^4	
^{90}Zr	5×10^4	2×10^4	
^{93}Nb	3×10^3	(2×10^4)	
^{99}Mo	$< 1 \times 10^3$	3×10^2	
^{110m}Ag	3×10^3	ND	
^{131}I	4	2	
^{132}I	1×10^1	2	
^{135}I	ND	8×10^1	
^{133}Xe	2×10^1	ND	
^{134}Cs	1×10^2	8×10^2	
^{136}Cs	$< 1 \times 10^3$	2×10^2	
^{137}Cs	2×10^2	8×10^2	

note: ND-not detected

Tritium was assumed to be the only radionuclide in water leaking from the secondary system, although no confirmatory measurements were made. The values in parentheses were inferred as described in Section 4.3.3. Unlisted radionuclides, such as ^{90}Sr , were not detected in any effluent samples; less-than values for some are given in Section 4.3.3.

The bulk of the effluent radioactivity was discharged from the reactor waste system. A few radionuclides, however, were in higher amounts in secondary system blowdown, which is discharged without storage or treatment.

8.2 Radionuclides in the Environment at the Haddam Neck Station

Radionuclides at low concentrations from the station were found in various media sampled in the aquatic environment:

- (1) The radionuclides ^3H and ^{131}I were in water at concentrations of approximately 10 pCi/ml and 1 pCi/liter, respectively. The radionuclides ^{58}Co and ^{134}Cs were at concentrations between 0.1 and 1 pCi/liter. Samples with these contents were obtained in the coolant water discharge canal and within a few kilometers of its mouth in the Connecticut River (see Sections 4.3.4 and 5.2).
- (2) Numerous radionuclides were in algae, plankton, and aquatic plants collected at the mouth of the canal and nearby in the Connecticut River. Iron-55 had the highest concentration in these samples, at 41 pCi/g wet weight (see Section 5.4).
- (3) Fish caught in the canal contained ^3H , ^{14}C , ^{131}I , ^{134}Cs , and ^{137}Cs . The highest concentration, of ^3H , was 2.9 pCi/g wet weight of tissue, compared to a background value of 0.6 pCi/g. One sample of shad, which swim up the Connecticut River for a brief period to spawn, showed a similar increase in ^3H concentration, but contained no other radionuclides attributed to the station (see Section 5.5). Oysters and clams from the mouth of the Connecticut River had no elevated levels of radionuclides (see Section 5.6). No shellfish were found in or near the coolant canal, although their presence had been reported.
- (4) Sediment from some locations that accumulate silt along the east bank of the Connecticut River at and just above and below the mouth of the canal contained ^{54}Mn , ^{57}Co , ^{58}Co , ^{60}Co , ^{134}Cs , and ^{137}Cs attributed to station effluents. The presence of ^{58}Co was usually most apparent; its highest concentration was 13.5 pCi/g (see Section 5.7).

Radioactive effluents and direct radiation from the station could not be readily detected in the terrestrial environment. No radionuclides attributed to the station were found in well water from just beyond the station boundary, in vegetables from nearby gardens, in milk from nearby dairy farms, or in the meat of deer killed near the station (see Sections 7.2, 7.3, 7.4, and 7.6). The following special measurements showed radionuclides or radiations from the station in the environment:

- (1) The thyroid of one dairy cow that had grazed on a hill 3.3 km distant from the station contained 21 pCi of ^{131}I (0.4 pCi/g thyroid) at the time when the ^{131}I discharge from the stack was approximately 10^3 $\mu\text{Ci/s}$ (see Section 7.5). A second thyroid, from a cow that had grazed at the same distance but at a lower elevation and during lesser ^{131}I discharges, contained no detectable ^{131}I (< 2 pCi).
- (2) While gas from the surge sphere was discharged for this purpose, the presence of ^{133}Xe in ground-level air was observed with a large, thin NaI(Tl) detector (FIDLER) with pulse height discrimination to count the characteristic 81-keV gamma rays. Measurements of ^{133}Xe at a stack discharge rate of approximately 2,000 $\mu\text{Ci/s}$ were possible on site. By collecting 0.5 m^3 of air, ^{133}Xe and ^{85}Kr were detected in ground level air at concentrations of 2×10^3 $\mu\text{Ci/m}^3$ and 4×10^3 $\mu\text{Ci/m}^3$, respectively. Indications of ^{133}Xe during releases from the waste gas surge sphere could occasionally be obtained off-site with the FIDLER survey instrument. Such measurements were not sufficiently sensitive to detect the much lower amounts of gaseous radioactivity discharged continuously from the stack (see Section 6).
- (3) Measurements with survey meters beyond the station boundary showed no observable increase over the background radiation exposure of approximately 8 $\mu\text{R/h}$ (70 mR/yr). Extrapolation of elevated radiation exposures within the boundary suggested that the highest exposure rate at nearby habitations was 0.3 $\mu\text{R/h}$ (2.6 mR/yr) due to direct radiation from the station. The exposure rate was estimated to be lower at the nearest population center, but higher on the Connecticut River, where persons would be exposed only briefly (see Section 7.7). The source of the radiation is believed to be stored radioactive waste.

On the basis of the observed effluent and on-site measurements, the highest population radiation doses were computed to be from consuming fish caught in and near the coolant-water discharge canal and from external radiation due to stored wastes and gaseous discharges:

- (1) fish consumption (Section 5.5.5) may have resulted in 3 mrem/yr to bone, 6 mrem/yr to thyroid, 0.4 mrem/yr to GI tract, and 0.3 mrem/yr to the total body;
 - (2) direct radiation (Section 7.7.4) may have resulted in 3 mrem/yr to the total body;
 - (3) airborne discharges (Section 3.3.14) may have resulted in 0.5 mrem/yr to the total body.
- The computations—particularly those of the dose from fish consumption—utilized several assumptions that require checking. The external radiation doses would be lower if adjusted for shielding and occupancy factor.

8.3 Monitoring Procedures

The following procedures were demonstrated in this and previous studies for monitoring effluents and environments of PWR stations:

- (1) measurement of effluent radionuclides other than the long-lived ones readily detected by a gamma-ray spectrometer; of particular interest, in addition to usually measured ^3H , ^{87}Sr , and ^{90}Sr , are ^{14}C , ^{32}P , and ^{59}Fe ;
- (2) measurement of ^3H and ^{14}C in their various gaseous species. Other recent studies suggest the inclusion of species measurements for radioiodine in air and radiocobalt in the aquatic environment;
- (3) surveillance of sediment with submersible gamma-ray detectors to indicate "hot spots" for detailed sampling and analysis;
- (4) use of concentration devices to collect ionic and insoluble radionuclides from water for measurements at concentrations of 10^{-10} $\mu\text{Ci/ml}$;
- (5) use of bovine thyroids to detect ^{131}I at very low concentrations (equivalent to 0.02 pCi/liter milk) in the terrestrial environment;
- (6) use of specialized survey instruments for detecting low levels of ^{133}Xe (400 pCi/ m^3) in ground-level air;
- (7) collection of fish that are under conditions of relatively restricted mobility in the environment to study their uptake of radionuclides.

8.4 Recommendations for Environmental Surveillance

The observations in this study support previously presented recommendations that assessment of population radiation exposures from routine facility operation be based on measuring the radioactive effluent and radiation flux at the station. Environmental radionuclide and radiation levels attributable to station operation were generally too

variable, obscured by the radiation background, or near instrumental detection limits to be measured precisely for evaluating exposure. Measurements at the source must include all significant pathways and radionuclides during the entire period of operation; critical radionuclides can be missed by monitoring only the obvious effluents and the easily measured radionuclides, or by ignoring the effects of changes in the operating cycle. Detailed studies of in-plant radionuclide pathways are needed for selecting an optimum program for sampling and analysis.

Environmental measurements were found to be useful for supporting and confirming the population radiation exposures computed from on-site monitoring, and for providing these computations with numerical factors applicable to the site. Such measurements, if performed reliably, can also be reassuring in demonstrating that no unexpected radioactivity is in the environment. For a station and site such as Haddam Neck, the following measurements provide useful information:

- (1) confirmation of critical pathways
 - a) measure critical radionuclides in fish caught in and near the coolant canal

- b) measure external radiation exposure rates on site and the decrease of the exposure rate as function of distance to off-site locations
- (2) determination of numerical factors for computing doses
 - a) compute X/Q values by measuring ^{133}Xe concentration in ground-level air relative to the release rate at the station
 - b) observe long-term accumulation of radionuclides in aquatic environment and possible transfer to the food chain
- (3) assurance that no significant exposure exists from unforeseen sources or occasional operational occurrences
 - a) measure radiation exposure at nearby habitations
 - b) measure radionuclides in milk, food, and drinking water obtained in immediate vicinity of station.

This program by EPA and cooperating groups will be concluded with a similar study at a commercially operated BWR nuclear power station. Generic radiological surveillance studies at other facilities in the nuclear fuel cycle are under consideration.

APPENDIX F

BOILING WATER REACTOR

In order to illustrate the required sensitivity and other measurement considerations we quote from "Radiological Surveillance Studies at the Oyster Creek BWR Nuclear Generating Station," Section F, Summary and Conclusions (Reference 10). This was a study conducted between 1971 and 1973 around the Oyster Creek reactor in Ocean County, New Jersey.

7.1 Radionuclides in Effluents from the Oyster Creek Station

Radionuclides were discharged by numerous pathways in small amounts relative to effluent limits. The most abundant constituents among radioactive effluents were ^3H , 61 percent in liquid waste, and the radioactive noble gases, mostly in airborne wastes. Also in the liquid wastes, the activation products ^{51}Cr , ^{54}Mn , ^{55}Fe , ^{60}Co and ^{134}Cs and fission-produced ^{131}I , ^{133}Xe , ^{135}Xe and ^{137}Cs were discharged in relatively large quantities. These observations appear to be generally applicable to large BWR nuclear power stations.

Results of effluent measurements in this study are summarized below, based on the information in Sections 3 and 4. For simplicity, they are given as annual releases. Because these values were obtained by occasional sampling, they should be considered only indications of the magnitude of radionuclide discharges. Exact values must be derived from frequent or continuous measurements at each discharge location.

The estimated amounts of radionuclides in airborne effluents during the second half of 1971 through the first half of 1973 are as follows:

Radionuclide	Radionuclides in airborne effluents, Ci/yr				
	(1) Main condenser steam jet air ejector	(2) Turbine gland seal condenser	(3) Building ventilation air	(4) Reactor drywell	(5) Stack
^3H	5.0×10^{-1}	$< 2 \times 10^{-3}$	2.7×10^1	8×10^{-4}	2.6×10^1
^{13}N	5.0×10^{-1}	5×10^2	NA†	NA	NA
^{14}C	3.0	5×10^{-3}	1.2	9.6×10^{-4}	9.1
$^{83\text{m}}\text{Kr}$	3.1×10^4 *	NA	NA	NA	NA
$^{85\text{m}}\text{Kr}$	6.9×10^4	8.2×10^1	NA	NA	NA
^{85}Kr	1.1×10^2	1.9×10^{-1}	2.0	2.8×10^{-2}	1.7×10^2
^{87}Kr	1.3×10^5	2.9×10^2	NA	NA	NA
^{88}Kr	1.4×10^5	1.4×10^2	NA	NA	NA
^{89}Kr	0*	6.2×10^2	$2.1 \times 10^{2*}$	NA	NA
^{131}I	1.7	NA	$5 \times 10^{-1*}$	NA	1.7×10^1
$^{131\text{m}}\text{Xe}$	$3.7 \times 10^{2*}$	NA	NA	NA	NA
$^{133\text{m}}\text{Xe}$	5.1×10^3	NA	NA	1×10^{-1}	7×10^3
^{133}Xe	1.6×10^5	2.1×10^2	1.0×10^3	2.2	1.2×10^5
$^{135\text{m}}\text{Xe}$	8.8×10^4	1.2×10^3	NA	NA	NA
^{135}Xe	3.0×10^5	4.7×10^2	4.0×10^3	NA	3.5×10^5
^{137}Xe	2.2	$1.1 \times 10^{3*}$	$3.6 \times 10^{2*}$	NA	NA
^{138}Xe	6.0×10^4	2.0×10^3	NA	NA	NA
Long-lived particulate**	NA	NA	NA	NA	5.0×10^1

* Calculated value, radionuclide not measured.

**Excluding particulate ^{131}I .

† NA - not analyzed.

The values for stack discharge in data column 5 reflect radioactivity from individual waste pathways, columns 1 through 4. Effluent radioactivity from reactor startup, not included since it was not measured, is expected to be a minor contributor. The ^3H and ^{14}C values are for all forms of the radionuclides; distinctions between tritiated water and gases and between ^{14}C in CO_2 and other gases are made in Section 3 for most pathways. The amounts of radionuclides in some waste streams were inferred when their contributions were expected to be significant. Short-lived progeny of noble gases, such as ^{88}Rb and ^{138}Cs , and relatively short-lived iodine isotopes, such as ^{132}I , ^{133}I , ^{134}I and ^{135}I , were also expected to be present.

Most stack radioactivity resulted from the air ejectors on the main condensers. Ventilation air contributed most of the ^3H effluent. Much of the ^{13}N and short-lived noble gases in stack discharge came from the turbine gland seal condenser.

Airborne effluents are expected to yield a total-body dose of 2.3 mrem/year to an adult residing where the highest annual average concentration occurs (see Section 3.3.10). The closest resident is estimated to receive 0.39 mrem/year, and a member of the closest population group, 2.1 mrem/year. (Actual dose would be lower since residential shielding and occupancy factors were not considered.) Dose to persons fishing in the discharge canal 700 hrs per year is expected to be about 0.1 mrem/year.

The estimated amounts of radionuclides in liquid effluents during the period from August 1971 to November 1973 are as follows:

Radionuclides in liquid effluents, Ci/yr		
Radionuclide	Waste sample tank	Laundry drain tank
^3H	4×10^1	1×10^{-1}
^{14}C	8×10^{-3}	1×10^{-4}
^{32}P	6×10^{-2}	2×10^{-4}
^{51}Cr	5×10^{-1}	3×10^{-3}
^{54}Mn	4×10^{-1}	2×10^{-2}
^{55}Fe	6×10^{-1}	4×10^{-3}
^{59}Fe	7×10^{-2}	4×10^{-3}
^{58}Co	5×10^{-2}	4×10^{-3}
^{60}Co	9×10^{-1}	5×10^{-2}
^{64}Cu	1×10^{-2}	ND
^{65}Zn	5×10^{-3}	ND
^{76}As	5×10^{-2}	ND
^{89}Sr	1×10^{-2}	3×10^{-4}
^{90}Sr	1×10^{-3}	3×10^{-5}
^{92}Zr	2×10^{-2}	1×10^{-3}
^{93}Nb	2×10^{-2}	2×10^{-3}
^{99}Mo	2×10^{-1}	ND
^{103}Ru	1×10^{-2}	2×10^{-4}

^{105}Rh	5×10^{-2}	ND
^{110m}Ag	2×10^{-3}	ND
^{124}Sb	9×10^{-3}	7×10^{-4}
^{131}I	1×10^{-1}	5×10^{-3}
^{133}I	5×10^{-2}	ND
^{133}Xe	9×10^{-1}	ND
^{135}Xe	1	ND
^{134}Cs	2	2×10^{-3}
^{137}Cs	4	4×10^{-3}
^{140}Ba	3×10^{-2}	9×10^{-4}
^{141}Ce	4×10^{-2}	4×10^{-4}
^{144}Ce	3×10^{-2}	9×10^{-4}
^{239}Np	3×10^{-2}	ND

Note: ND - not detected

The bulk of the liquid effluent radioactivity was discharged from the waste sample tanks after treatment and storage. Only a small quantity, generally less than 5 percent, was discharged directly from the laundry drain tanks to the discharge canal. For the 17 radionuclides which could be compared with annual discharges reported by the station operator, agreement was reasonable except for the relatively low measured quantities of ^{239}Np , ^{89}Sr and ^{90}Sr . Further evidence of agreement was derived from the ability to predict radionuclide concentrations in the circulating coolant canal during discharge from pre-discharge measurements of the sample test tank contents and appropriate dilution factors (see Section 4.4).

The results obtained in this study reflect the operations and conditions at the station during the study period, October 1971 to November 1973. According to the station operator, the replacement of original fuel and improved fuel cladding performance since the study period has significantly reduced off-gas release rates. Further reduction of radioactivity in the off-gas from the steam condenser air ejectors will be realized when the plant is fitted with an extended radwaste treatment system. Radioactivity in liquid effluents have also been reported by the station operator to have decreased due to improvements in the radwaste treatment system.

7.2 Radionuclides in the Aquatic Environment at the Oyster Creek Station

Radionuclides from the station were found at low concentrations in various media sampled in the aquatic environment:

- (1) The following radionuclides discharged by the station were at concentrations greater than 1

pCi/liter in the coolant canal: ^{51}Cr , ^{54}Mn , ^{60}Co , ^{99}Mo , ^{131}I , ^{134}Cs and ^{137}Cs . In addition, ^{59}Co , ^{59}Fe , ^{90}Zr , ^{95}Nb , ^{141}Ce and ^{144}Ce were detected at concentrations between 0.1 and 1.0 pCi/liter. Concentrations of ^{90}Sr in water from Barnegat Bay and the intake and discharge canals were generally near background levels. Manganese-54 and ^{60}Co were measured at levels up to 2.2 and 4.0 pCi/liter, respectively, in large water samples (76–380 liters) collected from the canals and bay. They were associated mostly with suspended material. Predicted radionuclide concentrations in the coolant canal during discharge agreed usually with measurements (see Section 5.2).

- (2) Radionuclides found in station effluents were observed in macro-algae and aquatic grasses collected from all sites in the bay and canals. The predominant radionuclides were ^{54}Mn (0.2–26 pCi/kg) and ^{60}Co (0.2–45 pCi/kg). Some samples contained ^{51}Cr , ^{59}Co , ^{106}Ru , ^{134}Cs and ^{137}Cs in quantities slightly exceeding background concentrations. Highest concentrations were observed usually in *G. verrucosa*, followed by *U. lactuca* and *C. fragile*. Radionuclide concentrations varied significantly with season of the year, presumably resulting from variations in atmospheric fallout, plant growing periods, and time of sample collection relative to discharge. The concentrations of ^{54}Mn and ^{60}Co in algae reflected relative amounts discharged and indicated little uptake from sediment. Algae proved to be sensitive indicators for monitoring radionuclides when water concentrations were below detectable levels (see Section 5.3).
- (3) Manganese-54 (to 34 pCi/kg) and ^{60}Co (to 54 pCi/kg) were predominant in fish muscle. Their concentrations generally increased with greater station discharges and decreased with distance from the mouth of Oyster Creek. Small quantities of ^{134}Cs were detected in fish that also contained ^{137}Cs above background levels (see Section 5.4).
- (4) Similar concentrations of ^{60}Co were detected in shellfish muscle and fluid, ranging from 120–260 pCi/kg. Almost all ^{60}Co in fluid was associated with protein. Although not detected in clam muscle, the shells of clams from Barnegat Bay contained twice as much

^{90}Sr as those from the background area, 190 vs. 105 pCi/kg. Barnacles collected from both canals contained ^{54}Mn , ^{59}Co , ^{60}Co , ^{90}Sr and ^{137}Cs from the station, and, being fixed in position, they provide good indicators of station discharges. Differences in concentration in barnacles from the discharge and intake canals indicated that 10 to 15 percent of the station effluent is recirculated. The radionuclide of highest concentration in clams was naturally-occurring ^{210}Po , 230 to 500 pCi/kg muscle, which is not attributed to reactor operations. An average $^{210}\text{Po}/^{210}\text{Pb}$ activity ratio of 9 indicated clam food (algae and plankton) was the probable source of the ^{210}Po (see Section 5.5).

- (5) No effluent radionuclides were detected in crab muscle, gills, gut or stomach, although the exoskeletons of some from Barnegat Bay contained more ^{54}Mn and ^{90}Sr than background samples. Because the exoskeleton is not eaten and is periodically molted, little useful information can be obtained from these analyses (see Section 5.6).
- (6) In sediment, ^{60}Co was the most widely distributed radionuclide, ranging from 0.26 to 18.6 pCi/g in the discharge canal to less than detectable quantities at the extremities of Barnegat Bay. Some sediment contained ^{54}Mn , ^{134}Cs and ^{137}Cs in excess of background. The highest ^{60}Co concentrations occurred in sediment from the wide area of the discharge canal that consisted of clay minerals and organic matter of low density. Core samples indicated that ^{54}Mn and ^{60}Co were deposited to at least 6 cm, and possibly to 12 cm below the surface. The underwater probe proved to be useful for locating areas of radioactive buildup above 0.5 pCi ^{60}Co /g (see Section 5.7).

Except for a few elements, the ability to determine concentration factors (CF) was not possible at Oyster Creek. Water concentrations were generally undetectable and, except for barnacles, effluent concentrations were unuseable due to uncertainty in the amount of dilution occurring in Barnegat Bay with its complicated hydrology (see Section 5.1.1). In a few cases, however, it was possible to estimate CF's for this site when the concentrations were constant and measurable. In other cases, the magnitude of published CF's could be evaluated from measured sample concentrations and knowledge of station discharges. CF's derived from this study are:

Element or radionuclide	Algae†	Grasses	Fish		Clams	Whole barnacles
			muscle	whole		
Fe	5000	6200	700	1850	ND*	ND
Sr	0.9	1.7	0.5	4	ND	1600**
Ca	1.4	1.9	1.8	19	ND	ND
K	14	10	15	11	7	ND
¹³⁷ Cs	13	23	30	23	ND	100
⁵⁴ Mn	ND	ND	ND	ND	< 1000	800
⁶⁰ Co	ND	ND	ND	ND	600	1000

* ND - not determined.

**Based on ⁹⁰Sr.

† The average CF's for the three species of algae are given, but significant species differences were noted for K and ¹³⁷Cs.

Published CF's of 100 and 600 for ⁶⁰Co and ⁵⁴Mn, respectively, in fish do not satisfy the observed data. The CF for ⁵⁴Mn is probably too high and may be nearer the value for ⁶⁰Co. Also, the CF for ⁵⁴Mn in clam meat was shown to be < 1000 rather than the published value of 12,000. The difficulties associated with the utilization of concentration factors are discussed in detail in Sections 5.4.4, 5.4.5, 5.5.3 and 5.5.4.

The highest population radiation doses from liquid discharges were computed from the annual average coolant canal concentrations to be from consuming fish caught in and near the coolant-water discharge canal. Fish consumption may result in 6 mrem/yr to bone, 0.9 mrem/yr to the GI tract, 1 mrem/yr to thyroid and 0.3 mrem/yr to the total body. These doses, although much greater than those based on measured radionuclide concentrations, are less than 5 percent of the limit recommended by the Federal Radiation Council and are almost entirely due to ³²P, ¹³¹I and ¹³³I. These radionuclides were generally not determined with sufficient sensitivity or in sufficient samples of fish or clams to confirm the calculations and should, therefore, be measured (see Section 5.4.5). Naturally-occurring ²¹⁰Po is a major contributor to the total clam-ingestion dose, and, although its content in fish was not measured, a similar situation may apply to fish. It would be desirable to determine this radiation background dose from consuming seafood whenever the dose due to nuclear operations is evaluated.

7.3 Radionuclides in the Terrestrial Environment at the Oyster Creek Station

Gaseous radioactive effluents and direct radiation from the station were detected in the terrestrial

environment. No samples of milk or food were obtained since they are not produced near the station in significant quantities due to poor soil conditions. The following measurements of radionuclides or radiation from the station in the environment were made:

- (1) By collecting large volumes of air during routine stack discharge, ¹³³Xe was measured in ground-level air at concentrations ranging from 3×10^{-3} to 3×10^{-2} $\mu\text{Ci}/\text{m}^3$, and ⁸⁵Kr at 3×10^{-4} $\mu\text{Ci}/\text{m}^3$ (see Section 6.2.5). Other short-lived gases would have been detected if analysis was initiated promptly. The short-lived progeny of ⁸⁵Kr and ¹³³Xe were measured by drawing 133 m³ of air through particulate filters and immediately analyzing them. Particulate or gaseous ¹³¹I could not be detected during brief sampling periods although large volumes of air were sampled.
- (2) Measurements of radioactive gases from the stack were made near the station with a muscle-equivalent ionization chamber, a pressurized ionization chamber and portable NaI(Tl) survey meters. The plume was readily detectable above the background radiation level. Computed radiation exposure rates at a location 1.5 km from the stack were two to three times higher than the measured rates (see Section 6.2.6).
- (3) A muscle-equivalent ionization chamber mounted in a helicopter was used to measure radiation exposure rates in the plume of radioactive gases from the stack. This technique was shown to be useful in measuring the rise of the plume and its vertical and horizontal spreading. Exposure rates computed with models generally exceeded measured values (see Section 6.3).

- (4) Direct radiation from station buildings measured with survey meters and a muscle-equivalent ionization chamber along the station boundary ranged up to 1.8 $\mu\text{R/hr}$ (16 mR/yr) above the background radiation exposure of approximately 4.3 $\mu\text{R/hr}$ (38 mR/yr). Extrapolation of elevated radiation exposure rates within the boundary to distant sites gives a result comparing well with measurements. The exposure rate at the nearest residence is estimated to be 0.08 mR/yr. The annual population dose to persons driving along Rt. 9, the eastern site boundary, is estimated to be 0.034 man-rem (see Section 6.4).
- (5) Long-term radiation exposures in the station's environment were measured with thermoluminescent dosimeters. Measured levels above the natural radiation background were correlated with estimated exposures computed from a model (see Section 6.5).
- (7) measurement of ^3H and ^{14}C in several gaseous species;
- (8) use of portable 5- x 5-cm NaI(Tl) survey meters as sensitive detectors of the plume from the stack;
- (9) use of muscle-equivalent ionization chamber and pressurized ionization chamber for quantifying the radiation exposure rate during brief periods within or beneath the plume;
- (10) use of thermoluminescent dosimeters to quantify the average long-term radiation exposure rate from the plume;
- (11) use of pressurized ionization chamber, large 15- x 23-cm NaI(Tl) detector and spectrometer, muscle-equivalent ionization chamber with Shonka electrometer, and a portable 5- x 5-cm NaI(Tl) survey meter to measure direct radiation from the station;
- (12) use of helicopter to characterize plume shape and dispersion;
- (13) collection of large volumes of environmental air and applying separation techniques for measurement of ^{131}Xe and ^{85}Kr at very low concentrations, (applicable also to short-lived noble gases);
- (14) collection of ^{87}Rb and ^{138}Cs in environmental air on filters with high-volume samplers, and analysis by gamma-ray spectrometry; and
- (15) use of measured release rates at the station, meteorological data and transfer coefficients to estimate radionuclide concentrations in samples for comparison with measured or minimum detectable values.

7.4 Monitoring Procedures

The following procedures were demonstrated in this study for monitoring effluents and environments of BWR stations:

- (1) analysis by gamma-ray spectrometry with Ge(Li) detectors of multiple radionuclides in samples of primary coolant and effluent water before discharge and dilution, and in off-gas from reactor coolant and in various airborne waste pathways;
- (2) measurement of effluent radionuclides other than the long-lived ones readily detected by gamma-ray spectrometry; of particular interest, in addition to usually measured ^3H , ^{89}Sr and ^{90}Sr , are ^{14}C , ^{32}P and ^{55}Fe ;
- (3) collection of ionic and insoluble radionuclides in fresh and sea water by concentration from 400-liter volumes for measurements at concentrations as low as 10^{-10} $\mu\text{Ci/ml}$;
- (4) collection and analysis of food samples, including fish, clams and crabs;
- (5) collection and analysis of environmental media that serve as indicators, including aquatic grasses, algae, barnacles and sediment;
- (6) surveillance of sediment with submersible gamma-ray detectors to indicate "hot spots" for detailed sampling and analysis, and the superiority of sediment sampling by hand (diver) rather than by dredge;

In addition, the following procedures were demonstrated in a previous study for monitoring environments of BWR stations:

- (1) collection and analysis of drinking water and food samples, including vegetables, milk, rabbits and deer;
- (2) collection of radioiodine from 22.5-liter volumes of milk on anion-exchange resin, and analysis by gamma-ray spectrometry; and
- (3) use of bovine thyroids to detect ^{131}I at very low concentrations (equivalent to 0.02 pCi/liter milk) in the terrestrial environment.

7.5 Recommendations for Environmental Surveillance

The fundamental objective of an environmental surveillance program is to measure radiation dose rates

and radionuclide concentrations in the critical exposure pathways in order to determine radiation doses to individuals and selected population groups from operation of a nuclear facility and to determine or confirm compliance with applicable standards. Other objectives are to confirm estimated environmental concentrations based on effluent data, determine any accumulation of long-lived radionuclides in the plant environs, and respond to public concerns and inquiries.

The recommendation for radiological surveillance programs conducted by nuclear generating stations to meet the above objectives, based on observations in this study and those at Dresden I BWR, and the Yankee and Haddam Neck PWR's, is that all radioactive effluents be analyzed to obtain in detail their radionuclide content. Environmental radionuclides and radiation levels attributable to station operation are generally too variable, obscured by the radiation background, or near instrumental detection limits to be measured with sufficient accuracy for evaluating exposure. The measurements at the source must include all significant pathways and radionuclides during the entire period of operation; critical radionuclides can be missed by monitoring only the obvious effluents and, as in the case of ^{32}P , the easily measured radionuclides, or by ignoring the effects of changes in the operating cycle. After all radionuclides in the effluent have been quantified and all critical pathways identified, analyses can be limited to the radionuclides at highest abundance and of greatest health significance in environmental samples from the critical pathways. Additional samples for analysis should include only media that are known or observed to concentrate radionuclides discharged by the station. As knowledge of the environment increases and the pattern of radionuclide discharges is established, fewer measurements will be needed. However, significant changes in station operation or radionuclide content of effluents will require at least a brief return to more detailed analysis.

The environmental program must be evaluated periodically to consider modification in response to changes in effluent radioactivity, new patterns of population distribution and environmental use, and increased knowledge of the behavior of radionuclides in the environment.

Adhering to these recommendations will insure a radiological surveillance program that will provide the necessary information to satisfy the above objectives at a lower cost than many current programs that include non-pathway type samples or samples that continually contain either less than measurable quantities or concentrations indiscernible from the natural radiation

background. Also, the recommended program will generate on-site transfer coefficients and concentration factors providing a better and more pertinent basis for calculating exposures at the site from station effluent data than most published values.

Environmental measurements at the Oyster Creek Station were found to be useful in developing the environmental surveillance recommendations described above, for supporting and confirming the population radiation exposures computed from on-site monitoring, and for providing these computations with numerical factors applicable to the site. Such measurements, if performed reliably, can also be reassuring in demonstrating that no unexpected radioactivity is in the environment. For a station and site such as Oyster Creek, the following measurements provide useful information:

- (1) confirmation of critical pathways
 - a) measure inhalation and external radiation exposure rates from the plume at off-site locations,
 - b) measure direct radiation exposure rates on site and the decrease of the exposure rate with distance to off-site locations,
 - c) measure critical radionuclides in fish, clams and crabs caught in the intake and discharge canals and in Barnegat Bay;
- (2) determination of numerical factors for computing radiation doses
 - a) determine the soluble and insoluble radionuclide fractions in liquid effluents and in the discharge canal,
 - b) confirm or ascertain applicability of aquatic concentration factors,
 - c) compute X/Q values by measuring ^{133}Xe or other radionuclide concentrations in ground-level air relative to the release rate at the station;
- (3) utilization of environmental concentration loci
 - a) measure critical radionuclides in marine grasses, algae and barnacles to determine the extent of contamination in the aquatic environment;
- (4) assurance that no significant exposure exists from unforeseen sources or occasional operational occurrences
 - a) measure radiation exposure at nearby habitations, canal banks utilized by fishermen and beaches in the immediate area,
 - b) measure radionuclides in seafood and water collected from the immediate vicinity of the station,

- c) if agricultural practices change in the vicinity of the station, measure radionuclides in milk and food products.

7.6 Suggested Future Studies

The following studies at nuclear facilities are suggested on the basis of the previous four field studies:

- 1) develop more sensitive techniques for measuring radioiodines in various chemical forms in airborne waste pathways through the station and in environmental air;
- 2) develop techniques to measure gaseous radionuclides, such as ^{83m}Kr and ^{131m}Xe that emit only low-energy photons, while being part of a noble gas mixture;
- 3) examine the effect of radioactive waste treatment on discharge practices in order to evaluate the cost of reducing the radionuclide content of effluents;
- 4) characterize the physical-chemical states of radionuclides in liquid wastes discharged to the environment and determine changes occurring after mixing with environmental waters;
- 5) measure critical radionuclides in environmental samples that are difficult to analyze, such as ^{32}P in Oyster Creek fish, to confirm hypothetical concentrations based on station effluent analysis;
- 6) perform a radiological surveillance study at a high temperature gas-cooled reactor (HTGR) similar to previous studies except focus effort on the gaseous effluents and their impact on the environment;
- 7) perform a surveillance study at a multiple reactor site to determine modeling parameters for dual stack releases and the existence of a scaling factor — quantities discharged vs. power generation; and
- 8) perform studies to validate atmospheric dispersion models used in dose assessment at sites with different meteorological and topographical characteristics.

These studies will further determine the environmental impact of nuclear facilities and develop better environmental surveillance techniques.

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C. Nuclear Spent Fuel Processing

1. Introduction

Nuclear reactor fuel is routinely removed, after some period of use, from the reactor and replaced by fresh fuel. The "spent" fuel that is removed, however, still contains substantial amounts of useful fuel materials, including uranium, plutonium, and even thorium (should this come into use). Moreover, this fuel contains most of the radioactivity that was present at the nuclear power plant. Choosing what to do with this fuel directly addresses two related questions: how or whether to make use of the fissionable material remaining in the fuel; and what to do with the large amount of radioactivity contained therein.

As suggested in Fig. 1, fuel removed from nuclear power plants is presently being left in storage, most often at the power plant site. However, for light-water reactors (LWRs) it has been anticipated that this fuel would be reprocessed both to extract uranium and plutonium and to concentrate the remaining large amounts of radioactive waste into a form suitable for disposal. Recycling the uranium and plutonium into fresh fuel would reduce the demand on uranium resources by about 30%. Until recently, it was thought that the cost of reprocessing was more than offset by the savings resulting from reduced purchases from uranium suppliers. In addition, it appeared that consolidating the radioactive wastes into a selected type of matrix would result in a superior disposal form, from the point of view of volume and, more importantly, stability. Finally, experience with reprocessing and recycle would provide a natural precedent for the liquid-metal fast breeder reactor (LMFBR) fuel cycle, which requires that plutonium be recycled.

In recent years, increasing attention has been given both to the possibility of avoiding or deferring recycle (by storing or disposing of fuel directly; see Fig. 1) and to alternative forms of reprocessing. This expansion of options has resulted from diverse factors, including: the recognition that at current uranium prices recycle is economically marginal; increased interest in the environmental aspects of spent fuel reprocessing and/or disposal; and substantially decreased projections of nuclear growth. This last factor could reduce the depletion rate of uranium resources and thereby the need for a breeder, and could also make alternative recycle schemes more attractive. The implications of basing nuclear power on a commerce in plutonium, a material that may also be used in nuclear weapons, have also strongly affected the interest in alternatives to the plutonium recycle that was originally conceived.

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Fig. 1. Possibilities for spent fuel handling. Spent LWR fuel is now being placed in storage; reprocessing, recycle, and disposal are not yet occurring on a large scale.

U. S. commercial experience with spent fuel handling has included a large number of spent fuel storage sites, a single reprocessing plant, and no disposal facilities. That reprocessing plant, owned by Nuclear Fuel Services (NFS) and located at West Valley, New York, began operations in 1966 with a capacity of 1 MIHM (metric tonne of heavy metal, i.e., uranium and plutonium) per day. By 1972, this plant closed for improvements and expansion to 2 to 3 MIHM/day. However, a decision was subsequently made to close the plant permanently. This plant has provided the only commercial U. S. experience with reprocessing operations and the resulting effluents.

Two other plants have been constructed but not operated. The first was the General Electric Midwest Fuel Recovery Plant (MFRP) at Morris, Illinois. MFRP was scheduled to begin operation during 1973. However, after difficulties discovered during preoperational testing, General Electric chose not to operate the plant. So far, it has only been used for spent fuel storage.

A 5 MIHM/day plant, the Barnwell Nuclear Fuel Plant (BNFP), has been built at Barnwell, South Carolina by Allied-General Nuclear Services. Although the facilities for separating plutonium, uranium, and wastes have been completed, construction of facilities for solidifying wastes, for converting plutonium to the proper form, and for handling materials contaminated with transuranics has not begun, pending decisions on the recycle of plutonium and on criteria for waste forms, as well as on the licensing of reprocessing per se.

Thus further development of commercial reprocessing and related forms of fuel fabri-

cation and waste disposal must await a decision in favor of this direction for commercial nuclear power. In the meantime, the need for spent fuel storage capacity increases, if only on an interim basis, while the need to develop and choose a satisfactory means for final disposal of nuclear wastes continues.

The questions proposed above are being urgently considered, both in the United States and abroad. They have been set forth in great detail in a number of reports of government agencies. Plutonium recycle in LWRs is examined in Ref. 1, waste disposal in Ref. 2, and the LMFBR program in Ref. 3. However, these reports arose largely out of a milieu that presumed plutonium recycle and the introduction of breeders. More recent studies^{4-6*} by other entities have examined a broader range of possibilities, with special emphasis on the implications of reduced growth rates on alternative forms of nuclear power, and on the risks of nuclear weapons proliferation. Alternatives for spent fuel handling and storage are considered in Ref. 7. Reference 8 gives an elementary summary of various possibilities and attendant issues.

2. Spent Fuel Radioactivity

As indicated in the section on nuclear reactors, an LWR now discharges about 30 Mg (metric tons) of fuel annually, containing billions of curies of radioactivity. This radioactivity is a major concern, regardless of the ultimate form of the fuel cycle. The amounts discharged are extremely large and, as indicated in Fig. 2 (from Ref. 9), substantial substantial amounts persist over not just decades - the lifetime of a nuclear power plant - but millenia - the length of human history. However, the amount of radioactivity does decrease relatively rapidly after removal of the fuel from the reactor. To facilitate subsequent handling and processing (if any), it had long been presumed that LWR fuel would remain in at-reactor-storage for a cooling period of 150 days.

As was seen in Table 1 of Section II.B on nuclear reactors, most of the radioactivity present in the discharge fuel is fission products, and the bulk of the remainder is actinides, including uranium, plutonium, americium, and curium. The bulk of the inventory consists of radionuclides with very short half-lives. However, a sufficiently large portion have half-lives of years or more, so that large amounts of radioactivity persist. Of the predominant fission products, xenon and iodine decline very rapidly in amount (within months); the rare earths and the "Zr-Sb" class (Fig. 2) then dominate until a few years pass, after which ⁹⁰Sr and ¹³⁷Cs contribute the largest amount for the first few centuries (with ⁸⁵Kr and certain of the rare earths making some

contribution during this period). After the sharp decline in total radioactivity occurring in the first few centuries, a plateau is reached that lasts millenia. This arises from certain fission products, as well as from actinides such as ²³⁹Pu (24,000 year half-life). Note, however, that Fig. 2 only displays 0.5% of the plutonium, the amount that nominally would remain in the "high-level" wastes were reprocessing to occur.

Such considerations give a rough indication of the need for sequestering the radioactivity in the spent fuel. Note that some of the major fission product contributors to the total radioactivity are radionuclides of major concern for routine emissions (I, Kr, Xe) or accidental releases (Sr, Cs) from nuclear power plants. However, whether for emissions or for waste isolation considerations, amounts of radioactivity are not necessarily a direct measure of biological significance. Thus, for example, ³H and ¹²⁹I are not major contributors to the total radioactivity, nor is ¹⁴C even shown in Fig. 2. As a minimum, a hazard index must consider the relative biological significance of the various nuclides. The usual maximum permissible concentrations (MPCs) could serve in as part of such an index. This simple approach would not, however, give any weight to the probability that radionuclides considered would reach humans. As noted in Ref. 4, it is difficult to construct a completely satisfactory hazard index. It is interesting to note, though, that following a sharp decline that ends after about 1000 years, MPC-based ingestion hazard indices change relatively slowly with time - only about an order of magnitude over the period one thousand to one million years (Ref. 4, particularly Figs. 7B1 and 7B2). This is consistent with the form of the curves in Fig. 2.

Returning to curies amounts, we may note that, for periods approaching the length of mankind's history, large amounts of reactor-produced radioactivity need to be isolated from the biosphere. The annual feed to a uranium-fueled LWR contains only about 100 Ci, in the form of ²³⁸U and ²³⁵U. The annual discharge contains billions of curies and this decreases to less than 1000 Ci (exclusive of the plutonium) only after about 1000 years. (The 4×10^6 Ci annual plutonium discharge, if not recycled, decreases to about 2×10^4 Ci by 1000 years.) Thus from the environmental point of view several needs have to be considered: 1) short to medium term possibilities, whether spent fuel storage or reprocessing, that adequately handle the fuel for periods of years to decades; 2) medium term waste disposal that effectively isolates the great amount of radioactivity, largely fission products, that remains for hundreds of years; and 3) very long term isolation of radioactivity, both fission products and actinides, that remains after this period.

*Sometimes the reference numbers appear as superscripts.
9/79

Fig. 2. Radioactivity produced in year by a 1000 MWe Light Water Nuclear Power Plant (Ref. 9).

3. Spent Fuel, Reprocessing, and Waste Facilities

a. Spent fuel storage. Most spent fuel is presently stored at reactor sites. However, this capacity is being pressed and - because of the much slower than anticipated development of reprocessing and recycle - a clear need has arisen for larger storage facilities at which spent fuel could remain for more than a year or so, perhaps decades. Such facilities, which may be away from reactor sites, need not differ greatly from at-reactor-storage, which places spent fuel assemblies in water pools for cooling and shielding. However, for relatively long-term retrievable storage, methods for dry storage may be feasible, because of lower cooling/shielding requirements, and attractive, because of simpler maintenance. A description of pool storage may be found in environmental statements and safety analysis reports for nuclear power plants. Reference 7 discusses alternative for LWR spent fuel handling and storage.

b. Fuel reprocessing. The prime function of a reprocessing plant is to dissolve the spent fuel and separate the resulting solution into waste and fuel streams. Such separation has been performed for more than three decades, but usually in military facilities whose aim is to extract plutonium for use in weapons. The technique employed for light-water reactor fuel would be, after 5 or 6 months storage, to dissolve the fuel in nitric acid, then to strip out and purify uranium and plutonium. The remaining nitric acid solution, containing both fission products and activation products, is often referred to as "high level wastes". However, beyond this separation, a related function that would ordinarily be performed at a reprocessing complex is to convert both waste and product streams to forms that are suitable for their disposal or use. A functional diagram of a reprocessing plant that includes such conversions is given in Fig. 3.

For uranium the product would be UF_6 , the chemical form needed for feed to uranium enrichment plants. The appropriate product

Fig. 3. Simplified flow diagram of Reprocessing Plant Complex (Ref. 1).

for plutonium would be plutonium dioxide (PuO_2), the chemical form used in fresh fuel. In fact 10 CFR 71.42 (Ref. 10) requires that plutonium be shipped in a solid form, so that such conversion is effectively required at the reprocessing plant. However, sufficient uncertainties still exist that the builder of a reprocessing plant (such as that at Barnwell) does not have design criteria for this conversion. For example, to reduce the weapons usability of transported material, reprocessors may be required to mix the plutonium with a larger amount of uranium.

Similarly large uncertainties exist for the high level wastes. 10 CFR 50, Appendix F, specifies that high level liquid radioactive wastes be converted to solid form within 5 years of generation and transferred to a federal repository within 10 years. However, a form for disposal has not been specified, so that - even if reprocessing itself were approved - it could not proceed because design criteria for waste conversion facilities are not available. The possibility given most attention is vitrification, i.e., drying the wastes and incorporating them into a glassy material.

The processes for the separations facility have been summarized in Ref. 11 as follows: "...Typically, a fuel recovery process entails shearing the fuel (to rupture the corrosion-resistant sheath and expose the fuel), dissolution of the fuel in nitric acid, separation and purification of the uranium and plutonium by solvent extraction and ion exchange, and conversion of the product nitrates to oxides suitable for refabrication into fuel elements...

"The spent fuel is transported from the reactor to the reprocessing plant in heavy, shielded casks. The cask is unloaded in a water-filled pool, and the fuel is stored under water, which serves both as a transparent radiation shield and as a coolant. The fuel elements to be processed are transferred to a head-end cell and sheared into two-inch lengths to expose the inner core, which is then leached with nitric acid in batch-dissolving tanks. The leached hulls constitute a solid waste that is ultimately disposed of by land burial. The nitric acid solution of the fuel, containing the uranium, plutonium, and nearly all of the fission products, is the feed solution for the solvent extraction process.

"Solvent extraction processes exploit the wide difference in concentration distribution between two immiscible phases - the organic and the aqueous. Nearly all major fuel reprocessing facilities employ some form of the Purex process, which makes use

of the organic complexing compound, tributyl phosphate (TBP), in an inert hydrocarbon diluent. When this organic mixture is brought into countercurrent contact with the aqueous feed solution, the TBP extracts both the uranium and the plutonium into the organic phase, leaving the fission and corrosion products behind in the aqueous phase.

"...In addition to the primary process, there are many auxiliary operations: treatment of the solvent to provide for its reuse, recovery of nitric acid from the aqueous streams, management of the gaseous, liquid, and solid waste effluents, and the specialized techniques and equipment required for process control and personnel protection."

As suggested in this discussion, radioactive waste from a reprocessing plant would include, not only the high level waste that draws most attention, but the cladding hulls, a variety of low level waste streams (see Fig. 3), and various kinds of contaminated material. A final residue to be considered is the reprocessing equipment itself.

c. Waste disposal. An approach to waste disposal has not been chosen. Although considerable effort has been expended on this question, both on methods for solidifying or encapsulating wastes and on selection of disposal sites, a sufficiently detailed understanding of the requirements or consequences of long term disposal does not exist to specify criteria for waste form or site design. Moreover, most work has presumed that reprocessing would occur and that the material of concern is "high level waste" in the usual sense, i.e., radioactivity separated from the spent fuel. Reference 12 provides a useful summary of recent understanding of such high level waste management.

As typically conceived, such wastes would be incorporated into a glass-like matrix, encapsulated, then buried in salt domes, formations that testify to the lack of groundwater by their very existence and may be appropriate sites, if undisturbed by human activity. On the other hand, serious attention is being given, both in the U. S. and abroad, to the possibility of storage in granitic formations. (See, for example, Ref. 2 and 4). An artist's conception of a salt-based repository is given in Fig. 4 (from Ref. 12).

Relatively little attention has been given, until recently, to the possibility of direct disposal of spent fuel. In principal, such disposal could occur, after encapsulation of the fuel, at sites such as mentioned above. From the point of view of possible emissions, this approach would differ from that involving reprocessing in two major respects. First, the uranium dioxide pellets, encased in zircaloy cladding, is a substantially different

Fig. 4. Artist's conception of waste disposal site.

matrix for containing radioactivity than the glass-like substances that are the usual end product for solidifying liquid high level wastes. The effect of this difference on the risk from possible leaking or leaching of this radioactivity is not now understood. Secondly, the fuel would retain the plutonium that, in reprocessing, is recovered to be recycled. Again, the significance of this for the biosphere is not understood. These uncertainties arise from insufficient study either of the stability of various waste forms and disposal methods for times and conditions of interest or of the modes of radionuclide transport from sites of the type being considered. See Ref. 2 for further discussion of disposal possibilities.

The choice of waste disposal mode, whether spent fuel or reprocessing wastes, depends on more than purely "environmental" considerations. It also depends on the perceived value of reprocessing from the point of view of nuclear demand and uranium resources and on external considerations, such as the effect that reprocessing and recycle may have on the proliferation of nuclear weapons. In any case, implementation of either reprocessing or spent fuel disposal would probably be preceded by interim storage of the sort suggested above, either pool storage or newer dry methods. This question is being given the urgent attention of concerned parties; see, for example, Ref. 7.

d. Recycle fuel fabrication. The products of a reprocessing plant would be used to produce fresh fuel. Basically the same processes are involved as for production of ordinary uranium oxide fuel. The steps from oxides of uranium and plutonium to fuel rods are (Ref. 1):

- blending of feed powders (PuO_2 , UO_2 , MOX, or recycled scrap)
- pretreatment of mixture by comminution, compaction, and granulation
- pelletizing
- sintering of the pellets
- grinding of the pellets to finished dimensions
- cleaning and drying the pellets
- loading the pellets into fuel rods, decontaminating the rod ends, and welding the end caps
- decontamination of the rod, if necessary
- inspection of the rod for integrity.

These rods are then assembled into bundles for insertion into reactors.

Several mixed-oxide (MOX) fueled fabrication plants have operated in the United States (Table 1), although not all of these are now active. Some of these have used nitrate as the feed material although, as noted above, new regulations require shipment of plutonium in a solid (oxide) form.

4. Radioactivity Control Systems

The primary emissions from stored spent fuel are the relatively volatile fission products discussed in the section on nuclear reactors. For extended storage, the emission of greatest significance is ^{85}Kr . At power plants, this emission is not controlled (although more short-lived noble gases are reduced by holdup). Presumably ^{85}Kr would also be released from storage facilities, particularly since the release rate (Ref. 1) is estimated to be rather low, about 1 Ci per year per metric ton of stored fuel.

On the other hand, the reprocessing plant frees all the radioactivity of the fuel from its solid matrix. Strict measures must be taken to control this large amount of radioactivity, both to protect the operating personnel and to minimize releases to the environment. Because of the intense gamma radiation from the fuel material, chemical processing equipment is heavily shielded, and relies on remote control and maintenance. Moreover, process and waste streams are carefully confined and treated. Figure 5 shows the effluent control systems associated with the process flow shown in Fig. 3. The principle means of control are summarized by the following excerpts from Ref. 11:

Table 1. Existing MOX Fuel Fabrication Plants (Ref. 1).

Licensee	Plant Location	Feed Material	Plant Product	Pu Possession limit (kg)	Est. production capacity (MI/yr)
Babcock & Wilcox	Parks Township, Pa.	Nitrate solution or PuO ₂ , U nitrate or (U and Pu) O ₂ UO ₂ + PuO ₂	(U,Pu)O ₂ fuel rods		20
Exxon Nuclear	Richland, Wash.	UO ₂ + PuO ₂	(U,Pu)O ₂ fuel assemblies	10 unencapsulated; 100 total	15
General Electric	Pleasanton, Calif.	Nitrate solution (U and Pu)	(U,Pu)O ₂ fuel rods	150	3
Kerr-McGee	Crescent, Okla.	Nitrate solution (U and Pu)	(U,Pu)O ₂ fuel rods	360	5-10
Westinghouse	Cheswick, Pa.	Pu nitrate solution PuO ₂ , U nitrate, or UO ₂	(U,Pu)O ₂ fuel rods	120	10-15

Fig. 5. Generic Reprocessing Plant Effluent Systems (Ref. 1).

"Radioactive gaseous wastes from these operations are treated chemically, as well as by filtration, sorption, and scrubbing in order to reduce their radioisotope content to levels that can be discharged to the atmosphere. The aqueous radioactive wastes that contain essentially all the fission products are generally concentrated by evaporation and stored on an interim basis in underground tanks. The evaporator overheads of radioisotopes to permit their discharge to the environment under existing regulations.

"Processing plants are designed to ensure containment of airborne radioactivity by providing increasing levels of vacuum in three successive envelopes so that all air leakage flows from areas of low to those of high contamination potential.

"Off-Gas Treatment. Because of its higher radioactivity and chemical fumes content, the dissolver off-gas in most radiochemical plants is treated, in turn, for nitric acid recovery, for iodine removal, and for removal of residual acid fumes before being blended with the off-gas from the vessels in the balance of the plant. The vessel off-gas is usually scrubbed with caustic, dried, and filtered through one

roughing and two high-efficiency filters (HEPA, asbestos-glass fiber paper, 99.97% DOP efficiency).

"Iodine in most of its chemical states is removed from gas streams by reaction with AgNO_3 impregnated on ceramic packing and by scrubbing with $\text{Hg}(\text{NO}_3)_2\text{-HNO}_3$ or caustic solutions. However, organic iodides, particularly methyl iodide, can be removed most efficiently by catalytic decomposition and sorption on silver, copper, or iodine-impregnated charcoal. The efficiency of iodine removal units is sharply dependent upon the concentration of the iodine, but 99.5% is a commonly quoted design efficiency in cases where organic iodides are not removed. The efficiency of charcoal impregnated with potassium iodide has been quoted at 99.99%.

"All off-gas streams from the plant are blended with the cell ventilation streams and passed through a sand filter, a deep-bed fiberglass filter, or a bank of high-efficiency particulate air filters before being monitored and discharged up a stack.

"Outgassing of Volatile Fission Products. One of the most difficult problems in reprocessing is that of containment of the volatile

fission products, especially ^{131}I . The volatile fission products (iodine, ruthenium, tellurium, cesium, tritium, krypton, and xenon) have been found to volatilize from oxide fuel at moderate temperatures (450 to 750°C)."

"Although the noble gases are chemically inert, they can be concentrated by cryogenic distillation, absorption in fluorocarbons, or diffusion through permselective membrane. The concentrated noble gases could then be compressed and stored, or immobilized with foam (plastic, glass, or metal), assuming that this method can be shown to be practical.

"Cell and Vessel Off-Gas. The cell-ventilation and vessel off-gas systems are primary sources of routine and accidental releases of radioactivity. The recycle of gas from both systems is feasible and will minimize the volume of off-gas needing routine treatment. Recycle will probably be economical, and the use of an inert cell atmosphere may become practical. This would practically eliminate the possibility of solvent fires in process cells.

"Liquid and Solid Wastes. It can be anticipated that all radioactive liquid wastes from fuel reprocessing will be evaporated and blended to yield only two streams: a high-level waste consisting of a highly concentrated solution of fission products and actinides, and a low-level aqueous waste that has been sufficiently decontaminated of radioisotopes to permit it to be either discharged to the environment or recycled to the process. The conventional practice of accumulating aqueous solutions of fission products in underground storage tanks will probably be curtailed because of the difficulty in providing safe containment of the more concentrated solutions derived from reprocessing highly exposed power-reactor fuels. Adequately engineered storage systems may be so complex and expensive, as compared with the cost of early solidification of the wastes, that liquid storage systems will be limited in capacity merely to those surge volumes needed for plant flexibility. Under these circumstances, liquid waste inventories in storage would represent, at most, only a few weeks (or months) of plant operations."

Table 2 (Ref. 1) summarizes the major waste systems for the three plants that have existed: NFS, MFRP, and BNFP. Note that, although the NFS plant permitted both gaseous and liquid releases, the more recent plants are designed to release only volatile radionuclides, principally tritium, noble gases, and ^{14}C . As discussed below, the amounts of tritium and ^{85}Kr released are much greater than that emitted at nuclear power plants. Reference 1 has a brief summary of possible improvements in control technology, based on a more complete treatment given in Ref. 13.

The Environmental Protection Agency has considered such technology in detail (Ref. 14) for use in setting standards (see below). An American Physical Society study group has recommended⁴ that technology be developed for sequestering ^3H , ^{14}C , and ^{85}Kr in a stable form. Should a breeder be introduced, improved iodine removal would be required if - as has been anticipated - the fuel were reprocessed only a short time (less than 5 months) after discharge.

Disposal sites for spent fuel or solid high-level waste from reprocessing have not, of course, been operated. The effluent control is expected to consist of HEPA filters to remove radioactive particulates. The principal routine effluent would be very small amounts of tritium and ^{85}Kr .

The operations performed at the fuel fabrication plant (See Section 3d) are mechanical and leave the plutonium in dioxide (powder or ceramic) form. Workers in the plant are protected by confining these operations to lightly-shielded sealed (glove) boxes. Emissions are controlled by carefully designed HEPA filtration systems.

5. Radioactive Emission, Exposures, and Regulations

Of the spent fuel handling facilities discussed above, the reprocessing plant emits by far the largest amount of radioactivity, an amount that, per gigawatt-year, is far greater than power plant emissions. Essentially all of the tritium, ^{85}Kr , and ^{14}C contained in the fuel is released. As discussed in Ref. 41, "The gases are released in the early stages of reprocessing, usually during chopping or dissolution. ^{85}Kr is released in the chopping and dissolving stages and goes into the off-gas system. Under current practice, some of the tritium is released from fuel or cladding during dissolution and goes into solution as HTO with only a small fraction (1 to 10%) going into the off-gas system. The remainder of ^3H is retained by the Zircaloy hulls. ^{129}I exists in a wide variety of chemical forms within the fuel and accordingly is released in many forms, but usually as elemental iodine or organic iodides. It has been found largely in the off-gas, but significant amounts can enter a liquid effluent stream if one exists. ^{14}C is produced as an activation product from the action of neutrons on the ^{17}O naturally occurring as a constituent of UO_2 , and on ^{13}C and ^{14}N impurities. Its chemical form in the fuel or cladding is unknown, although it is believed that its release to off-gas would be as CO or CO_2 ".

In Table 3, from Ref. 1, the amounts of tritium, ^{85}Kr , and ^{14}C that could be released without the incorporation of additional controls are estimated. Decontamination factors are presumed for the other radionuclides and

Table 2. Fuel Reprocessing Plants--Major Waste Systems¹

Plant	Gaseous Effluents	Liquid Wastes		Solid Wastes
		High Level	Other than High Level	
NFS	To atmosphere via iodine scrubber, HEPA filters, 61 meter stack	Stored onsite as neutralized liquid in tanks in vault*	Controlled release to creek	Stored onsite in drums in impervious clay deposits
MFRP	To atmosphere via iodine scrubber, sand filter, 91 meter stack	Converted to solid. Stored onsite in containers under water in storage basin.**	Stored onsite in tanks in vault as immobile salt cake	Stored onsite in underground lined vault
BNFP	To atmosphere via iodine scrubber, silver zeolite adsorber, HEPA filters, 100 meter stack	Stored onsite as acidic liquid in tanks in vault***	Stored onsite in tanks in vault	Stored onsite in concrete or steel drums

Note: Information in this table has been compiled from the safety analysis reports for these facilities.

* Plans to add tanks for storage of high level radioactive liquids in acidic conditions for ultimate conversion to solids and transfer to a Federal storage facility.

** Prior to shipment to Federal storage facility.

*** Plans for solidification of high level radioactive liquid wastes for transfer to Federal storage facility.

¹Table reproduced from Ref. 1

these depend strongly on the control technology (cf. Ref. 13 and 14). The amounts cited are released in the gaseous effluent system. Current designs incorporate radwaste systems that, rather than making controlled releases to liquid effluent streams, retain and recycle liquid and solid radioactive materials, ultimately for shipment to waste sites. However, as noted in Table 2 and discussed below, older plants (such as the NFS facility) had both gaseous and liquid radioactive effluents. For the radionuclides that are not controlled, such as tritium or ⁸⁵Kr, one would expect estimated releases such as those given in Table 3 to correspond closely to experience at operating reprocessing plants.

Because of their transport properties in the environment, the releases of ³H, ¹⁴C, and ⁸⁵Kr are relatively quickly dispersed generally throughout the earth's atmosphere. Thus they can result in world-wide exposure commitments. Reference 11 has made estimates of the worldwide buildup of tritium and ⁸⁵Kr (Fig. 6), assuming a rapid growth of nuclear power and associated reprocessing activities.

However, because of its much longer half-life, the relatively small Curie amounts of ¹⁴C released can eventually contribute a larger dose commitment.⁴

Table 3 also gives estimates of (whole body) population dose commitments associated with a year's operation of the model reprocessing plant of Ref. 1. Reference 1 also estimated the annual dose to the individual living nearest the plant (assumed distance: 1300 meters) to be about 8 mrem/yr (whole body). (See Tables IV E-9 to E-13 in Ref. 1 for details). Finally, Ref. 4 estimated the average dose rate as a function of time of operation of a standard reprocessing plant, given in Fig. 7.

Although no reprocessing plants or large-scale mixed oxide fabrication plants are now operating, some regulations applicable to their emissions or resulting exposures have already been formulated. The NRC review procedure for fuel reprocessing and fabrication plants utilizes Section 3 of the Regulatory Guides, including 3.10, "Liquid Waste Treat-

Table 3. Estimates of Radioactive Releases from Reprocessing
Ci/yr from 2000 MTHM/yr Model Plant (Ref. 1)

	UO ₂ Fuel	MOX + UO ₂ Fuel
Gases (including entrained matter)		
³ H	1,030,000	1,040,000
¹⁴ C	1,360	1,300
⁸⁵ Kr	22,800,000	21,900,000
⁹⁰ Sr	0.32	0.31
¹⁰⁶ Ru	8.13	8.77
¹²⁹ I	1.91	1.98
¹³¹ I	7.5	47.8
Other fission products	10	10
U	2.23×10^{-3}	2.69×10^{-3}
Pu α	0.05	0.07
Pu β	1.28	1.76
²⁴¹ Am	8.13×10^{-4}	1.33×10^{-3}
²⁴³ Am	6.44×10^{-5}	1.54×10^{-4}
²⁴² Cm	8.49×10^{-2}	1.66×10^{-1}
²⁴⁴ Cm	6.11×10^{-3}	2.27×10^{-2}
Population Dose Commitments (person-rem)		
Occupational	1,250	1,350
U.S. Population	18,800	18,600
Foreign Population	12,200	11,700
Total	32,250	31,650

ment System Design Guide for Plutonium Processing and Fuel Fabrication Plants", 3.12, "General Design Guide for Ventilation Systems of Plutonium Processing and Fuel Fabrication Plants", 3.18, "Confinement Barriers and Systems for Fuel Reprocessing Plants", and 3.32, "General Design Guide for Ventilation Systems for Fuel Reprocessing Plants". Guides specifically applicable to monitoring are discussed in the next section.

The EPA standard for the uranium-fueled nuclear fuel cycle, 40 CFR 190, sets limits for emissions and (general public) exposures, exclusive of mining, but including processing. It is evident that this standard, with its limit of 50,000 Ci/GWe-yr on ⁸⁵Kr emissions, presumes noble gas control equipment at the reprocessing plant. Moreover, the ¹²⁹I limit of 5 mCi/GWe-yr will require an order of magnitude improvement in decontamination (c.f. Table 3). The basis for this standard is discussed in Ref. 15.

In addition to standards that apply to particular facility types, more generally applicable standards, such as 10 CFR 20 (Standards for Protection Against Radiation) and Regulatory Guide Section 8 (occupational

exposures), must be considered in the design and operation of fuel handling facilities. See "Introduction" and "Nuclear Reactors" sections for discussion.

We note in closing that radioactive emissions from the fuel cycle depend significantly on whether or not fuel is reprocessed. For example, in an LWR system without reprocessing, most of the tritium, ¹⁴C, and ⁸⁵Kr releases from the fuel cycle are avoided. On the other hand, the ²²²Rn and other emissions associated with mining and milling are increased by about 30%, as are related occupational exposures. In a fast breeder system, mining and milling is virtually eliminated, but the volatile fission products are released at reprocessing; a particular problem for fuel that is reprocessed soon after removal from the reactor is the high concentrations of ¹³¹I, with its 8-day half-life.

6. Monitoring Techniques

The capability to monitor radiation fields and radionuclide emissions at fuel handling facilities is important, both for the protection of workers at these facilities, and for evaluating impacts on the environment and,

Fig. 6. Estimated production of ^{85}Kr and ^3H (Ref. 11).

Fig. 7. Global whole body dose rate increment due to T years of operation of a reference (1500 Mg/y) LWR reprocessing plant with no engineered retention of effluents (from Hebel et al., Ref. 4, with permission by American Institute of Physics © 1978).

ultimately, on members of the general public. The most immediate goal is to assure compliance of the facility with the applicable limitations on exposures or emissions, discussed above. Monitoring requirements at the facility are therefore oriented toward providing a long-term record, as well as a short-term warning of abnormal operation leading to unusually large exposures or releases. Monitoring the environment provides a direct record of the increases in radiation levels or radionuclide concentrations in the vicinity of those facilities.

The basic requirements for monitoring at fuel handling facilities are not dissimilar from those at nuclear power plants. This is particularly true of occupational exposure monitoring, since dose limitations are the same at all nuclear facilities and, in the cases of concern, exposures arise from external radiation. In general the emissions will be found to be significantly greater at reprocessing plants, where larger quantities of radioactivity are present and could be released (under different conditions) than at power plants. Similarly, a plant fabricating plutonium-bearing fuel has a significantly greater potential for transuranic releases than a power plant. However, this is a relatively narrow problem, because of the relative uniformity of material at a fabrication plant, which can be controlled by reliance on particulate filters and monitored by alpha detection systems (see Plutonium section). Because of the greater likelihood of emissions this section emphasizes effluent and environmental monitoring at fuel reprocessing plants.

The principal guidance for fuel reprocessing plant monitoring is NRC Regulatory Guide RG 4.16, "Measuring, Evaluation, and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Airborne Effluents from Nuclear Fuel Reprocessing and Fabrication Plants". This general guide, in turn, depends on other documents (including publications of industry standards) for detailed information on monitoring program design and measurement requirements. Consideration also has to be given to the emission and dose limitations cited above.

Some guidance on monitoring emissions from, and environmental concentrations around, reprocessing plants is available from the three commercial plants that have been built, as well as from many of the facilities that have been operated in the weapons program. However, it is important to note that future commercial reprocessing facilities would be operated in such a way that radionuclides are emitted only in the gaseous effluent stream (much as at the Barnwell facility). Furthermore, judging from present EPA regulations, controls techniques would have to be applied to radionuclides (such as ^{85}Kr) that had not been controlled in earlier facilities. It is

also useful to keep in mind that, although stack monitoring techniques for reprocessing plants may have general similarities to those for other facilities, including power plants, a significant difference is introduced by the fact that the gaseous effluent may be relatively corrosive because of the nature of the basic operation at a reprocessing plant which involves the dissolution of the spent fuel in nitric acid.

The discussion that follows emphasizes stack monitoring and environmental measurements. It also summarizes the effluent monitoring systems at NFS, MFRP, and BNFP, as well as the results of environmental monitoring at the first.

a. Airborne Emissions. Table 3 indicated the expected emissions, all via the gaseous effluent, from a large reprocessing plant. The only large plant now existing in the United States is the Barnwell facility, and we will indicate its stack monitoring capabilities, although the plant has not been put into operation. All continuous stack measurements at Barnwell are performed on the stream from a single sampling nozzle, as indicated in Fig. 7 (from Ref. 16). Because the

only commercial plant to have operated in the United States is the NFS Facility, we will also summarize the results of environmental monitoring in its vicinity; the NFS facility released both gaseous and liquid radioactive effluents.

Krypton-85. For a reprocessing plant that does not control noble gases, the ^{85}Kr monitor may be regarded to be a process monitor, since the amount released may be predicted solely on the basis of the fuel history. Hence a simple sodium-iodide based monitor such as indicated in Fig. 8 merely provides a check on the plant operation by measuring a weak gamma ray of the large ^{85}Kr flux. On the other hand, for facilities that control ^{85}Kr , more sensitive monitors would be required. See Section IV for ^{85}Kr instrumentation.

Studies (Ref. 17) by the Northeastern Radiological Health Laboratory (NERHL), part of the U. S. Bureau of Radiological Health, confirm that virtually all of the ^{85}Kr for a particular reprocessing cycle was emitted directly from the stack at NFS. At BNFP, conservative calculations yield 6 mrem/year potential skin exposure at the site boundary (Ref. 18). Similar doses have been estimated

Fig. 8. BNFP Main Stack Monitor (Ref. 16).

on the basis of the NFS emission data (Ref. 19).

Tritium. For plants such as BNFP, all the tritium from the fuel is released via the stack, primarily as water vapor. At BNFP, tritium may be monitored off-line by drawing a sample of liquid condensate from the dehumidifier (Fig. 8), from which water can be separated for laboratory scintillation counting. (See Section IV for a discussion of tritium monitoring). However, the tritium emissions can also be estimated from fuel history, since it is not controlled. At NFS, most of the tritium was relegated to liquid wastes (Ref. 17), discussed below.

Iodine-129 and -131. Because of the high efficiency of iodine scrubbed/absorbed in the off-gas system, only a small fraction of these isotopes will normally be emitted from the plant. Because of the large amounts of iodine available, compared with applicable standard, it is important that the iodine output be monitored to verify the continued effectiveness of the waste treatment system. The monitor at Barnwell consists of silver zeolite absorber, on which buildup of ^{131}I is observed by a sodium iodide detector. See Section IV for a discussion of iodine monitoring, both stack and environmental.

For spent fuel that has not been stored for a long period, the short-term iodine activity released at reprocessing is predominantly ^{131}I . This isotope could cause particular difficulty in a breeder system since the fuel may be reprocessed within a few months of discharge. For the longer term, ^{129}I is the greater problem, persisting because of its very long half-life - either in the environment or in the residue from the iodine control system. Particular attention has been given to environmental concentrations of ^{129}I in the vicinity of the NFS facility (Ref. 20), where however the iodine control system was neither highly efficient nor consistently operated.

Particulates. Airborne radioactive particles are usually removed from the gaseous effluents by filtration, the effectiveness of which is checked by the particulate monitoring system. A class of particular interest at reprocessing plants is the transuranics, especially plutonium. Although it is expected that these transuranic emissions can effectively be controlled by filter systems, the strict limitations on concentrations prompt effective monitoring. At Barnwell, only gross particulate monitoring would be performed: a gamma monitor uses a sodium iodide crystal, and an alpha monitor uses zinc sulfide scintillators. A more sophisticated and sensitive system for transuranic monitoring using alpha spectroscopy has been developed at Lawrence Livermore Laboratory.²¹ See Sec. IV for a

discussion of plutonium monitoring instrumentation.

Reference 22 reports emission of approximately 4 mCi/GW-yr of particulate alpha activity from NFS operations, substantially greater than the EPA limit of 0.5 mCi/GW-yr. Reference 20 reports measurements of alpha and beta particulate activity at NFS site perimeter stations.

Other radionuclides (gross monitors and laboratory analyses). Monitoring systems have ordinarily not been designed for continuous monitoring of radioactive species other than those mentioned above, although others may be of interest. Measurements of such radionuclides may be made by using laboratory analysis of samples taken at the stack. One such radionuclide is carbon-14, whose emissions may be estimated on the basis of fuel history, if controls are not implemented. Overall estimates of other species (such as ^{89}Sr , ^{106}Ru) may be obtained from the liquid condensate monitors and the particulate monitors: these are often not very effective limits, so that laboratory analyses are necessary as part of a complete monitoring program. Reference 16 summarizes the status of airborne effluent monitoring at fuel reprocessing and waste management facilities.

b. Liquid Effluents. Large reprocessing plants such as that at Barnwell are designed to release no radioactive liquid effluent, and the MRFP was similarly designed. The NFS had no such control and we shall indicate some results of environmental monitoring in the vicinity of NFS. However, even for plants as now conceived, effluent and environmental monitoring for possible liquid releases is prudent. These may depend on gross alpha and beta monitors, as indicated above for particulates, and on laboratory analyses of liquid and environmental samples.

At NFS, the major source of low level liquid waste was the "condensate from the series of evaporators which are used to concentrate high level liquid waste and to recover the acid" (Ref. 23).

The low level liquid wastes passed from the main plant into interceptor tanks, then into a series of three lagoons, and finally into the environment at Erdman's Creek, which flows eventually into Cattaraugus Creek, which flows downstream into Lake Erie some 39 miles away. Precipitation and sedimentation of insolubles, as well as holdup for the decay of short-lived species, are the important functions of the interceptor and lagoon systems.

The fate of liquid discharged from the system can be divided into several areas: some of the radionuclide activity flowed down Carraraugus Creek to Lake Erie; some settled

Table 4. Radionuclides Contributed to the
Environment in Liquid Effluent from NFS (Ref. 24)

Radionuclide	Quantity Discharged May to October 1969 (Curies)	Percent of 10 CFR 20 Concentration Limit in Cattaraugus Creek Water	
		June 17, 1969	November 4, 1969
Tritium	1700	1.1	0.10
Ruthenium-106	52	5.2	0.75
Cesium-137	8.0	0.008	0.15
Cesium-134	2.0	<1	<1
Strontium-90	8.3	17	23
Cerium-144	0.16	<1	<1
Promethium-147	0.092	<1	<1
Zirconium-95	0.0016	<1	<1
Cobalt-60	0.20	<1	<1
Antimony-125	0.59	<1	<1
Manganese-54	0.0027	<1	<1
Plutonium	0.0018	<1	<1
Uranium	0.032	<1	<1
Gross Alpha	0.048	<1	<1

out onto the bottom of the creek bed; some made its way into the biosphere through plant and animal takeup (small animals which drink the water; fish; aquatic-plant takeup); and perhaps some became airborne through evaporation.

The NERHL, during 1969, took measurements at several points in the liquid system and also studied creek bed samples.²⁴ For periods during this relatively active year at NFS, Table 4 shows radioactive discharges, as well as concentrations as a percentage of maximum permissible concentrations.

7. Summary

The handling of spent fuel may involve a variety of facilities providing for spent fuel storage, fuel reprocessing, mixed oxide fabrication, and geologic disposal. Monitoring requirements for spent fuel are not dissimilar from those for power plants. However, because fuel reprocessing plant free the radioactivity contained in the fuel rods, the possi-

bility of radioactive discharges and the corresponding monitoring requirements are more severe at such plants. We have summarized anticipated reprocessing emissions, noting their dependence on control design, and have indicated the monitoring instrumentation that has been designed for these facilities. More detailed information on monitoring specific radionuclides is presented Division IV of this volume.

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1. Introduction

This section on x-radiation is intended to provide background information about the extent of radiation exposures arising from use of x-radiation in modern medical and industrial practice and from incidental production in various products. For discussion of measurement techniques and instrument comparison, the reader is referred to section III.D, "X and Gamma Monitoring Instrumentation", elsewhere in this volume. The following discussion should be viewed as complementary to the instrumentation section.

2. Sources of X-Radiation Exposure

The number of x-radiation sources in the world today defies accurate quantification. Since the discovery of x rays by W. K. Roentgen in 1895, their use in industrial and medical applications has become widespread, and new applications are continually being developed or exploited.

a. Medical and Dental X-Radiation Exposures.

Surveys performed by the U. S. Public Health Service in 1964 (Ref. 1) and 1970 (Ref. 2) serve as a primary source of information on medical and dental x-ray exposures of the U. S. population. On the basis of these data, workers have been able to estimate doses to members of the population (Refs. 3 and 4). Some of the results of these surveys and the subsequent analyses are cited below. They are of particular interest in the context of human radiation doses since medical and x-ray exposures are the only human activity that contribute an average radiation dose of the same order of magnitude as natural background doses (Refs. 5, 6, and 7). Reference 6 contains a useful summary of results from the United States survey, as well as from those in other countries.

The methods of quantifying medical x-ray dosages differ among various researchers. One concept is the "whole-body dose", which is the easiest to understand, but difficult to interpret since exposures are seldom to the whole-body. Another is the "genetically significant dose", in which expo-

sure that are genetically significant are weighted heavily, while exposures to body extremities or to individuals of non-reproductive age receive little weight. A third concept is the dose to individual organs (e.g., thyroid doses from dental diagnostic procedures) or to a more general organ, i.e., bone. We shall not attempt to give detailed data here on all of these categories or their significance; however, we shall try to highlight some of the findings to demonstrate the extent and nature of medical and dental x-ray exposures. For purposes of general comparisons, doses from use of radiopharmaceuticals will also be indicated. Accelerator therapy is, however, excluded.

Genetically significant dose (GSD) is an index of radiation received by the genetic pool. This index permits comparisons between diverse national surveys and serves as an important measure of x-ray exposure. Mathematically the GSD is expressed (Ref. 8) as:

$$\text{GSD} = \frac{\sum D_i N_i \hat{P}_i}{\sum N_i \hat{P}_i}$$

where

D_i = the average gonad dose to persons age i who receive x-ray examinations,

$\hat{N}_i$ = the number of persons in the population of age i who receive x-ray examinations

P_i = the expected future number of children for a person age i , and

N_i = the number of persons in the population of age i .

Thus GSD weights the dose to individuals by their expected future number of children.

Because GSD has not generally been known for radiation sources other than medical radiology (e.g., environmental and occupational exposure), and because it gives no indication of somatic dose, the exclusive use of this index in the reporting of population exposures is not adequate for many purposes. (Somatic dose is that which affects the individual directly.) For this reason, although several other indices seem to be roughly proportional to the GSD over wide ranges of variables, it is useful to report such indices separately. It is also possible to formulate a "somatic dose index", as is done in Ref. 9.

In 1970, 130 million individuals made 179 million visits, and received 212 million x-ray examinations in which 661 million films were exposed (Ref. 2). These figures were all some 20% to 30% greater than those of 1964. However, it should be noted that the U. S. population increased 7% during this period. Table 1 (Ref. 2) gives the estimated annual rate of persons x-rayed during 1970. Note that 65% of the population had an x-ray examination of some kind. Figure 1 displays the age distribution for these data and for the 1964 data. For the 1964 and 1970 surveys, Figs. 2,3,4, and 5 (from Ref. 2) give respectively: the annual rate of medical x-ray procedures by body area; the distribution of examinations by number of films per examination; ratios of beam area to film area; the distribution of films by film diameter for dental x rays. Many of these parameters are significant when evaluating possibilities for reducing x-ray exposure. Table 2 (from Ref. 6) indicates some changes in the data between the two U. S. surveys.

Based on the 1964 and 1970 U. S. surveys, estimates have been made of various doses. However, the most recent estimates of genetically significant dose (GSD) have changed significantly from those calculated earlier. The total GSD for medical radiography to the U. S. population in 1964 was originally estimated (Ref. 1, publication 2001) to be 53 mrem per person. However, more recent analyses (Ref. 3) of the 1964 data, as well as that from the 1970 survey, give substantially smaller doses, in the vicinity of 17 and 20 mrad annual GSD per capita, respectively, for 1964 and 1970. The apparent increase from 1964 to 1970 is not statistically significant (Ref. 3).

However, other measures of dose appear to show a statistically significant increase. In particular, the mean annual active bone marrow dose to the adult U. S. population from diagnostic radiology shows an increase from the 1964 value of 83 ± 4 mrad per capita to 103 ± 5 mrad for 1970. Table 3 indicates the principal contributors to this dose, and also gives average doses per examination. The variation in x-ray examination rate by age group implies a significant increase in per capita dose with age. Assuming that for each particular type of examination the dose does not vary with age group, Ref. 4 estimates a per capita mean active bone marrow dose (1970) ranging

from 52 mrad for the 15-24 yr age group to 151 mrad for those 65 or older. On the other hand, about 90% of the GSD is contributed by exposures of individuals less than 30 years old (Ref. 3).

As indicated in Table 3, dental x rays contributed about 3% of the mean active bone marrow dose. They contribute an even smaller portion of the genetically significant dose.

Although therapeutic radiography was included in the 1964 and 1970 surveys, the corresponding contributions to GSD and bone marrow dose have not been estimated in Refs. 3 and 4. The portion of x-ray procedures employed for such purposes is relatively small, less than 1% of the total. These procedures, however, contribute a somewhat larger proportion of dose, since therapeutic radiological procedures typically entail larger average doses than diagnostic procedures (Ref. 6).

Radiopharmaceuticals are also used for diagnosis and therapy. However, resulting annual per capita doses have not been estimated on the same basis as those from diagnostic radiology. As noted in Ref. 5, radiopharmaceuticals and radiation therapy contribute far less per capita dose than diagnostic x rays. Data on U. S. use of diagnostic radiopharmaceuticals are given in Ref. 6, Table 3 (p. 339), based on Ref. 8. The dose to specific organs depends on the examination type; the total contribution to per capita gonad dose is approximately 0.5 mrad/yr.

The estimated doses given above may be compared, to some extent, with natural background doses. Assuming an RBE of 1, the GSD of 20 mrad/yr from diagnostic radiology in 1970 is considerably less than the 80 mrem/yr gonad dose estimated (Ref. 7) from natural sources. On the other hand, the 103 ± 5 mrad 1970 mean bone marrow dose exceeds the estimated whole body dose of 80 mrem/yr from natural sources. Even though these doses are not strictly comparable, they suggest that in Fig. 6, the dominant contributions to U. S. average whole body doses, i.e., natural and medical, should be reversed in relative importance. Extrapolations to the year 2000 in Fig. 6 are, of course, only estimates; the indicated total could be strongly affected by changes in the medical contribution (due to changes in the frequency of or average dose from diagnostic procedures) or in the natural background contribution

Table 1. Estimated annual rate of persons x-rayed by type of procedure, sex, and age, United States, 1970 (reproduced from Ref. 2).

Sex and age	Rate per 100 persons					
	All types	Medical			Dental	
		Total	Radio-graphic ^a	Fluoro-scopic	Thera-peutic	
Both sexes	65.0	38.6	38.3	4.6	0.2	29.6
Under 15	42.4	17.8	17.7	0.7	(*)	25.9
15-29	77.3	41.1	40.8	2.0	(*)	40.8
30-44	76.9	47.1	46.8	6.4	(*)	34.3
45 and over	70.9	52.1	51.6	9.3	(*)	22.0
45-64	76.4	53.6	52.9	8.8	(*)	26.7
65 and over	58.8	49.1	48.5	10.3	(*)	11.7
Male	62.6	38.6	38.2	4.6	(*)	27.2
Under 15	44.1	19.8	19.8	1.0	(*)	25.9
15-29	73.4	43.6	43.3	2.0	(*)	34.7
30-44	71.5	44.1	43.8	5.8	(*)	31.7
45 and over	68.8	51.6	50.7	9.8	(*)	19.8
45-64	71.4	51.0	50.0	9.0	(*)	23.6
65 and over	62.4	53.2	52.4	11.7	(*)	10.5
Female	67.2	38.6	38.3	4.6	(*)	31.9
Under 15	40.6	15.8	15.6	(*)	(*)	25.9
15-29	80.8	38.9	38.5	2.1	(*)	46.3
30-44	81.8	49.8	49.6	6.9	(*)	36.7
45 and over	72.7	52.6	52.3	8.9	(*)	23.8
45-64	81.0	55.9	55.6	8.7	(*)	29.5
65 and over	56.1	46.0	45.7	9.3	(*)	12.5

^aIncludes persons with photofluorographic examinations.

* Figure does not meet standards of reliability or precision (more than 25 percent relative standard error).

Fig. 1. Estimated annual rate of persons x-rayed during medical and dental visits by age, United States, 1964 and 1970 (reproduced from Ref. 2).

Fig. 2. Estimated annual rate of medical x-ray procedures by body area, United States, 1964 and 1970 (reproduced from Ref. 2).

(which may, for example, be effectively altered by changes in housing construction practices).

Reference 6 reviews medical exposures from a number of countries, and substantial variation is evident. For example, a 1969 Japanese survey gave a per capita annual bone marrow dose of 189 mrad from diagnostic radiology; the value had decreased to 132 mrad by 1974.

Average U. S. population doses may have changed significantly since 1970. The increase that had taken place since 1964 may have continued, particularly as new techniques have been introduced. The most notable of these is computerized axial tomography (CT); by 1979, 700 CT scanners were in use, examining approximately 2 million patients per year, with typical doses to the head of 1 to 5 roentgens (Ref. 11); a study of one scanner concluded that an average head dose of 2.1 or 2.4 rad (depending on the kVp) resulted from a typical procedure (Ref. 12). On the other hand, attention to x-ray equipment and procedures serves to limit average doses, as discussed below.

b. Occupational Exposures to X-Radiation.

The widespread and diverse use of x-radiation in medicine and industry makes it difficult to estimate occupational exposures. References 5 and 6 survey available information for the United States and the world, respectively. In both cases, much of the United States data is drawn from Ref. 10, which - in turn - bases its estimates of U. S. exposures on very limited data.

Based on available data on medical exposures around 1970, Ref. 10 adopted average annual per capita whole body doses of 320 mrad for about 195,000 non-federal medical x-ray workers and 125 mrad for 171,000 non-federal dental x-ray workers. These approximate values are roughly consistent with the data cited in Ref. 6. As for other radiation-related occupations, the value of such averages is diminished by ambiguities in defining who is an x-ray worker, as opposed to an auxiliary worker. Similarly, based on data from a small survey, Ref. 10 adopted a value of 540 mrad/year for 38,000 radium workers. The value adopted for 50,000 radioisotope workers was 262 mrad/yr. The result of a different type of sur-

Fig. 3. Estimated percent distribution of radiographic examinations of number of films per examination for selected body areas, United States, 1964 and 1970 (reproduced from Ref. 2).

Fig. 4. Estimated mean ratio of beam area to film area for radiographic films for selected body areas by type of facility, United States, 1964 and 1970 (reproduced from Ref. 2).

Fig. 5. Estimated percent distribution of dental films by film diameter, United States, 1964 and 1970 (reproduced from Ref. 2).

Table 2. Changes in data pertaining to diagnostic x-ray procedures in a six-year period: United States of America, 1964 and 1970 (reproduced from Ref. 6, p. 305).

	1964	1970	Increase (per cent)
Number of persons having x-ray examinations	108x10 ⁶	130x10 ⁶	+ 20 ^a
Number of x-ray procedures	173x10 ⁶	212x10 ⁶	+ 22 ^a
Number of films exposed	506x10 ⁶	661x10 ⁶	+ 30 ^a
Average number of films per examination	2.2	2.4	+ 9
Fraction of thoracic examinations with two or more films	31 %	47 %	+ 52
Mean ratio of beam area to film area (in hospitals)	1.9	1.2	- 37
Estimated mean skin exposures for posterior-anterior and anterior-posterior views of the abdomen	480 mR	620 mR	+ 29 ^b
Mean skin exposure per dental film	1140 mR	910 mR	- 20 ^c

Sources: See Ref. 6.

^aThe population of the United States increased by 7 per cent over the six-year period.

^bThis increase may be due to the increased frequency of high-exposure examinations, the increase in the use of grids, and the use of higher tube potentials and currents without a corresponding increase in filtration.

^cThis decrease is due to the increased use of faster film.

Table 3. Mean bone marrow dose to the adult U. S. population (1964 and 1970)^a

Type of examination	Annual per capita dose ^b (mrad)		Dose per examination ^b (mrad)	
	1964	1970	1964	1970
Hip and upper femur	0.7	0.7	97	72
Pelvis	1.4	1.1	116	93
Lumbosacral joint	4.0	5.7	418	450
Lumbar spine	6.7	8.1	336	420
Urography	9.9	10.1	453	420
Stomach, upper GI tract	17.9	24.3	408	535
Small intestine	0.7	1.0	271	422
Colon	13.7	21.2	624	875
Abdomen	3.6	2.9	183	147
Cholecystography	3.2	3.7	183	168
Dorsal spine	2.0	2.5	232	247
Lung	2.0	3.2	10	10
Lung (photofluorography)	7.8	3.2	65	44
Head	1.0	1.6	65	78
Dental	3.5	2.9	13.2	9.4
Other	4.9	10.8	-	-
total	83 ± 4 ^b	103 ± 5 ^b		

^aBased on Refs. 4 and 6.

^bOf the standard errors given in Ref. 4, we cite only those for the total annual per capita dose.

Fig. 6. Summary of estimated average whole body radiation doses in the United States (mrem/person), reproduced from Ref. 10).

vey, i.e., of 4/ nuclear medicine departments, yielded values for average annual per capita doses to employees of about 470 mrad for the period 1970-1973, with somewhat lower values in 1968-1969 (Ref. 13). In many of the survey results available (see Ref. 6), about 10% of the individuals included receive annual doses in excess of 1 rad.

Similar difficulties arise in characterizing industrial dose rates. One set of data arises from reports of Nuclear Regulatory Commission (NRC) licensees, including industrial workers not connected with nuclear power. Average doses reported to the NRC on about 9000 industrial radiographers ranged from 0.41 rem/yr in 1973 to 0.30 in 1975. During this period the average annual dose to the 50 to 60% of these radiographers with measurable doses was consistently 0.6 rem; about 10% of the 9000 received doses in excess of 1 rem during 1975. The dose data were similar for a smaller group of workers employed by NRC "manufacturing and distribution" licensees (Refs. 5 and 6). Most industrial workers exposed to x rays are not NRC licensees, but they are often "agreement state" licensees (see discussion in Section II.A, Particle Accelerators). Reference 10 estimated workers for such licensees (from about half the states) to total about 24,500, with a mean dose of 274 mrem/yr, consistent with the NRC data.

In some cases, surveys have been performed of particular types of accidental exposures. For example, Refs. 14 and 15 discuss exposures from units used for x-ray analytical work (diffraction studies, x-ray spectrography, x-ray fluorescence) in Pennsylvania (as a sample of the United States) and Sweden. The data suggest a probability of severe injury to be of the order of 1/100 per unit per year (for 1968), and 3 or 4 times higher for less serious injuries with clinical symptoms. These exposures make an insignificant impact to the average population dose, but are very significant from an occupational health viewpoint.

The total contribution of occupational exposures to the average population dose is relatively small. In Ref. 10, it was estimated to be slightly less than 1 mrem/yr (see Fig. 6). This is roughly consistent with the fact that about 0.4% of the U. S. population works in occupations exposed to radiation (Ref. 6) at typical dose rates (as noted above) of 200 to 300 mrem/yr.

c. Other Sources of X-Ray Exposure.

X-ray exposures that are neither medical nor occupational in nature are not well quantified. Some of the specific sources are: color television receivers in private homes; cold cathode gas discharge tubes, used for demonstrations in school science classes; accelerators and medical facilities that expose passers-by to small doses; high-voltage electronic devices (e.g., klystrons) that function improperly so as to leak x-radiation; and baggage x-ray systems.

Some limited surveys of these potential sources have been completed, often by the Bureau of Radiological Health (BRH), which -- as discussed in the next section -- has the responsibility for formulating electronic product radiation safety performance standards. NCRP report 56 (Ref. 16) summarizes information on doses from many such products. Color televisions received specific attention when it was found that a significant percentage had exposure rates that exceeded the NCRP recommended limit of 0.5 mR/hr at 5 cm distance, which was adopted by the FDA in 1970 (see below); average values were much lower, and it has been estimated that a rate one tenth of the limit would cause annual exposures averaging about 1 mrad/year; newer televisions would yield even less (Ref. 16). A BRH in-plant survey of television sets is described in Ref. 17. The BRH surveyed cold cathode gas discharge tubes in 1969 (Ref. 18), finding that some tube types emit excessive radiation (see also Ref. 13). Few data are available on exposures of passers-by near medical and accelerator x-ray sources, although these would be controlled through proper engineering practice; Ref. 5 cites values of 0.04 to 4 mrem/yr at accelerator site boundaries (not necessarily from x rays). See also Section II.A, on particle accelerators. Typical exposures from x-ray baggage inspection devices are about 0.001 mR or less per inspection (Ref. 16).

3. Licensing and Design for X-Ray Facilities

Most medical and industrial sources of x-radiation in the United States are subject to license by the several states. Practices have varied

Table 4. FDA electronic product radiation safety performance standards and date of applicability (cf. Ref. 20).

<u>Performance Standard</u>	<u>Date of Applicability</u>
Television Receivers	Jan. 15, 1970
Cold Cathode Gas Discharge Tubes	May 19, 1970
Microwave Ovens	Oct. 6, 1971
Diagnostic X-ray Systems and Their Major Components	Aug. 1, 1974
Cabinet X-ray Systems	April 10, 1975
Baggage X-ray	April 25, 1974
Laser Products	Aug. 2, 1976
Ultrasonic Therapy Products	Feb. 17, 1979
Mercury Vapor Discharge Lamps	March 9, 1980

widely in the past, ranging from regular licensing and surveillance of essentially all sources to almost complete lack of regulation. Similarly, there have been wide differences in the licensing or certification of operators of x-ray equipment. Some states have certification programs for all medical x-ray workers; in California, even physicians are subject to this certification (Ref. 19). The licensing of individual industrial x-ray workers has been practically non-existent, which is particularly unfortunate since most large accidental exposures arise from human error rather than from mechanical failure.

Progress is being made in various of these areas. Federal performance standards have been set for x-ray equipment. State programs for inspecting x-ray machines are assuring an increasing degree of compliance with applicable codes. Increasing attention is being given to proper use of x-ray equipment, partly as a result of State-Federal cooperative programs. And, to an increasing degree, recommendations on equipment and operation are becoming available from national (and international) bodies concerned with radiation protection.

Under the U. S. Radiation Control for Health and Safety Act of 1968 (PL 90-602), the Bureau of Radiological

Health of the Food and Drug Administration is empowered to set performance standards for electronic products that emit radiation as discussed in Ref. 20. Table 4 indicates the applicable FDA standards as of 1979; many of these pertain to x-ray emitting equipment, of which that for diagnostic x rays is the most important contributor to average population doses.

Proper design and maintenance of equipment, together with correct use, can have substantial effect on doses from diagnostic x rays. Key questions include the voltage and current at which the x-ray beam is generated, the collimation of the beam, use of image intensifiers, and choice of film. Often these questions and the exposure used are determined, not by equipment design or condition, but by the procedure chosen by radiology personnel. Efforts to survey and control such factors are indicated in the next section.

Aside from the recent FDA performance standards, a substantial basis for the design and operation of x-ray equipment and facilities is provided by recommendations of radiation protection organizations. Relevant NCRP publications include: "Medical X Ray and Gamma Ray Protection for Energies up to 10 MeV, Equipment Design and Use," Report No. 33 (Ref. 21); "Structural Shield-

ing Design and Evaluation for Medical Use of X Rays and Gamma Rays of Energies up to 10 MeV", Report No. 49 (Ref. 22); "Dental X-Ray Protection", Report No. 35 (Ref. 23); "Radiation Protection for Medical and Allied Health Personnel", Report No. 48, (Ref. 24). (See also ICRP Publ. No. 16, Ref. 25.) Recommendations that apply specifically to industrial use of x rays include: NCRP Report No. 61, "Radiation Safety Training Criteria for Industrial Radiography" (Ref. 26); NBS Handbook 114, "General Safety Standard for Installations Using Non-Medical X-Ray and Sealed Gamma-Ray Sources, Energies up to 10 MeV" (Ref. 27); NBS Handbook 111, "Radiation Safety for X-ray Diffraction and Fluorescence Analysis Equipment" (Ref. 28); and BRH's "Radiation Safety Recommendations for X-Ray Diffraction and Spectrographic Equipment" (Ref. 29).

4. Monitoring X-Ray Exposures

Adequately protecting patients and operators from excessive x-ray exposures or from accident requires a systematic program of monitoring and training. Many elements of such programs were indicated in the last section, particularly for protecting operators of x-ray equipment. On the other hand, most of the population dose from x rays is to patients in medical procedures. Monitoring patient exposure and its dependence on various factors requires a well-designed program, and such a program - together with operator training - has a great potential for leading to significantly reduced patient exposure.

One of the most important contributors to unnecessary patient exposure can be inadequate collimation of the x-ray beam. For example, the ratio of beam area to film area has been larger than necessary in some cases (cf. Fig. 4). This ratio decreased significantly between the 1964 and 1970 surveys and is still a parameter of great interest in current monitoring and training programs. Figure 5 showed the distribution of dental films by beam diameter; an increasing percentage of films are taken with beams no larger than the 2.75 inch maximum diameter recommended by the American Academy of Dental Radiology.

Other factors, too, are important. These include the kilovolt peak (kVp) of the x-ray generator, the film speed, and so on. In some cases, the impacts of changing these factors is correlated. For example, with the relatively fast film now used in most dental radiography, use of a higher kVp reduces exposure (Ref. 30). Assurance of photographic quality is an important related consideration (Ref. 31).

Because patient exposures can be significantly affected by x-ray procedures, state and national agencies are making substantial efforts to monitor patient exposures and to effect reductions through programs of education and training. A useful avenue for effectively communicating information among interested parties has been the series of Annual Conferences on Radiation Control. Since the 1970 exposure study, new data on the effects of training have become available through the Nationwide Evaluation of X-Ray Trends (NEXT), a cooperative x-ray survey program between the Bureau of Radiological Health and State radiation control programs. In this survey, the x-ray equipment operator is asked by an inspector to set up the machine for performing an examination of a patient of standard size. The measurements taken permit determination of the dependence of exposure and other parameters on training and equipment. For example, an examination of 1972-1975 data indicates that training improves collimation and decreases exposures (Ref. 30).

Such surveys, as well as radiation control efforts on the part of individual institutions, depend on effectively calibrated monitoring instruments. Some surveys, such as the 1964 and 1970 exposure studies, depend directly on use of standard films. On the other hand, direct measurements can be taken with instruments - including ion chambers - that are sensitive to the x-ray spectrum being employed.

With a few exceptions, most medical diagnostic procedures are in the region above 50 kVp. Only in soft tissue radiography (such as in mammography) is lower energy radiation common: in such cases, 20 to 30 kVp is typical. Another low-energy application is superficial therapy (such as acne treatment, where energies as low as 5 kVp are used. At the high end, energies above 150 or 175 kVp are rarely found, although there is some use of several MeV linear accelerators for diagnostic

chest studies. Dental diagnostic equipment is also almost exclusively in the 50-100 kVp range.

From these comments, it is clear that a large portion of the surveys requires an instrument whose energy response is flat down to at least 50 keV and that lower-energy monitors are needed for special applications. Actual x-ray monitoring instrumentation is discussed in Section III.D.

5. Summary

In this section, we have outlined the principal sources and magnitudes of x-radiation exposure in the United States. Of these sources, medical and dental x-radiation, principally for diagnostic purposes, produces the largest dose to humans, a dose that is larger than that from any other man-made source of radiation by more than an order of magnitude, and that is comparable to the dose from natural background. Occupational sources, on the other hand, contribute significant doses to those occupationally exposed, and hence are of great importance to this population, but contribute little to the average dose to the general population. Other miscellaneous sources, such as consumer products, typically contribute little either to individuals or to the average population dose. Doses from occupational and miscellaneous sources are controlled by attention to the equipment or devices being used

and also - in the occupational situation - to adequate training of personnel employing the equipment or sources.

Although the effects of exposure control efforts in medical and dental use of x rays has been evident, further significant reduction can be achieved. This may be accomplished primarily through the more widespread application of collimators, the use of more sensitive film, and proper technical training. A factor of two reduction below the levels cited in this section is conceivable without any significant technological breakthrough, but with diligent efforts on the part of interested agencies and practitioners.

For any x-ray exposure control effort, effective monitoring techniques are required. In many cases, particularly for diagnostic applications, standard film packages may be used for monitoring. In all cases, other types of instrumentation may be required, and these are discussed in Section III.D.

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7. References

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E. Mining and Milling of Uranium Ores

1. Introduction

In this section, we give an overview of the radiological problems associated with the mining and milling of uranium ores, activities that are increasing with the rising worldwide demand for nuclear fuel. Requirements for radiation-measuring instrumentation are emphasized, noting how specific aspects of the mining and milling industry have created the need for unique instrumentation. Radiation protection guides and general radiological safety considerations are also summarized. Although this section is devoted to uranium mining and milling, many of the same considerations apply to the extraction of other materials, particularly thorium but also (because of the pervasive distribution of uranium and thorium) non-nuclear materials, such as the Florida phosphates. Ore mining and processing enhances the contact and distribution of natural decay chain radionuclides in the biosphere.

We attempt to emphasize the important radiological impacts and the resulting instrumentation requirements for the uranium industry, giving little attention to measurement problems that are not unique to the industry or that are of very restricted interest. We therefore discuss primarily measurements inside underground mines, of radon-222 and its daughters (or of the "Working Level" a quantity defined for occupational exposures) and measurements outside mills (e.g., mill tailings) of radionuclides in air and water. We will also discuss briefly external γ measurements, bioassay, and airborne particulate monitoring. In many respects, this section complements that on "Radon and its Daughters", which includes detail on the properties of radon, its decay chains, and other physical considerations. The section on "Radium" is also of interest, largely because of its presence in mill tailings. Since these radionuclides tend to be α -emitters, we also refer the reader to "Alpha Particle Instrumentation" (Section III.B)

2. Environmental Radiation Hazards

Although there are other environmental impacts associated with mining and milling (Ref. 1), we will discuss only those that derive from radioactivity. The details of uranium mining and milling techniques may be found elsewhere (Refs. 2,3). Briefly, uranium is extracted principally from underground or surface (open pit) mines, although solution mining is becoming more widespread (Ref. 4). Milling ores from mines involve crushing, leaching, extraction, and conversion operations. Radiological hazards may arise from uranium, or its daughters, most importantly, ^{238}U , ^{235}U , ^{234}U , ^{230}Th , ^{226}Ra , and Radon and its daughters.

(See Table 1 and Section IV.G, "Uranium".) In mining and milling, α , β , and γ radiation from these nuclides can be important with hazards arising primarily from internal exposure and to a lesser extent from external radiation.

The hazards from exposure to β and γ radiation, uranium dusts and surface contamination in open pit mining for uranium are about the same as those encountered in underground mining (Ref. 5), but underground mining has more significant problems associated with radon and radon progeny. Although mining operations will release radioactivity in the air, with proper ventilation it can be dispersed and will dilute rapidly in the atmosphere. Radon is a noble gas, unlike its precursors; therefore it can easily become airborne. The airborne radionuclides consist of radon and its daughters, which constitute the most serious hazard, and long lived α -emitters, which are present in dusts of the mine atmosphere from the decay chain of natural uranium. Where underground mines are poorly ventilated and radon can collect, radon and its daughters have been found to pose substantial radiological hazards for uranium miners. Gamma radiation levels greater than 2 mrem/hr sometimes pose an external hazard even in lower grade mines due to buildup of ^{214}Bi (RaC') (Ref. 6). The other members of the uranium series, with less mobility and long half lives, do not ordinarily pose a comparable hazard, either from external exposure or from internal exposure due to respiration or ingestion of these nuclides in the mineral dusts.

Mine effluent wastes include liquids, solids (particulates and tailings), and gases. The liquids are mostly water from mine drainage and water used in drilling. Solid waste is primarily waste rock and low-grade ore. Leaching of mine tailings will be an additional source of low-level liquids waste. Since radon is soluble to a certain extent, sometimes water seeping into a mine may be a source of radon. About 1% of Ra in the ore is soluble and can also be contained in water, resulting in contamination if it is discharged directly to the environment from mining operations. Contamination of surface or ground water could cause a potential exposure from drinking water to be important. Although ground water impacts and disposal of liquid wastes are the primary environmental concerns associated with in-situ solution mining, similar problems associated with conventional mining and milling operations are actually larger. When a 0.1% grade of ore is assumed, an estimated maximum of 16 tons of solid waste per ton of produced yellowcake would be generated by solution mining compared with 1000 tons of radioactive wastes (tailings) associated with conventional milling methods. In addition, the tailings from conventional operations contain almost all the ^{226}Ra and

Table 1. Some Important Radionuclides and Their Properties

Isotope	Type of Activity	Half-Life	α Energies (if any) (Branching Ratio)
^{238}U	α, β^-, γ	$4.51 \times 10^9 \text{ y}$	4.20 MeV
^{235}U	α, γ	$7.1 \times 10^8 \text{ y}$	4.58 MeV
^{234}U	α, γ	$2.45 \times 10^5 \text{ y}$	4.7 MeV
^{222}Rn	α	3.82 d	5.48 MeV
^{218}Po	α	3.05 m	6.00 MeV
^{214}Pb	β^-, γ	26.8 m	
^{214}Bi	β^-, γ	19.7 m	
^{214}Po	α	$1.6 \times 10^{-5} \text{ s}$	7.68 MeV
^{210}Pb	β^-, γ	22.3 y	
^{210}Po	α, γ	138.4 d	5.3 MeV
^{230}Th	α	$8.0 \times 10^4 \text{ y}$	4.68 MeV
^{232}Th	α	$1.4 \times 10^{10} \text{ y}$	4.01 MeV(76%) 3.95"(24%)
^{228}Th	α	1.9 y	5.42 MeV(70%) 5.34 MeV(28%)
^{226}Ra	α, β^-, γ	16.02 y	4.78 MeV
^{224}Ra	α	3.665 d	5.3 MeV
^{220}Rn	α	55.3 s	6.29 MeV
^{216}Po	α	0.15 s	6.78 MeV
^{212}Pb	β, γ	10.64 hr	
^{212}Bi	α, β, γ	60.60 m	6.09 MeV(10%)
^{228}Ra	β, γ	6.7 y

other daughter products associated with the ore, whereas only <3% of the radium in the ore is brought to the surface during in situ mining (Ref. 7). In order not to negate any advantage gained from smaller quantities of radioactive materials reaching the surface environment, deposition of the solid wastes generated must be carefully planned to prevent proliferation of small waste disposal sites. We see that the mining process exposes previously inaccessible uranium so that the daughter activities, particularly radium and radon, have a greatly increased probability of transport to the general environment. The primary population at risk is the uranium miner.

Uranium is extracted by a milling operation that uses sulfuric acid (or, less often, an alkali) to leach the uranium from ore that has been crushed and ground. The uranium is subsequently recovered from solution by solvent extraction or ion exchange. Its product is a semi-refined uranium compound called "yellowcake" which is the feed material for the production of uranium hexafluoride. The uranium mill produces large volumes of both solid waste, basically tailings from the ore, and liquid waste, from the various mill processes. When initially discharging its waste, the bulk of the wastes is a mixture of the solids and liquids forming a slurry for easy movement to the disposal site. Most of the non-volatile uranium decay daughters are relegated to these waste streams.

Volatile daughters, notably radon, may escape throughout the milling operation. Airborne particulates may be released from yellowcake drying and packaging operations (depleted in uranium daughters), crushing and grinding operations, and the ore storage pads. However, over the long term, the principal source of radon and its daughters is the decay of longer-lived uranium daughters, most importantly ^{230}Th and ^{226}Ra , present in the waste tailings (Ref. 8). These two radionuclides can also have direct radiological importance, either because of their presence at relatively high concentration in the liquid milling effluents (Ref. 2) or because of their presence in tailings allowing them to become airborne in particulate form (Ref. 8). Airborne releases from the milling process thus include both particulate matter and gases. Low stacks are the means by which these dusts or gases are discharged to the atmosphere from milling operations.

The liquid effluent from a process mill consists of waste solutions from the leaching, grinding, extraction, and washing circuits of the mill. The radioactive daughter products of radon may be present in small concentrations. Since the concentrations of ^{226}Ra and ^{230}Th are about an order of magnitude above specified liquid limits in federal regulations 10 CFR 20 (Ref. 9), considerable effort must

be exerted to prevent any release of the material from the site. The solid and liquid waste is stored as a slurry in a tailings retention pond which should be constructed to prevent discharge into the surface water system and to minimize percolation into the ground.

The tailings retention system or "tailings pond" will have a radiological impact on the environment through (1) the air pathway by continuous discharge of radon-222 gas (2) external exposure through γ rays given off by decay of radium-226, radon-222 daughters (primary ^{214}Bi), and (3) through air, water, or food pathways if radium-226 and thorium-230 are blown off dried areas of the tailings ponds by wind or are leached from the pond into surface waters (Ref. 8).

When a retention system involving a dam is initially put into use, because all the waste is under water, it is expected that there will be small losses of radioactive slurry waste as seepage through and around the dam. Radium-226 is a radionuclide of concern because levels as high as 32 pCi/l have been found in seepage from current operating mills (Ref. 2). Unless the impoundment area is lined with an impervious material, considerable quantities (10% - >80%) of the liquid effluent from the mill will leak out through the bottom of the pond (Ref. 10). This is a potential problem requiring continuing monitoring programs to insure that there is no significant movement of contaminated liquids into the offsite environment.

Toward the end of the operating lifetime of a tailing retention system, some of the tailings will no longer be under water and will dry out to form a beach. Wind erosion can carry off tailings material from these beaches as airborne particulate matter. The amount of radioactive particulate material removed from the tailings beach by wind erosion depends on the area and topography of the beach, the wind velocity, and particle size distribution of the tailings (Ref. 8). Long-lived alpha-emitting radionuclides U-238, Th-230, and Ra-226 are some of the source terms and the most significant contributors to the lung dose. Since the airborne particulate effluent discharges of tailings dust (as well as calcinated yellowcake and ore dust) are considered insoluble aerosols when they are inhaled, these aerosols tend to remain in the pulmonary region of the lung so that the lung becomes the critical organ when the radiation dose is calculated. A recent study (Ref. 11) has indicated that windblown tailings from inactive unstabilized tailings piles have caused elevated γ exposures >25 mrem/yr at distances up to one mile from the pile. However, the inhalation pathway is considered to be the critical pathway because levels of control sufficient to limit radiation exposure through inhalation will also prevent exposures through the

ground deposition whole body exposure pathway. From the exposure point of view, the primary concern of tailings piles is long-range atmospheric diffusion of the noble gas ^{222}Rn and resultant air concentrations of daughters. Radiation exposure to radon gas and radon daughters constitutes over 98% of the environmental impact of such piles (Ref. 12). A recent study (Ref. 13) was undertaken that provided estimates of inhalation dose from ^{222}Rn , ^{210}Pb , and ^{210}Po to the population in the Western U.S. resulting from unit releases of ^{222}Rn from tailings piles. Table 2 is taken from that report. Due to lack of proper treatment in the past, significant problems have arisen from inactive uranium milling sites (Ref. 14,15). Table 3 summarizes the hazards from tailings. Compared with other types of nuclear waste, uranium mill tailings are generated in large volume, about 10-15 million tons annually (Ref. 16) and are long half-life, low-level radioactive wastes (Ref. 17).

In the absence of very large controlled areas, or unless individuals live more than 1 mile from the tailings pile, the pile tailing source term must be kept very low by using preventive measures. Many studies (Refs. 18-23) have been done to determine the source terms and ambient levels from the tailings piles, both active and inactive. In addition, studies on transport and distribution in the environment have also been carried out (Refs. 24,25).

In summary, for the entire mining and milling process (including after shut-down) the potential critical pathways to be considered are inhalation of α -emitting radioactive airborne particles and radioactive gas,

deposition of airborne radioactive particles causing whole body exposure from γ rays, and radioactive contamination of surface and/or ground water by seepage from the tailings pond or by discharge of process water, and potentially the food pathway via airblown deposition or irrigation with contaminated water. The reader should be aware that mining and milling each significantly impact different groups of populations. Mining tends to be an occupational exposure problem, while milling tends to affect the general population exposure level.

3. Units of Exposure

Radiation protection guides for most radiation exposures are expressed in the familiar units: For external whole-body exposures or exposures of specific organs, the limits are stated in absorbed dose equivalent (rem); and for radionuclide concentrations in air and water, the limits are stated as maximum permissible concentrations (in $\mu\text{Ci}/\text{cm}^3$, for example).

For the purposes of radiological protection of the lungs in the uranium mining industry, a specialized unit of exposure to radon-222 and its daughters has been developed. This is the *Working Level* (WL), defined as "any combination of radon daughters in one liter of air that will result in the ultimate emission of 1.3×10^5 MeV of alpha energy" (Ref. 26).

The reason for the specialized unit is mainly operational: the WL is a concept having validity in any mixed concentration of radon and its daughters, whether or not they are in equilibrium. Just as important, it

Table 2. Population dose per unit radon release to inhabitants of the United States from inhalation of ^{222}Rn , ^{210}Pb , and ^{210}Po in 1978. (From Travis et al., Ref. 13, with permission by American Nuclear Society © 1978.)

Release site	Population dose per kCi ^{222}Rn released in 1978			
	Man-rem (Lung ^a)	Bone ^b	Organ-rem Kidney ^b	Liver ^b
Casper, Wyoming	84.8	24.3	0.5	0.4
Falls City, Texas	109.5	21.3	0.5	0.4
Grants, New Mexico	71.5	19.9	0.4	0.4
Wellpinit, Washington	61.5	20.7	0.5	0.4

^aDose to bronchial epithelium of tracheobronchial tree.

^bDose from uniform distribution of radionuclides in organs.

Table 3. Principal radionuclides involved in the exposure pathways from tailings piles to man. (From Goldsmith, Ref. 14, with permission by Nuclear Safety © 1976.)

Pathway	Radionuclide	Pathway Processes	Exposure Hazard
Radon diffusion	^{222}Rn	1. Diffusion to pile surface 2. Atmospheric transport 3. Inhalation	Inhalation of daughters of ^{222}Rn , ^{210}Pb , ^{214}Pb , ^{214}Bi , and ^{214}Po
Airborne activity	^{230}Th	1. Resuspension from pile surface 2. Atmospheric transport 3. Inhalation 4. Body surface contamination	Inhalation of airborne activity carried by respirable particles
Terrestrial	^{226}Ra , ^{210}Pb	1. Resuspension from pile surface 2. Atmospheric transport 3. Deposition on soil or foliage 4. Uptake by plants 5. Uptake by animals 6. Consumption of contaminated plants, meat, and milk 7. Consumption of contaminated soil	Ingestion of ^{226}Ra and ^{210}Pb
Aquatic	^{226}Ra , ^{210}Pb	1. Dissolution of radionuclide in pile 2. Migration through soil to water 3. Migration in water 4. a. Uptake by aquatic organisms b. Consumption of aquatic organisms 5. a. Use of water for irrigation b. Uptake by plants c. Consumption of plants 6. Consumption of contaminated drinking water 7. Immersion in contaminated water	Ingestion of ^{226}Ra and ^{210}Pb
External gamma	^{226}Ra daughters	1. Attenuation by pile material 2. Air-distance attenuation 3. External exposure	External exposure to gamma radiation produced by the decay of ^{226}Ra and its daughters

lends itself to practical measurements in the mines.

We note that the WL considers only radon-222 daughters and not radon gas itself. This is because radiation dose to the lung comes primarily from the retention in the lung of the daughters. Radon itself is chemically inert, is not preferentially retained, and hence gives substantially less dose to the lungs.

An extension of the WL concept is the "Working Level Month" (WLM), which expresses

a cumulative exposure. It is defined as follows: "Inhalation of air containing a radon daughter concentration of one WL for 173 working hours results in an exposure of WLM" (Ref. 27).

4. Epidemiological Data

Today, there seems to be little doubt about the epidemiological evidence connecting uranium mining and lung cancer. This is true despite some disagreements about the *detailed* conclusions which can be drawn from the available evidence. Holaday (Ref. 28) has written

an excellent historical account of the uranium miners' radon exposure problem. (This account is part of the May 1969 issue of *Health Physics*, which issue is entirely devoted to various reports in the same field of study (Ref. 29). The epidemiological data were reviewed in 1967 by the now-defunct Federal Radiation Council (FRC) (Ref. 26), and again in 1971 by the National Institute for Occupational Safety and Health (Ref. 28). The FRC noted "a clear association between exposure to radon daughters in mine air expressed as cumulative WLM and the number of lung cancer deaths in the study group," and this is confirmed in more recent reviews (e.g., Ref. 31). Figure 1 (from Ref. 26) shows some of this evidence, indicating increased mortality at cumulative exposures greater than 1000 WLM.

The FRC indicated that there seems to be a synergistic association with cigarette smoking as well. Of other possible agents that might cause lung cancer (silicon, hydrocarbons, longer-lived radioactive materials such as thorium or uranium) none seem to have supportable epidemiological evidence. (Ref. 26).

One of the principal difficulties in these epidemiological studies was determining the average WL in uranium mines over the two dec-

Fig. 1. Observed and expected annual lung cancer mortality per 10,000 mines and 95% confidence limits in relation to exposure (Ref. 26).

ades studied. However, measurements in 108 U. S. mines in 1956 and 231 mines in 1966 revealed the following distribution by WL ranges (Ref. 26):

WL Range	Fraction of U.S. mines in WL Range	
	1956	1966
Less than 1.0 WL	19%	44%
1.0 to 2.9 WL	25%	42%
3.0 to 4.9 WL	33%	9%
5.0 to 10.0 WL	33%	4%
Greater than 10.0 WL	23%	1%

In the mid-1950's a sizeable fraction of miners were exposed to atmospheres at cumulative levels in excess of 100 WLM per year. Substantial progress was subsequently made (and is still being made) in reducing WL values in mines as is seen in Table 4. For 1977, about 89% of uranium mines had concentrations less than 1 WL, with most of the remainder less than 2 WL.

Although the association between radon daughters and lung cancer was first identified in underground uranium miners, radon also occurs in other types of mines, and recent evidence (e.g., Ref. 33) links resulting exposures to increased incidence of lung cancer. Based upon such evidence, radiation protection guides have been established.

5. Radiation Protection Guides, Criteria, and Standards

In the United States, the WL and WLM concepts are used for exposure to radon and its daughters. (A limit for radon-222 gas itself is typically not considered separately, because of the general recognition that the impact of the radon daughters is the more important consideration.) Mining standards are promulgated under the authority granted to the Secretaries of Labor and of Health, Education, and Welfare (HEW) by the Federal Mine Safety and Health Amendments Act of 1977 (Public Law 95-164). The operating standards for mines are given in Title 30 of the Code of Federal Regulations Parts 55, 56, 57, and 58. Standards regarding radiation in underground mines are given by 30CFR57.5-35a through 30CFR57.5-47a. Table 5 paraphrases the current radiation standards. These standards, now considered the operational guidelines for exposure of miners to radon daughters, provide that exposure to radiation other than the inhalation of radon daughters must be considered separately. The usual occupational limits apply, that is, 5 rem/year for whole body external exposure, and so on. These limits have been discussed in detail in the introduction of this volume (see Section I.D).

Table 4. Radon daughter concentration and its exposure in uranium mines. (From EPA, Ref. 32.)

Year	Average Concentrations	Average Exposures
1975	0.71 WL	5.68 WLM
1976	0.58 WL	4.64 WLM
1977	0.51 WL	4.08 WLM

Table 5. Current radiation standards, underground only (February 17, 1978). (Ref. 6)

57.5-37	<u>Mandatory.</u> Mine atmospheres shall be sampled to determine if hazardous concentrations of radon daughters are present. Where potentially hazardous concentrations are found, or known sources of radon exist, each active work area shall be sampled as often as necessary by a qualified person.
57.5-38	<u>Mandatory.</u> No person shall be permitted to receive an exposure in excess of 4 WLM in any calendar year.
57.5-30	<u>Mandatory.</u> Except as provided by standard 57.5-5, persons shall not be exposed to air containing concentrations of radon daughters exceeding 1.0 WL in active workings.
57.5-40	<u>Mandatory.</u> (a) Where uranium is mined, if measurements in areas indicate exposure to concentrations of radon daughters in excess of 0.3 working level, complete individual exposure records shall be kept for all employees entering these areas. (b)...
57.5-41	<u>Mandatory.</u> Smoking shall be prohibited in all areas of a mine where exposure records are required to be kept in compliance with standard.
57.2-42	<u>Mandatory.</u> If levels of permissible exposures to concentrations of radon daughters different from those prescribed in 57.5-38 are recommended by the Environmental Protection Agency and approved by the President, no employee shall be permitted to receive exposures in excess of those levels after the effective dates established by the Agency.
57.5-44	<u>Mandatory.</u> The wearing of respirators approved for protection against radon daughters shall be required in environments exceeding 1.0 WL and respirator use shall be in compliance with standard 57.5-5.
57.5-45	<u>Mandatory.</u> Inactive workings, in which radon daughter concentrations are above 1.0 WL, shall be posted against unauthorized entry and designated by signs indicating them as areas in which approved respirators shall be worn.
57.5-46	<u>Mandatory.</u> Where radon daughter concentrations exceed 10 WL, respirator protection against radon gas shall be provided in addition to protection against radon daughters. Protection against radon gas shall be provided by supplied air devices or by face masks containing absorbent material capable of removing both the radon and its daughters.
57.5-47	<u>Mandatory.</u> Gamma radiation surveys shall be conducted annually in all underground mines where radioactive ores are mined. (a) Surveys shall be in accordance with American National Standard ANSI N13.8-1973, entitled "Radiation Protection in Uranium Mines", Section 14.1 page 12, which is hereby incorporated by reference and made a part hereof. (b) Where average gamma radiation measurements are in excess of 2.0 mR/hr in the working place, gamma radiation dosimeters shall be provided for all persons affected, and records of cumulative individual gamma radiation exposure shall be kept. (c) Annual individual gamma radiation exposure shall not exceed 5 Rems.

The International Commission on Radiological Protection in Publication 24 (Ref. 34) outlines the principles of monitoring and limitation of radiation exposure in uranium mines, based on the ICRP guide (Ref. 35) recommending that the annual average concentration of ^{222}Rn and its daughters should not exceed 30 pCi/l in occupational settings. This results in a little less than 4 WLM in a year. Quoting from ICRP Publication 24 (Ref. 34):

"For the purpose of radiation protection in mines, it is understood that this limit refers to ^{222}Rn in equilibrium with its short-lived daughters, where the daughters release, through their decay to ^{210}Pb , an alpha-energy equal to that released by the daughters of 30 pCi/l of radon in equilibrium with its daughters (3.9×10^4 MeV/l).

"The annual limit of exposure to radon daughters can therefore be expressed as a time integral of alpha-energy concentrations: $2000\text{h} \times 3.9 \times 10^4 \text{ MeV/l} = 7.8 \times 10^7 \text{ MeV h/l}$. An additional quarterly requirement is that the exposure in a quarter of a year should not exceed $3.9 \times 10^7 \text{ MeV h/l}$."

In addition, the ICRP limits the annual exposure to pure ^{222}Rn , namely to 6×10^6 pCi h/l, (Ref. 34). This corresponds to an average concentration of 3×10^3 pCi/l. This limit is applicable in situations where the potential alpha energy of the daughters is a small fraction of the value in equilibrium, such as may result from the use of respirators, and from air filtration or electrostatic precipitation of daughters. Also, the ICRP limits uranium ore dust; the operational limits employed are based on the total alpha activity of the long-lived nuclides present in the dust, under the assumption that these nuclides are all members of the ^{238}U series (excluding radon and its daughters) in secular equilibrium. It has taken into account the fact that only five-ninths of the relevant disintegrations are alpha. The annual operational limit for alpha activity in ore dust is 70 pCi h/l; the quarterly limit is 35 pCi h/l, (Ref. 32) corresponding to the average concentrations of 3.5×10^{-2} pCi/l and 1.75×10^{-2} pCi/l, respectively.

We note here that there is no explicit guideline for exposure of the general public to radon and its daughters. However, the ICRP has recommended in its general overview that individuals in the general public should be limited to exposures that will not exceed the risk of 0.5 rem whole body exposure in a year (Ref. 36).

The Environmental Protection Agency regulations (Ref. 37) requires that operations of mining and milling be conducted such that the

annual dose equivalent to any member of the public does not exceed 25 millirem (0.025 rem). The EPA further regulates the mining and milling industry by requiring that all operations which can possibly contaminate surface and/or ground water that is used for drinking water adhere to the standards set by the National Interim Primary Drinking Water Regulations for Radioactivity (Ref. 38). In order to provide for the regulation of uranium mill tailings under the Atomic Energy Act of 1954, the U. S. Congress passed the Uranium Mill Tailings Radiation Control Act of 1978 (Ref. 40). This Act charges the Environmental Protection Agency (EPA) with the promulgation of health and environmental standards for uranium mill tailings and the Nuclear Regulatory Commission with insuring proper waste management.

6. Safety and Monitoring Codes

The complexity of adequate radiation protection in uranium mines and mills has motivated several codes of practice, which we shall mention briefly here.

The U. S. Bureau of Mines has issued a two volume handbook (Ref. 41) which is a prime source of detailed recommendations on control of exposures in underground uranium mines. The main strengths of this compilation are detailed discussions of ventilation practices, mine planning, sampling procedures for measurement, and the Kusnetz method for measuring WL (see "Radon and Its Daughters", Section IV.E). This handbook is highly recommended to any reader interested in the fine points of exposure control.

The International Atomic Energy Agency (IAEA), jointly with the International Labor Office (ILO) has published a Code of Practice (Ref. 42), which outlines the general duties of mine employees and employers for radiation safety. It recommends a maximum permissible dose equivalent of 15 rem/year to the bronchial mucosa of the lung. It also describes procedures for medical surveillance; protective measures (ventilation, dust abatement, safety); and protection equipment. The IAEA/ILO Code is accompanied by a Technical Addendum giving detailed considerations for control of radon and daughters; for monitoring radiation in mines; and for radiation protection in uranium and thorium mines and mills.

In addition the IAEA and ILO have issued a manual (Ref. 43) for the guidance of those responsible for the regulation, management, and operation of uranium and thorium mines and mills. The manual (IAEA Safety Series No. 43) follows and supplements IAEA Safety Series No. 26, and is intended as an operating manual with particular emphasis on implementation methods. The scope of the manual is limited to occupational hazards in mining and milling,

but includes a section in which liquid, solid, and gaseous wastes produced by the various processes are identified although not treated in detail. For the monitoring and control of such radioactive wastes, the reader is referred to the IAEA Code of Practice on the Management of Wastes from the Mining and Milling of Uranium and Thorium Ores, the first part of IAEA Safety Series 44 (Ref. 44). The Code of Practice identifies wastes, presents environmental problems and sets out minimum requirements for their competent management. The second part of Ref. 44 is a Guide to the Code, indicating ways in which the requirements of the Code may be met.

Another code of practice, ANSI N7.1, was published in 1960 by the American National Standards Institute, (Ref. 45) with a revision, ANSI N13.1, in 1972 (Ref. 46). ANSI N7.1 contains details of air and breath sampling protective measures, and medical procedures. The 1972 revision contains more recent techniques

and suggestions for ventilation, air sampling patterns, record keeping, and control responsibilities. Most important, it contains suggested procedures for measuring gamma radiation, radon gas and radon daughters.

Another code of practice is the 1957 U. S. Public Health Service Publication 494 (Ref. 47). Also, the 1963 IAEA Symposium (Ref. 48) contains much information on control and monitoring procedures.

The Nuclear Regulatory Commission has issued a guide (Ref. 49) describing programs it finds acceptable for measuring, evaluating, and reporting releases of radioactive materials in liquid and airborne effluents from typical uranium mills. The radiological effluent monitoring program is summarized in Table 6. The NRC has also released a guide (Ref. 50) useful in the preparation of environmental reports for uranium mills.

Table 6. Summary of proposed NRC radiological effluent monitoring program for uranium mills. (From Hickey, Ref. 51, with permission by American Nuclear Society © 1977.)

Sampling Location and Frequency	Type of Analysis
A. AIRBORNE	
Site Boundary, continuous	U-natural, ^{230}Th , ^{226}Ra , ^{222}Rn , ^{210}Pb
Drier Stack, Semiannual	U-natural, ^{230}Th , ^{226}Ra
Other Stacks, Semiannual	U-natural
B. LIQUID	
Discharge Point, Continuous	U-natural, ^{230}Th , ^{226}Ra
Wells, Quarterly	U-natural, ^{230}Th , ^{226}Ra
On-site Water Bodies, Quarterly	U-natural, ^{230}Th , ^{226}Ra
Tailings Seepage, Quarterly	U-natural, ^{230}Th , ^{226}Ra

7. Instrumentation Requirements

There are five different classes of measurements within the uranium mining and milling industry:

i. Measurements of Working Levels, to ensure the radiological protection of miners.

ii. Measurements of the individual radionuclides (radon-222, and each individual daughter separately) by ventilation engineers and safety officers to aid in the design of adequate safety systems. Also in this class is the determination of the uncombined fraction of Radium-A. None of these measurements has become widely used in the uranium industry.

iii. Measurements of other sources of occupational radiation exposure. These include external gamma radiation, particulate airborne radionuclides, thorium, uranium, and radium. At present, these latter measurements are not widely performed.

In addition, there are two other classes of measurements:

iv. Measurements of bio-assay media (for example, urinalysis). These can be valuable indicators of radionuclide body burdens. They are principally used at mills, and much less often at mines.

v. Measurements of the radiological impact of mines and mills on the environment. These include measurements in waste waters; in ambient air near mill tailings; in soil samples; and in plants and animals.

The uranium mills' environmental radiological impact can be important, while that of uranium mines is nearly always minor by comparison.

We shall discuss these categories separately below, outlining the requirements on instrumentation.

a. Measurements of Working Level. The most obvious requirement for occupational protection is the measurement of Working Level. Indeed, as mentioned above, the original choice of WL as a measure of exposure was largely operational.

Direct measurements of WL itself can now be made by the well-established but time-consuming and approximate Kusnetz method; by the Tsivoglou method; and by more rapid but less accurate methods using Instant Working Level Meters. In addition, when thoron is present, the potential alpha energy (working levels) from thoron daughters should be determined by methods outlined in a publication (Ref. 52) by the Mining Enforcement and Safety Administration (predecessor of MSHA) (Ref. 52). In

mines containing thorium, its daughters will be present, including radon-220, comparable to radon-222, the distant daughter of uranium-238. These are all discussed in "Radon and Its Daughters".

The instrumentation needed for the proper measurement of WL should be sensitive down to at least 0.03 WL (10% of the present average guidelines of 0.33 WL), and be capable of measuring WL in the range of about 0.1 to 1.0 WL with accuracies of ± 10 to $\pm 20\%$. It should be reliable in the working conditions found in mines. This implies ruggedness, convenience, portability, and small size and weight. When determining exposure, a rapid readout is not important whereas for ventilation engineers it can be important (the Kusnetz method requires 45 minutes, start to finish).

Several devices have been proposed (Ref. 53) for monitoring individual exposure of persons to radon and, more importantly, to radon progeny attached to aerosol particles. A working level dosimeter has been developed and tested (Ref. 54). Its response is linear over several decades of cumulative working level exposure and the limit of sensitivity is about 1.4 WL-h as long as controlled conditions exist; in a mine environment trouble with the operation of the pump exists. Several "instant working level meters" are available commercially (e.g. MDA) or are in various stages of development (e.g., Kerr-McGee, Argonne National Laboratory).

b. Measurements of Individual Radionuclides. Another class of measurements is the separate determination of individual radionuclide activities in the radon chain; principally, radon-222; polonium-218; lead-214; and bismuth-214. If simple instrumentation for these were available it would enable industrial hygienists and ventilation engineers to assess the extent to which a particular atmosphere is in equilibrium.

There are several methods for measuring *radon-222 activity*. Two common methods are by using a "Lucas chamber" or by the "two-filter" method, but neither is entirely satisfactory. Sensitivities are adequate (better than 10 pCi/liter in each case) but each suffers from some drawbacks, and a truly satisfactory radon gas monitor is still required and unavailable.

Measurements of various *individual radon daughters* are desirable, and are possible. Very good sensitivity for the daughters of radon can be obtained by the Thomas modification of the Tsivoglou method (Ref. 55). Sensitivities as low as pCi/l for each daughter can be obtained (Ref. 56). Unfortunately, measurements cannot now be made with simple, portable instruments. In any case the desired instruments should be capable of measuring individual daughter activities, at levels in

the range of 10 pCi/liter. Use of air filters and alpha spectroscopy is possible but difficult. All of this is discussed in "Radon and Its Daughters" (Section IV.E)

c. Measurements of Other Occupational Exposures. There are several other occupational measurement problems besides the measurement of radon and its daughters. In most cases, instruments adequate for this occupational radiological surveillance are available. This includes:

(1) *External beta and gamma radiation.* These can be adequately measured with personnel dosimeters or hand-held meters described elsewhere in this volume (see "Personnel Dosimetry," "Combination Instrumentation," "X- and Gamma Radiation," and "Beta Particle Instrumentation"). Gamma measurements are required by 30 CFR 57.5-37 for underground measurements only. At the present time, there are no surface standards, but work is in progress for developing them (Ref. 6).

(2) *Beta radiation in water.* See "Beta Particle Instrumentation" for discussions of these measurements, which, if required (mainly at mills) can now be adequately performed using any of a variety of instruments.

(3) *Uranium and thorium.* The measurement of these nuclides in air and water is discussed in the section on "Uranium". Adequate instrumentation exists for most of the measurements required to monitor occupational exposures, which occur mainly at uranium mills.

d. Measurements in Biomedical Media. One common approach to measuring occupational uptake of radionuclides is bioassay. The most common procedure is *urinalysis*; techniques have been developed for radioactive uranium, radium, polonium, and lead in urine (Ref. 48). Other possible measurements include analysis of exhaled breath for radon and tissue assays for radium and thorium (Ref. 57).

Typical specific activities requiring measurement are small. For example, workers exposed to 20-30 WL for about 3 years exhibited 5-10 pCi of lead-210 (RaD) per 24-hour urine sample (Ref. 58). In Japanese workers chronically exposed to 1-10 WL (Ref. 59) levels of urinary polonium-210 (RaF) were in the range of about 0.05-0.5 pCi/liter. Another study (Ref. 60) indicates that exposure at 1 WL results in polonium-210 urinary excretions of about 0.25 pCi/day. Uranium concentrations in urine at levels near 100 µg/liter were found in still another study (Ref. 61). It needs to be pointed out that correlations of WL with ²¹⁰Pb and ²¹⁰Po in urine are difficult to interpret if there is ignorance of the concentrations of ²¹⁰Pb and ²¹⁰Po in the ambient air.

Other media which have been studied include blood, hair, and whiskers. However, none of these shows enough quantitative correlation to be used for true monitoring purposes: from measurement of ²¹⁰Pb or ²¹⁰Po in these media, predictions of mean WL exposure are accurate only in the ±50% range at best (Ref. 62). Fortunately, this is adequate. However errors almost as large are also found for urinalysis correlation with WL. The activity levels found in whiskers were a few pCi of RaD per gram, for current exposure in the range of 0.2-0.7 WL (Ref. 62).

Measurement techniques for these small bioassay activity levels are usually wet-chemical analytical methods. These are typically quite sensitive, but are expensive, susceptible to analytical errors, and are very laborious. They are discussed in Volume 4 ("Biomedical") of this Survey. The NRC has issued a regulatory guide (Ref. 63) that stipulates an approved bioassay method for uranium mill workers.

e. Measurements in Environmental Media. Uranium mining and milling activities may produce measurable radiological impacts on the immediate vicinity. Among the measurements required are the following:

(1) Measurements of radon gas exhaled from mill tailings, exposed ore bodies, and occasionally from concentrated (processed) uranium products. Because radon gas is constantly being produced by decay from the parent chain, its activity level must be evaluated periodically. Radon measurements in the range of a few tenths to a few hundred pCi/liter of air are typically required.

(2) Measurements of radium, uranium, and thorium in surface water, ground water, waste water and ores, and mill tailings. The processing of ore may augment the naturally-occurring levels of these radionuclides in local water. Both the short-term and long-term effects must be quantified. In particular, leaching from long-standing tailings can sometimes produce significant contamination in waters whose direct contamination is otherwise small. Typical natural radium-226 concentrations in surface waters near uranium-rich regions range from 0.05 to 0.3 pCi/liter (Ref. 64). Natural uranium content is the range of 3 to 10 µg/liter (approximately 2 to 6 pCi/liter); its contribution to gross-alpha levels is typically ten times greater than that of radium-226. Sill has developed a method for the determination of thorium and uranium isotopes in ores and mill tailings (Ref. 65).

(3) It has become known that the long-lived α-emitters contained in airborne particulates can present a potential exposure. Some specific radionuclides that should be measured are ²²⁶Ra, ²³⁰Th, ²³⁸U.

Recently, Kordas and Phelps reviewed the measurement techniques for such radionuclides (Ref. 66). Yule has developed an on-line monitor capable of detecting less than one maximum permissible airborne concentration, MPCa, of these hazardous aerosols in less than one-half hour (Ref. 67).

Unfortunately these techniques involve bulky apparatus and are not simple portable instruments.

(4) Measurements in biological media. These include aquatic biosystems, local grasses and other flora, animal species, and soils. These media can be indicators of environmental reconcentration mechanisms. They are performed around mills more often than around mines, and even then the measurements are now usual for research rather than for routine surveillance. Typical radium activities in soil of 0.1 to 5 pCi/g occur naturally in uranium areas, so measurement techniques must be sensitive to at least that level (Ref. 38). Reconcentrations of radium have been studied in a number of aquatic biota, and reconcentration factors (activity in medium/activity in water) of 500 to 1000 have been found in algae as aquatic insects, with a factor of 3 typical for fish flesh and 100 typical for fish bone (Ref. 64). The reader is advised to consult LBL-1, Volume 4, Bio-medical for procedures.

8. Summary and Conclusions

The purpose of this section has been to outline (1) the extent of the radiological hazard and environmental impact resulting from the uranium mining and milling industry; (2) the radiation protection guides and codes of practice developed in response to this result and (3) some (brief) discussion of the instrumentation needed to monitor the radioactive pollution. This section is partly a companion to the section "Radon and Its Daughters" (elsewhere in this volume), in which instrumentation is discussed in detail, (other sections are also mentioned above.)

The epidemiological situation for uranium miners can be summarized briefly by noting that there is no real doubt as to the problem; that exposures as large as those commonly experienced decades ago will certainly not be allowed to recur; and that present standards and radiation protection guides seem to be adequate for the protection of the miners, if properly applied.

The monitoring situation around uranium mines can be summarized as follows: Through research and development that has been going on, a more sophisticated approach to the entire measurement problem has become available. Such an approach emphasizes three areas: automated measuring instruments with direct readout and recording capability; improved portable survey instruments; and sensitive personnel dosimeters.

At uranium mills, the principal measurement problems concern occupational exposures (bioassay measurements and external-exposure data) and environmental impacts. Both bioassay procedures and radionuclide measurements in the environment are in need of further development, so that their use can become more widespread without undue economic impact. Especially in the case of radionuclide in the environment, the need is for measurement techniques for routine use in surveillance... today such measurements are still largely confined to research studies.

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F. Natural Background Radiation and Fallout

1. Introduction

This section briefly summarizes background doses contributed from natural sources, including those whose exposure is enhanced by human activities, and from fallout from nuclear explosions. The section first characterizes average "natural background", i.e., dose levels from radiation that is neither manmade nor greatly affected by technology. The section next indicates how the dose from natural sources may be "technologically enhanced" by human endeavors. Finally, the dose from fallout is discussed. These questions are not treated in great detail in this section; some are treated at greater length elsewhere in this volume (see "uranium mining and milling", Section II.E, and "radon," Section IV.C).

In recent years, several publications have undertaken to collect information on natural background, fallout, and - in some cases - technological enhancement. For worldwide data, the 1977 report of the United Nations Scientific Committee on the Effects of Atomic Radiation, entitled "Sources and Effects of Ionizing Radiation" (Ref. 1), is most useful. The National Council on Radiation Protection and Measurements has recently published a document on "Natural Background Radiation in the United States" (Ref. 2). The Environmental Protection Agency has recently published annual reports on the "Radiological Quality of the Environment" (Ref. 3), treating all three of the areas summarized in this section. For many years, the Department of Energy's Environmental Measurements Laboratory (previously the Health and Safety Laboratory) has been actively involved in environmental radiation measurements, including those of the natural radiation environment (Ref. 4) and of fallout (issued quarterly). Finally, a very extensive source of information has been a series of symposia on "The Natural Radiation Environment", held in Houston, 1963,⁵ 1972,⁶ and 1978.

2. Natural Background Radiation

Cosmic radiation and terrestrial radioactivity provide a background of radioactivity and resulting exposures that exist largely independently of human activities and that must be considered if measurements of human-caused radiation are to be performed or their significance understood. It is useful to divide the natural human radiation doses into three parts: that caused by direct exposure to cosmic rays (cosmic radiation); that arising from terrestrial radionuclides external to the body (primarily gamma radiation); and that caused by materials inside the body (internal exposure).

a. Cosmic Radiation. Cosmic radiation arises mostly from galactic sources (but

may also refer to high energy solar radiation) and is taken to include both the energetic primary particles impinging on the earth's atmosphere and secondary particles generated by resulting interactions. The galactic radiation at the top of the atmosphere is composed largely of protons (87%) and alpha particles (11%), with about 1% each of heavier nuclei and electrons. Energies range at least to 10^{20} eV, but the bulk are at lower energies (10^8 to 10^{11} eV). The spectrum falls rapidly with increasing energy, roughly as $1/E^2$ for the higher energies. At the lower end of this energy range (esp. 10^6 to 10^{11} eV) galactic radiation may be augmented substantially by particles ejected from sun during periods of flare activity (Ref. 7).

The cosmic radiation field at ground level is almost entirely secondary particles from highly energetic ($> 10^{10}$ eV) primary particles from galactic sources. The flux near the surface is predominantly muons from the decay of pions produced in the upper atmosphere; these muons are accompanied by an equilibrium mixture of electrons and photons. In addition, there is a small flux of neutrons and protons (Ref. 7).

As indicated in Fig. 1 (Ref. 2), the dose equivalent at ground level is dominated by muon exposure. However, the dose rate increases rapidly with altitude, as does the relative contribution of various components of the flux. In addition, at higher altitudes, there is a significant dependence on latitude,

Fig. 1. Absorbed dose rates at 5 cm depth in a 30 cm thick slab of tissue, from various components of the cosmic radiation at solar minimum and at 55°N. (From NCRP, Ref. 2, with permission by National Council on Radiation Protection and Measurements © 1976.)

*Sometimes reference numbers appear as

due to the earth's magnetic field. At ground level, this dependence is much reduced, since the dose there arises largely from highly energetic particles that are little affected by the earth's field. Such data indicate a dose equivalent from cosmic radiation at sea level of about 28 mrem/yr.¹⁻² Considering the altitudinal population distribution,⁸ as well as shielding by structures (assigned an average decrease of 10%), the average U. S. population dose is approximately 28 mrem/yr.² This is somewhat smaller than doses cited by the Environmental Protection Agency: Ref. 3 cites estimated doses varying from 35 to 120 mrem/yr; the estimates in Ref. 9 vary from about 40 to 80 mrem/yr from one state to another. (The average elevation of U. S. residents is approximately 200 m.⁸). Near sea level, a kilometer increase in elevation corresponds to an increase in dose equivalent of approximately 10 mrem/yr.² However, at high altitudes, the flux increases more rapidly and the component with higher quality factor increases, so that at altitudes above 10 km, the dose equivalent rates are orders of magnitude higher than at sea level. At the altitude of jet flights (about 11 km), the dose rate is approximately 0.5 mrem per hour, so that doses of a few mrem could be sustained from long distance flights. As noted above, the dose at high altitudes is also more dependent on latitude and the level of solar activity; see Ref. 2 for further discussion.

b. External Terrestrial Radiation.

Naturally occurring radionuclides produce external gamma exposures (the principal topic of this discussion) as well as much smaller beta exposures, and negligible external alpha exposures. The dominant contributors to the gamma field are potassium-40, thorium-232 and its daughters, and uranium-238 and its daughters, all of which are present in soil, rocks, and other earth materials. The distribution of radionuclides in the United States is discussed in Ref. 2. Methods for calculating gamma fields and dose rates are set forth in Ref. 10. Much of the dose from the uranium and thorium series arises from radon and its daughters.

External dose estimates for the United States are summarized in Refs. 2-3. The data from Ref. 8 indicate a population averaged outdoor absorbed air dose of 40 mrad/yr. To obtain average dose equivalent requires consideration of screening by structures and tissue, yielding a dose of about 26 mrem/yr. However, most cited values do not appear to apply this correction. An earlier estimate¹¹ gave a range of 30 to 95 mrem/yr for 90% of the U. S. population, with an average of 55 mrem/yr, divided into 17, 25, and 13 mrem/yr for ⁴⁰K, the ²³²Th series, and the ²³⁸U series respectively. A similar range was given by Ref. 8, varying - for the sites examined - from about 23 mrad/yr for the coastal

plain to about 90 mrad/yr for the Colorado Plateau area (Denver). Worldwide external dose estimates are summarized in Ref. 1, which states an estimated average absorbed dose of 32 mrad/yr.

As indicated, the radiation field depends on structural shielding. The degree of shielding (or increase, in some cases) depends on the structure type. In addition, soil and meteorological conditions affect the outdoor levels. In particular, moisture in the soil shields gammas emitted from radionuclides in the soil and affects the migration of radon from the soil. See summary discussion in Ref. 2.

c. Internal Exposures. Radionuclides present in the environment may be ingested or inhaled, leading to deposition at various sites in the body. Radiation from these internally deposited emitters constitutes a significant portion of natural background exposure. The radionuclides of particular importance are ⁴⁰K, ²¹⁰Po, ²²²Rn, and ^{226,228}Ra. Carbon-14 (produced by cosmic rays) and ⁸⁷Rb are also contributors. The heavy isotopes, all from the uranium-238 or thorium-232 series, are alpha-emitters that - because of their deposition sites - expose bone cells. The lighter isotopes, gamma or beta emitters, distribute their dose more uniformly - Tables 1 and 2 (taken from Ref. 2) summarize the dose equivalent from the important internally deposited radionuclides. Reference 2 discusses the pathways for deposition of these radionuclides. See also Refs. 1 and 3. Table 3 shows doses (in mrad/yr) of internally deposited radionuclides as estimated in Ref. 1.

Note that, although the dose (in mrad/yr) is dominated by gamma rays, beta rays, and ionizing cosmic rays, the alpha dose can make an important contribution to the dose equivalent (in mrem/yr) for lung and bone, especially for exposure due to radon daughters. Concentrations of airborne radioactivity, including radon and daughters, exhibit substantial variations with time, place, and meteorological conditions. The average dose to lung is about 100 mrem/yr², (largely arising from the radon daughter ²¹⁸Po) for typical concentrations in the United States. For more detailed discussion of radon and daughters, see Section IV.E on radon. It is also important to note that increased exposure to radon and daughters can be the most significant radiological result (aside from medical procedures) of routine human activities; see part 3 of this section, as well as the mining and milling section.

d. Summary of Natural Background Exposures. Based on the information indicated above, estimates may be made of total natural background exposures. Table 4 is a summary of average U. S. doses given in Ref. 2. The indicated whole body dose of 80 mrem/yr arises

Table 1. Annual internal beta and gamma dose equivalent rates (mrem/y) in tissue from internally deposited naturally occurring radionuclides^a. (From NCRP, Ref. 2, with permission by National Council on Radiation Protection and Measurements © 1976.)

Radionuclide	Soft tissues (Gonads)	Cortical bone		Trabecular bone	
		Osteocytes	Haversian canals	Surfaces ^b	Marrow
³ H	~0.001	~0.001	~0.001	~0.001	~0.001
¹⁴ C	0.7	0.8	0.8	0.8	0.7
⁴⁰ K	19	6	6	15	15
⁸⁷ Rb	0.3	0.4	0.4	0.6	0.6
Total	20.0	7.2	7.2	16.4	16.3

^aUNSCEAR (1972) gives the data as absorbed dose in tissue in mrad/y (Ref. 25)

^bCells close to surfaces of bone trabeculae.

Table 2. Annual alpha dose equivalent rates (mrem/y) from naturally occurring radionuclides. (From NCRP, Ref. 2, with permission by National Council on Radiation Protection and Measurements © 1976.)

Radionuclide	Concentration		Dose equivalent rates				
			Cortical bone			Trabecular bone	
	in air	in bone ^a	Gonads	Osteocytes	Haversian canals ^b	Surfaces ^c	Marrow
	(pCi/m ³)	(pCi/kg)					
²³⁸⁻²³⁴ U ^d	—	6.9	0.8	12.4	7.7	4.8	0.9
²³⁶ Ra ^d	—	7.8	0.2	16.4	10.2	6.6	1.2
²²⁸ Ra ^d	—	3.8	0.3	19.0	11.0	8.0	1.0
²²³ Rn ^e	150	—	0.4	0.2	0.2	0.4	0.4
²²⁰ Rn ^e	1	—	0.01	0.1	0.1	0.2	0.2
²¹⁰ Po ^d	—	60	6	60	36	24	4.8
Total			8	110	65	44	8.5

^aThe alpha-emitting nuclides are assumed to be uniformly distributed in mineral bone, although this may not be the case (ICRP, 1968).

^bCells lining the Haversian canals.

^cCells close to surfaces of bone trabeculae (dose) averaged over the first 10 μm.

^dCalculated by the method of Spiers (1968).

^eDerived from UNSCEAR (1972).

Table 3. Estimated annual tissue absorbed dose from natural sources in "normal" areas.
(From UNSCEAR, Ref. 1, with permission by United Nations © 1977).

	Annual tissue absorbed dose (mrad)						
Source of irradiation	Gonads		Lung	Bone lining cells		Red bone marrow	
External irradiation							
Cosmic rays:							
Ionizing component	28	(28)	28	28	(28)	28	(28)
Neutron component	0.35	(0.35)	0.35	0.35	(0.35)	0.35	(0.35)
Terrestrial radiation: (γ)	32	(44)	32	32	(44)	32	(44)
Internal irradiation							
Cosmogenic radionuclides:							
^3H (β)	0.001	(0.001)	0.001	0.001	(0.001)	0.001	(0.001)
^7Be (γ)	—	—	0.002	—	—	—	—
^{14}C (β)	0.5	(0.7)	0.6	2.0	(0.8)	2.2	(0.7)
^{22}Na ($\beta^+\gamma$)	0.02	—	0.02	0.02	—	0.02	—
Primordial radionuclides:							
^{40}K ($\beta^+\gamma$)	15	(19)	17	15	(15)	27	(15)
^{87}Rb (β)	0.8	(0.3)	0.4	0.9	(0.6)	0.4	(0.6)
^{238}U - ^{234}U (α)	0.04	(0.03)	0.04	0.3	(0.3)	0.07	(0.06)
^{230}Th (α)	0.004	—	0.04	0.8	—	0.05	—
^{226}Ra - ^{214}Po (α)	0.03	(0.02)	0.03	0.7	(0.6)	0.1	(0.1)
^{219}Pb - ^{210}Po ($\alpha^+\beta$)	0.6	(0.6)	0.3	3.4	(1.6)	0.9	(0.3)
^{222}Rn - ^{214}Po (α) inhalation	0.2	(0.07)	30	0.3	(0.08)	0.3	(0.08)
^{232}Th (α)	0.004	—	0.04	0.3	—	0.04	—
^{228}Ra - ^{208}Tl (α)	0.06	(0.03)	0.06	1.1	(0.3)	0.2	(0.06)
^{220}Rn - ^{208}Tl (α) inhalation	0.008	(0.003)	4	0.1	(0.05)	0.1	(0.05)
Total (rounded)	78	(93)	110	86	(92)	92	(89)
Fraction of absorbed doses delivered by alpha particles or neutrons (%)	1.2	(1.3)	31	8.5	(4.1)	2.1	(1.2)

Table 4. Summary of average dose equivalent rates (mrem/y) from various sources of natural background radiation in the United States. (From NCRP, Ref. 2, with permission by National Council on Radiation Protection and Measurements © 1976.)

Source	Gonads	Lung	Bone		G.I. Tract
			Surfaces	Marrow	
Cosmic Radiation ^a	28	28	28	28	28
Cosmogenic Radionuclides	0.7	0.7	0.8	0.7	0.7
External Terrestrial ^b	26	26	26	26	26
Inhaled Radionuclides ^c	—	100 ^d	—	—	—
Radionuclides in the Body ^e	27	24	60	24	24 ^f
Rounded Totals	80	180	120	80	

^a"Cosmic Radiation" includes 10% reduction to account for structure shielding.

^b"External Terrestrial" includes 20% reduction for shielding by housing and 20% reduction for shielding by the body.

^cDoses to organs other than lung included in "Radionuclides in the Body."

^dLocal dose equivalent rate to segmental bronchioles is 450 mrem/y.

^eExcluding the cosmogenic contribution shown separately.

^fThis does not include any contribution from radionuclides in the gut contents.

Reproduced from Ref. 2 (p. 108).

Fig. 2. Ground level inventory of ⁹⁰Sr in late 1959. (From Kulp, Ref. 21, with permission by Elsevier-North Holland Biomedical Press, Amsterdam © 1965.)

from almost equal contributions from cosmic radiation, external terrestrial radiation, and internally deposited radionuclides. Bone sur-

faces and lungs receive substantial additional doses from internal alpha emitters. However, as indicated above, estimates vary substantially and also have large geographical and temporal dependences. Estimates from Ref. 1 were given in Table 3 (stated however, in mrad/yr).

3. Technologically Enhanced Natural Radiation

Human activities can alter the degree to which people are exposed to naturally occurring radiation. These include not only structures, which - as noted above - can even decrease exposure to external cosmic or terrestrial radiation, but also more specific industrial activities, particularly those for energy production. The potential for increasing exposure to natural radiation arises largely from radon (see Section IV.E, "radon") generated in earth materials with which humans work or live. This section very briefly indicates several of the principal possibilities for increased exposure, many of which are discussed in Ref. 3 as they pertain to the United States. The reader is also referred to Ref. 1 for a survey of worldwide information. As noted in Section 2, exposure to cosmic rays can also be increased

technologically, i.e., by jet flights which for long distances-contribute doses of few mrem.

a. Uranium Mining and Milling.

Occupational exposure of uranium miners has given one of the most unambiguous examples of the effects of elevated exposure to radioactivity, in this case radon daughters. Exposures at mines and mills, as well as control methods and monitoring instrumentation, are discussed in the section on uranium mining and milling.

These activities also subject members of the general public to increased radiation exposures, largely through the radon emanating from tailings piles associated with milling operations. The most notorious increased exposures have arisen from cases where tailings were used in landfill operations. However, even in ordinary circumstances, some portion of the short-lived radon may reach human occupied areas, even at a distance. Because this radon ultimately arises from a long-lived parent (^{230}Th , half-life 80,000 years), inactive tailings piles can constitute a low-level, but very long-lived source of general population exposures. The integrated population dose can exceed other exposures resulting from the nuclear industry (with the possible exception of ^{14}C , which also results in a long-term dose commitment). See, for example, estimated doses in Ref. 12. Reference 1 also gives a brief summary of the major considerations as well as doses and dose estimates from active and inactive mining and milling operations.

b. Non-Nuclear Industries. Uranium, thorium, and their daughters occur in phosphate ores. Substantial attention has recently been drawn to the phosphate industry because of resulting occupational and general public exposures, particularly to radon and daughters. Exposures may occur as a direct result of mining, from the processing plant, or from disposition of the product or tailings. Thus workers are subject to some increased exposure, possibly comparable¹³ to dose limitations for radiation workers. In addition, radioactivity may be released to the air (see for example, Ref. 14), subjecting members of the general public to slightly increased radiation exposures. Exposures may also arise from use of by-products¹⁵ or from construction of residences on reclaimed land.¹⁶ The last point arises particularly in cases where the structure retains radon (see below). See Ref. 3 for further discussion of radiological questions associated with the phosphate industry.

Non-nuclear energy industries can also release naturally-occurring radionuclides into the human environment. For example, fossil-fuel fired power plants can release the trace amounts of radium and thorium present in

these energy resources. Amounts released from coal-fired plants without particulate control are in the range of curies per gigawatt-year; amounts from oil-fired plants are somewhat less. Geothermal plants release radon that has diffused into geothermal resources within the earth's crust and the amounts vary substantially with the resource that is developed. Reference 7 contains a brief discussion of radionuclides from fossil-fuel and geothermal power generation. See Ref. 3 for preliminary dose estimates from coal-fired plants. In a simple model for coal-fired power plants without good particulate control, Ref. 1 estimates dose commitments of 10^4 man-rad per gigawatt-year from inhalation directly from the passing cloud in a region with a uniform population density of 100 per km^2 .

c. Buildings. Residential, commercial, or industrial structures can provide significant shielding from external radiation sources. However, particular types of material may themselves constitute significant sources of external radiation and of radon and daughters. Moreover, depending on ventilation rates, buildings can effectively concentrate levels of pollutants that arise from sources inside the buildings, including radon and daughters. These radionuclides may arise from building materials such as concrete, from the soil on which the building is constructed, and - in some cases - from water or gas utility systems. Reference 2 contains a brief survey of results on indoor radon concentrations, and it is clear that these concentrations can be substantially greater than outdoor levels. This question may achieve wider significance as energy conservation measures that would reduce air infiltration or ventilation rates are implemented on a large scale.¹⁸ The question of radon levels in buildings has already received wide attention in connection with the uranium and phosphate industries (see above).

4. Fallout

In this section, we discuss briefly the various types of radioactive fallout, and their relative intensities and concentrations as a function of time. This section is meant primarily as background information on fallout. The techniques and problems of fallout measurement will not be discussed in detail.

The term "fallout" describes the radioactive component of the debris resulting from atmospheric nuclear (fission) and thermonuclear (fusion) explosions. Since the 1940's, the U. S. has maintained a nation-wide (and with other nations a world-wide) fallout-measurement program. The U.S.A.E.C. maintained both a Fallout Studies Branch and a Fallout Program at the Health and Safety Laboratory (HASL) in New York (Ref. 19), now the Department of Energy's Environmental Measurements

Fig. 3. Monthly deposition of ^{90}Sr in New York City.

Laboratory. From these sources, from Fowler's compilation (Ref. 20), and from Ref. 1, a comprehensive picture of the fallout situation can be obtained.

Distinctions have generally been made between three different kinds of fallout. Local fallout, confined to the vicinity of a detonation, usually occurs within about the first 48 hours. Tropospheric fallout, occurring in a band around the world near the general latitude of the detonation, occurs in the first 2 to 4 weeks. Finally, stratospheric fallout occurs over a longer period. Small weapons contribute all of their fission products to local and tropospheric fallout; the truly world-wide distributions come from stratospheric fallout after the largest (> 1 megaton) detonations. Only these large weapons are capable of depositing significant debris in the stratosphere. Once products are carried into the stratosphere, they become widely dispersed and return to earth over periods of from 3 months to several years. Since ~ 90% of all weapons-test fission products has come from multimegaton weapons, stratospheric fallout is the dominant vehicle for world-wide contamination.

From a biological standpoint, the most important long-term fallout constituents are ^{90}Sr and ^{137}Cs , because of their long half-lives (28 and 30 years, respectively) and relatively high uptake and damage in bio-systems. Strontium is chemically similar to calcium, which is used for bone formation, while cesium simulates potassium, crucial for the function of cells. Total weapons-tests production to date of ^{90}Sr and ^{137}Cs has been about 1 million curies (MCi), respectively. The activity is not distributed uniformly in space or time after a detonation. In addition (from Ref. 21) the latitudinal variation of ^{90}Sr surface deposition levels is shown in Fig. 4. The large peak in the northern temperate zone and the secondary peak in the southern temperate zone are due to the dynamics of stratospheric-tropospheric mixing: the poorly defined or discontinuous tropopause in the temperate latitudes provides a vehicle for preferential stratospheric deposition.

The time distribution is shown in Fig. 3 for ^{90}Sr depositions in New York City (Ref. 22). The time distribution shows the very obvious effects of the large-scale weapons tests of the early 1960's. The ($^{137}\text{Cs}/^{90}\text{Sr}$) ratio is found to be between 1.5 to 2.0 for almost all measurements (Ref. 23).

Human exposure to fallout is from three sources: ingestion, inhalation, and whole-body external exposure. Of the nuclides ingested, the main contributors to dose are ^{90}Sr and ^{89}Sr (to whole-body and to bone); ^{131}I (to thyroid); ^{137}Cs (to whole-body); and ^{14}C (to whole body and to bone). Because strontium is deposited permanently in bone, the ^{90}Sr doses have been estimated based upon a 50-year (or lifetime) delivery of dose. All other doses are assumed to be delivered in the year of ingestion; Table 5 (from Ref. 24) shows the 1963, 1965, and 1969 estimated annual fallout dose per capita from ingestion. Figure 4 (from Ref. 22) shows semi-annual averages of the ^{90}Sr intake in New York City from 1961 to 1970.

The estimated dose from inhalation is also shown in Table 5. Note that, while the whole-body dose from inhalation is quite small, the 50-year respiratory lymph-node dose from plutonium is large indeed.

For external gamma dose rates, the dominant contributor was ^{137}Cs . Including shielding factors (0.4 due to buildings, 0.8 due to body screening), the 1963, 1965 and 1969 estimated annual average external gamma doses (Ref. 24) were 5.9, 1.8 and 0.9 mrem/capita.

In recent years, the average radiation dose from fallout has been about 4 mrem/yr, considerably smaller than during the mid-1960's a period of heavy weapons testing - see Table 5.

Table 6, taken from Ref. 1, shows the dose commitment arising from radionuclides released in testing before 1976. As for dose rate, the long range dose commitment is dominated by exposures from ^{90}Sr and ^{137}Cs . However, short-lived nuclides also made a substantial contribution. Moreover, were the ^{14}C contribution to be integrated over its long half-life, the contribution from ^{14}C alone

Table 5. Estimated average U.S. doses from fallout (mrem per capita per year), (except [*] which is 50 year dose from nuclide inhaled or ingested in that year). (Ref. 24).

Source	Organ	1963	1965	1969
<u>Ingestion</u>				
^{14}C	whole-body	0.3	0.7	0.6
^{89}Sr	bone	1.3	0.2	0.2
^{90}Sr	whole-body	73.0*	72.0*	32.0*
^{90}Sr	bone	7.3	7.2	3.2
^{131}I	thyroid	25.0	4.0	3.0
^{137}Cs	whole-body	4.3	2.3	0.4
<u>Inhalation</u>				
^{144}Ce	lung	8.2	0.4	0.3
^{144}Ce	bone	10.6	0.5	0.4
^{238}Pu	respiratory lymph nodes	79.0*	14.0*	19.0*
^{239}Pu	respiratory lymph nodes	1000.0*	175.0*	36.0*
^{239}Pu	lung	4.5*	0.8*	0.2*
Several others	lung	4.3	0.4	0.1
<u>External Gamma Dose</u>				
^{137}Cs	whole-body	5.9	1.8	0.9
Others	whole-body	7.0	5.0	3.0

Table 6. Summary of dose commitment in mrad from radionuclides produced in all nuclear tests carried out before 1976. (From USCEAR, Ref. 1, with permission by United Nations © 1977.)

Source of radiation	Northern temperate zone				Southern temperate zone				World population			
	Gonads	Bone marrow	Bone living cells	Lung	Gonads	Bone marrow	Bone living cells	Lung	Gonads	Bone marrow	Bone lining cells	Lung
External												
Short-lived nuclides	48	48	48	48	11	11	11	11	11	30	30	30
^{137}Cs	62	62	62	62	18	18	18	38	38	38	38	38
Internal												
^3H	2	2	2	2	0.2	0.2	0.2	0.2	2	2	2	2
$^{14}\text{C}^a$	7	32	29	9	7	32	29	9	7	32	29	9
^{54}Mn				1								1
^{55}Fe	1	0.6	1	1	0.3	0.2	0.3	0.3	0.7	0.4	0.7	0.7
^{90}Sr		84	120			24	33			52	71	
^{89}Sr		0.4								0.3		
^{106}Ru				41				10				24
^{137}Cs	27	27	27	27	8	8	8	8	17	17	17	17
^{144}Ce				65				15				38
$^{239}\text{Pu}^b$			1	1			0.3	0.3			0.9	0.9
Total ^c	150	260	290	260	45	93	100	72	94	170	190	160

NOTES: 1. The dose commitments for ^{85}Kr , ^{136}Cs and ^{140}Ba , although discussed in the text, are not shown in this table because they are negligible compared with the values included.

2. For internal irradiation, where body activities have been measured, uncertainties are probably within a factor of 2. For external irradiation and for lung-dose estimates, where the models used were not checked by direct measurement, the uncertainties are probably within a factor of 5.

^aDose accumulated up to the year 2000. The total dose commitment to gonads and lung is about 120 mrad, to bone lining cells 414 mrad and to bone marrow 455 mrad, delivered over some 8300 y.

^bThese dose commitments appear to be the same in different organs because of rounding. Only inhalation contributions are shown; for discussion on the ingestion pathway see Chapter II, Section N.

^cRounded to two significant figures.

Fig. 4. Average ^{90}Sr intake in New York City (pCi/day) during the 1960's.

would be comparable to the summed contribution from the other radionuclides, e.g., the order of 200 mrad.

5. Conclusions

The total average radiation dose from natural sources is 100 mrem/yr, with considerable variation, depending on place, time, and conditions. This constitutes an irreducible exposure background which, along with fallout radiation in some cases, can often hinder measurements of doses from other sources, particularly in view of the variation in the natural levels. An important part of formulating an environmental monitoring program to measure increases in concentration or dose due to human activities is to characterize adequately the natural radiation environment in the vicinity of interest. For the most part, instruments are available for measuring the individual contributors to exposure. These are discussed

both in the sections on radionuclide instrumentation and in the sections discussing instrumentation by radiation type. A useful general reference on environmental radiation monitoring is a recent NCRP report, No. 50 (Ref. 26).

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III.A.1. COMBINATION INSTRUMENTS

This short section is intended only to point out where the information exists in this volume for understanding the operating characteristics, as well as the components (detectors, electronics, etc.), of combination instruments. Combination instruments are simply those instruments capable of measuring a variety of radiation types. Unfortunately, due to the widely varying properties of the different ionizing radiations, a single instrument cannot simultaneously determine and/or differentiate α , β , γ , x , and neutron radiation at the same time. For a discussion of the physical properties of radiation, as well as the basic detection methods, the reader is referred to Section I.E, "Basic Radiation Detection Methods," and the references mentioned therein.

Normally, combination instruments are composed of several components. Basically they tend to consist of a single electronics unit, with a variable high voltage supply and a count rate meter or other readout device, and an assortment of interchangeable probes or detectors. Some basic electronic components are briefly discussed in Section I.E. To gain insight into the operating characteristics and possible deficiencies with specific detectors for the different types of radiation, the reader is referred to the following discussions and the references contained therein:

- Section III.B, "Alpha Particle Instrumentation"
- Section III.C, "Beta Particle Instrumentation"
- Section III.D, "X and Gamma Monitoring Instrumentation"
- Section III.F, "Neutron Monitoring Instrumentation"

The simplest combination instrument is an ionization or GM detector having a thin window and a sliding thicker window cover to distinguish between beta and photon (x and gamma) radiation. Unfortunately, other such simple compact instruments do not exist for differentiating such combinations as neutrons and photons, alpha and neutrons, or making further differentiation in a more complex radiation environment. For this reason, it is wise for the user of radiation measuring instrumentation to be aware of the types and ener-

gy spectrum of radiation that he will normally encounter.

Information regarding the type of radiation normally encountered at several sources is discussed in Division II. These sections attempt to give an insight into where radiation will appear and its type, as well as an idea of the energy spectrum one can normally expect to find. Such discussions appear in the following sections:

- Section II.A, "Particle Accelerators"
- Section II.B, "Nuclear Reactors"
- Section II.C, "Nuclear Spent Fuel Processing"
- Section II.D, "Sources of X-Radiation"
- Section II.E, "Mining and Milling of Uranium Ores"
- Section II.F, "Natural Radiation Background and Fallout"

Furthermore, relevant discussions of generally applicable basic concepts that must be borne in mind when making radiation measurements appear in Section I.F, "General Analytical Considerations."

III.A.2. INSTRUMENTATION BY TYPE OF RADIATION OR INSTRUMENT - EXPLANATION OF GREEN SHEETS AND INSTRUMENT NOTES

The green sheets and the instrument notes are useful for locating an ionizing radiation measuring instrument that may be needed, as well as for making comparisons among a possible group of appropriate instruments. We discuss first the format of the instrument notes since that will facilitate a discussion of the format of the green sheets.

Table 1 illustrates the format used for the instrument notes. The generic name of the instrument is that given by the manufacturer, such as Survey Meter, Air Monitor, Liquid Monitor, etc. The heading class indicates whether the instrument is portable or only for use in a laboratory. The principle of operation states the kind of detector used, whether gas-filled (Geiger-Muller, ionization, or proportional), scintillator, semiconductor or other; also the associated electronics that are supplied or needed are indi-

Table 1. Instrument Note Format

Generic Name of Instrument	
Model Name and Number	
Glossy Photograph	
Class	
Principle of Operation	
Sensitivity and Range	
Sampling	
Performance	
Requirements	Power:
	Size:
	Weight:
Features	
References	
Cost	
Address	

cated. Sensitivity and range gives information on the range of applicability, whether the instrument measures integrated dose and/or dose-rate, the energy range of detection for the radiation measured, and the degree to which a reading presents the true value, as well as any energy dependence it may have. Sampling indicates whether the instrument functions continuously or intermittently, as well as whether sample preparation or special sampling procedures may be required. Performance gives specifications on temperature, humidity, timing and/or background radiation response; other information such as its behavior in a high radiation field, the signal size outputs, whether analog or digital output, etc., may also be found. Requirements provides the size, weight, and power requirements of the instrument. In features, information on any available options is given, as well as any special information that does not fit into the above categories. References gives the source of information for compiling the note. The cost is the latest price available. The address gives the current mailing address for the manufacturer's main office, the phone number, and sometimes the name of a person to contact for more information. The first page of each instrument note also gives a date (in the upper right-hand corner) indicating the last alteration based on information obtained from the indicated references.

Instrument notes appear behind each discussion section in Division III,

while all the instrument notes in Division IV appear at the end of the last discussion section. Each such group of instrument notes is arranged alphabetically by manufacturer.

From the information on these instrument notes, green sheets summarizing all the instrument notes in Division III and IV were compiled and placed in front of the notes in each division. Table 2 lists the format used. The information given should be evident from the discussion on instrument notes above. The location gives the place of the relevant instrument note in the volume and is designated by giving the first and third line of the mnemonic appearing in the upper right hand corner of the note. The first line (e.g., RAD-ALP) indicates in what section the instrument note appears, while the next line identifies where in that section the note appears. This second line names the manufacturer, and - as noted above - the notes are arranged alphabetically according to the manufacturer; a numeral may identify whether it is the first, second, etc. note for that particular manufacturer.

From these two sources of information the reader should be able to find instruments for a given application. Although the attempt was made to make this compilation complete, some manufacturers are not included because of incomplete or lack of response. In some cases, the reader may wish to use this volume in conjunction with compilations by professional societies; al-

Table 2. Green Sheets Format

Manufacturer and model
Detector
Range
Power
Response time
Radiation detected
Detection limits
Location

though giving less detail on specific instruments, these sources typically include up-to-date lists of instruments and their manufacturers. For example, Analytical Chemistry publishes annually

a Lab Guide listing instruments, equipment, and supplies available commercially in the U.S. Another source is the annual "Buyer's Guide" published by Nuclear News.

Portable Instruments

Manufacturer Model	Detector	Range	Power	Response Time	Radiation Detected	Detection Limits	Location
Baird 904-121	GM tube	1-100 mR/h 500-50K cpm	5 batteries		γ , x ray		RAD-GAM, XRA Baird
Capintec 174, 192	Ionization chamber	2 mR/min - 2000 R/min depending on model	batteries or ac line de- pending on model	0.61 sec or 0.01 sec in integrate mode	β , γ , x ray		RAD-GAM, XRA Capintec 6
Eberline RO-2, RO-3	Ion chamber	0-5 to 0-5000 mR/h 4 ranges	3 batteries 9v		β , γ , x ray		RAD-ALP, BET, GAM, XRA Eberline 2
Eberline RO-4, RO-5	Ion chamber	0.1 to 2000 mR 0.1 mR/h to 200 R/h	3 batteries 9v and 2 batteries 30v		β , γ , x ray		RAD-ALP, BET, GAM, XRA Eberline 2
Eberline E-120/HP-270	GM tube	0-0.5 to 0-50 mR/h 3 ranges	2 batteries "D"		β , γ , x ray		RAD-ALP, BET, GAM, XRA Eberline 2
Eberline E-520/HP-270	GM tube	0-0.2 to 0-2000 mR/h 5 ranges	2 batteries "D"		β , γ , x-ray		RAD-ALP, BET, GAM, XRA Eberline 2
Eberline PRS-1/AC3 PRS-2/AC3 PAC 4G-3 PAC-6, PAC-7	Scintillation or gas proportional depending on model	0-500 to 0-100K depending on model	Gel-cell, "D" cell or 9v battery depending on model		α		RAD-ALP Eberline 3
Eberline RT-1A, E-130A PIC-6A, E-520/HP-270 E-530/HP-270 6112, E-120/HP-270, RO-7	GM tube or ionization chamber depending on model	0-1 mR/h to 0-20 KR/h depending on model	"D" cells, 9v, 30v, "C" or "C-Zn" cell battery depending on model		β , γ		RAD-BET, GAM Eberline
Eberline PRS-2P/NRD	BF ₃ tube	0.2 mrem/h to 10 rem/h	rechargeable Gel-Cell		neutrons		RAD-NEU Eberline 2

(continued)

Portable Instruments (continued)

Manufacturer Model	Detector	Range	Power	Response Time	Radiation Detected	Detection Limits	Location
Elscint GSM-1	GM tube	0-0.5, 0-5, 0-50, 0-infinity mR/h	4 batteries 1.5v type "C"		β , γ , x ray		RAD-ALP, BET, GAM, XRA Elscint
Elscint GSM-1	GM tube	0-0.5, 5.0, 50 mR/h	4 batteries, 1.5v	2 to 4 sec time constant	γ , x ray		RAD-GAM, XRA Elscint
Gamma Industries 150B, 155B, 250B	GM tube	0-2 mR/h to 0-1000 mR/h depending on model	2 batteries, "D"	Ca. 5 sec time constant	γ , x ray		RAD-GAM, XRA Gamma Industries
Health Physics Instruments 1075	Ionization chamber	0.001 to 200 R/h in 3 ranges	batteries, 4 "D" cells, 1 "AA" cell	3.5 sec on 2 R/h range	β , γ		RAD-BET, GAM Health Physics Instruments
Health Physics Instruments 1010	Ionization chamber	0-0.1, 1, 10, 100, 1000 mR/h	4 batteries, "D"	<1 to 10 sec depending on range	γ , x ray		RAD-GAM, XRA Health Physics Instruments
Health Physics Instruments 1072	Ionization chamber	0.1 to 199.9 mR/h 1 to 1999 mR/h	8 batteries, "D"	3 sec	γ , x ray		RAD-GAM, XRA Health Physics Instruments 2
Johnson and Associates GSM-5	28 different probes	0-0.2, 2, 20 mR/h; 500, 5K, 50K, cpm	3 batteries "D"	5 sec on lowest range, 1 sec on highest	α , β , γ , and neutrons depending on probe		RAD-ALP, BET, GAM, XRA Johnson and Associates
Johnson and Associates GSM-8	GM tube	0-10, 100, 1K mR/h	3 batteries, "D"	10 sec time constant	γ , x ray		RAD-GAM, XRA Johnson and Associates 2
Ludlum 2 & 3	Various	0-5 mR/h, x0.1, x1.0, x10; 0-2 mR/h x0.1, x1.0, x10, x100	2 batteries "D"		α , β , γ , depending on probe		RAD-ALP, BET, GAM, XRA Ludlum

(continued)

Portable Instruments (continued)

Manufacturer Model	Detector	Range	Power	Response Time	Radiation Detected	Detection Limits	Location
Ludlum 12, 16	Various	0-500 cpm x1, x10, x100, x1K	2 batteries "D"	11 sec for Model 16	α, β, γ , depending on probe		RAD-ALP, BET, GAM, XRA Ludlum 2
Ludlum 15	BF ₃ detector	0-500, 5K, 50K, 500 K cpm	2 batteries "D"	8 sec or 15 sec	α, β, γ , or neutron		RAD-ALP, BET, GAM, XRA Ludlum 3
Ludlum 5	GM tube	0-0.2 to 0-2000 mR/h in 5 linear ranges	2 batteries, "D"		γ , x ray		RAD-GAM, XRA Ludlum
Ludlum 14-C	2 GM tubes	0-0.2 to 0-2000 mR/h in 5 decades	2 batteries, "D"	3 sec or 11 sec	γ , x ray		RAD-GAM, XRA Ludlum 2
Ludlum 2 & 3	ZnS(Ag), ⁶ LiI(Eu)	0-5 to 0-100 mR/h depending on model	2 batteries, "D"	3 or 11 sec	neutrons		RAD-NEU Ludlum
Ludlum 15	BF ₃ and GM tube	0-500 cpm x1, x10, x100, x1000	2 batteries, "D"	3 or 11 sec	α, β, γ , neutrons		RAD-NEU Ludlum 4
Nuclear Associates 05-700	GM tube	0-0.5, 5, 50 mR/h 300, 3K, 30K cpm	batteries, 4 "D" cells	1 to 3 sec depending on scale	β, γ		RAD-BET, GAM Nuclear Associates
Nuclear Associates 05-571, 572	GM tube	0-10, 100, 1000 mR/h 0-500, 5K, 50K cpm depending on model	4 batteries, "AA"	0.3 - 30 sec depending on model	γ , x ray	0.005 Ci	RAD-GAM, XRA Nuclear Associates
Nuclear Associates 05-670	Ionization chamber	from 0-0.3 mR to 0-3000R, 15 linear scales	ac line	1, 3, and 10 sec time constants	γ , x ray		RAD-GAM, XRA Nuclear Associates 6
Nuclear Enterprises 0500	Ionization chamber	0-3 mR/h to 0-10 R/h depending on range selected	batteries, 2 1.5v and 2 22.5v		β, γ		RAD-BET, GAM Nuclear Enterprises

(continued)

Portable Instruments (continued)

Manufacturer Model	Detector	Range	Power	Response Time	Radiation Detected	Detection Limits	Location
Nuclear Enterprises NM-1	BF ₃ proportional counter	0.1 mrem/h to 10 rem/h, 5 decades	2 batteries, "pp9" and 2 batteries, "pp7"	60 sec	neutrons		RAD-NEU Nuclear Enterprises
Nuclear Enterprises MK-7	³ He surrounded by Cd and moderator	0.1 mrem/h to 1000 mrem/h	2 batteries, "R20"		neutrons		RAD-NEU Nuclear Enterprises 2
Nuclear Research CP2, CP5, CP10		from 0-0.25 to 0-10K mR/h depending on model	4 batteries 22.5v 1 battery 1.3v		α, β, γ		RAD-ALP, BET, GAM, XRA Nuclear Research Corp. 3
Nuclear Research ASM-30	Scintillation	10 ³ , 10 ⁴ , 10 ⁵ and 10 ⁶ cpm	2 batteries "D"		α		RAD-ALP Nuclear Research Corp. 2
Nuclear Research Corp. NP-2 and E-10	BF ₃ and Hurst chamber	0-2, 20, 200, 2K mrem/h 0-5, 50, 500 mrep/h depending on model	3 batteries, "D" or 6 batteries, "D"		neutrons		RAD-NEU Nuclear Research Corp.
Nucleus S-102	GM tube	β, γ, x ray 0-0.5, 5, 50 mR/h; α, β, γ, x ray 0-500, 5K, 50K cpm	2 batteries "D"	2 to 10 sec	α, β, γ, x ray		RAD-ALP, BET, GAM, XRA Nucleus
Technical Associates TBM-3	GM tube	0-0.15, 1.5, 15 mR/h 0-500, 5K, 50K cpm	1 battery 1v		α, β, γ		RAD-ALP, BET, GAM, XRA Technical Associates
Technical Associates CP-44, CP-5, CP-6, CP-7, ERJ-9	Ionization chamber	from 0-0.5 mR/h to 0-250 R/h depending on model	1 battery "D"		β, γ, x ray		RAD-ALP, BET, GAM, XRA Technical Associates 3

(continued)

INSTRUMENTATION
FOR ENVIRONMENTAL
MONITORING

RAD-ALP, BET, GAM, XRA
Table
Page 4

Portable Instruments (continued)

Manufacturer Model	Detector	Range	Power	Response Time	Radiation Detected	Detection Limits	Location
Technical Associates CP-TP-54A, 54B, 10K	Ionization chamber	0-0.5 R/h to 0-10 KR/h depending on model	battery, "D"		β, γ		RAD-BET, GAM Technical Associates
Technical Associates REM-PUG	BF ₃ plus moderator	0.1 to 10 rem/h, log response	3 batteries, "D"		neutrons		RAD-NEU Technical Associates
Texas Nuclear 2590 series 2650 series	Ionization chamber	from 0-0.1 mR/h to 0-1050 R/h depending on model	3 or 4 "D" cell batteries de- pending on model	time constants from 0.5 to 8 sec depending on model	α, β		RAD-ALP, BET, GAM, XRA Texas Nuclear
Texas Nuclear 9120, 9121 9122	GM tube	0-0.02 to 0-20K mR/h depending on model	batteries, 3 1.5v Ni-Cd	1 sec to 1 min depending on decade	β, γ		RAD-BET, GAM Texas Nuclear
UKAEA 3247, 3248	GM tube	0.1 - 100 R/h, 5-5K R/h depending on model	batteries, 2 U2 cells		β, γ		RAD-BET, GAM UKAEA
UKAEA 0949, 0949-2	³ He detector	1 mrem/h to 10 rem/h	2 batteries, "PP6"	2.5 - 20 sec	neutrons		RAD-NEU UKAEA 1, 2
Victoreen 400 series 700 series, and 885	Ionization chamber	from 0-0.2 mR/h to 0-100 R/h. From 0-150 to 0-80K cpm depending on model	various batteries depending on model		α, β, γ, x ray		RAD-ALP, BET, GAM, XRA Victoreen
Victoreen 488A	B lined proportional counter	0-800, 8K, 80K and 800 K cpm	2 batteries, "D"	1.5, 5, 15 sec	neutrons		RAD-NEU Victoreen
XETEX 306A 308A	GM tube	0-100 to 0-9.99K cpm depending on model	batteries "AA" or 9v depending on model	6 sec to 300 sec depending on model	α, β, γ, x ray	0.002 μ C	RAD-ALP, BET, GAM, XRA XETEX
XETEX 304A, 305A	GM tube	0.1 to 99.9 mR/h or 99.9 R/h depending on model	batteries, 4 "AA" alkaline or 4 "AA" Ni-Cd depending on model	1 sec	β, γ		RAD-BET, GAM XETEX

Stationary Instruments

Manufacturer Model	Detector	Range	Power	Response Time	Radiation Detected	Detection Limits	Location
Baird 450	GM tube	0-100, 300, 1K, 3K, 10K, 30K, 100K, 300K, 1000K cpm	ac line	0.1, 1 and 10 sec time constants	α, β, γ, x ray		RAD-ALP, BET, GAM, XRA Baird 2
Baird 912-000	gas flow, proportional		ac line		α, β	$\alpha = 1$ cpm $\beta = 25$ cpm (background)	RAD-ALP, BET, GAM, XRA Baird
Canberra 2200/2200D/2201	gas flow proportional		ac line		α, β	$\alpha = 0.035$ cpm $\beta = 1.0$ cpm (background)	RAD-ALP, BET, GAM, XRA Canberra
Eberline SAC-4	Scintillator	6 decade readout	ac line		α	0.3 cpm	RAD-ALP Eberline 2
Gamma Cat.# 831-0010-049	GM tube	0-2, 0-20, 0-200 mR/h	ac line				RAD-ALP, BET, GAM, XRA Gamma
Harshaw TASC-12	Phoswich scintillator		ac line		α, β	$\alpha = 0.05$ cpm $\beta = 0.7$ cpm (background)	RAD-ALP, BET, GAM, XRA Harshaw
Johnson and Associates RML-1A	30 different probes	500, 5K, 50K cpm	ac line	1.5 sec or 4.5 sec	α, β, γ , and neutron depending on probe		RAD-ALP, BET, GAM, XRA Johnson and Associates 2
Nucleus ML	GM tube	500 to 50K cpm	ac line	2 sec	α, β, γ		RAD-ALP, BET, GAM, XRA Nucleus 2

Dosimeters

Manufacturer Model	Detector	Range	Power	Accuracy	Radiation Detected	Leakage	Location
Dosimeter Corp. 608, 619, 686, 862, 883, 866	Ionization chamber	0-0.2 to 0-600R depending on model	charger uses one battery	$\pm 10\%$	γ , x ray	2% per day	RAD-DOS Dosimeter Corp.
Dosimeter Corp. 886, 884, 609	Ionization chamber	0-200 mR, 0-120mrem depending on model	External charger	$\pm 10\%$ or $\pm 20\%$	γ only, γ and neutrons, or neutron only	2% per day	RAD-DOS Dosimeter Corp. 2
Eberline RT-1A	GM tube	chirps from 0.1 mR/h to 5000 R/h continuous chirp at high fields	battery 9v		γ		RAD-DOS Eberline
Eberline TLD service	TLD	5-30,000 mrem		4% - 20% depending on range	γ , x ray		RAD-DOS Eberline 2
Eberline TLD reader TLR-5	TLD	1 mR to 10^5 R in two ranges	ac line				RAD-DOS Eberline 3
Harshaw TLD materials	LiF_2 CaF_2 $\text{Li}_2\text{B}_4\text{O}_7$ CaSO_4	10^{-6} R to 10^6 R depending on material			γ	2% to 50% per day depending on material	RAD-DOS Harshaw
Harshaw TLD reader 2271	TLD	10 mR to 10^5 R	ac line	$\pm 3\%$ - $\pm 10\%$	β , γ , x rays, neutrons		RAD-DOS Harshaw 2
Landauer Film badge Service	Film badge	10 mR to 10^5 R			β , γ , x rays, neutrons		RAD-DOS Landauer 2
Landauer TLD service	TLD	10 mR to 10^5 R		<25% (10 mR) <10% (>100 mR)			RAD-DOS Landauer 2
Nuclear Associates 06-002,012,050, 099,608,638,686 866,869,883	Ionization chamber	0-200 mR to 0-600R depending on model	use charger	$\pm 15\%$ for 30 KeV x rays	γ , x rays	2% per day	RAD-DOS Nuclear Associates 1,2

(continued)

Dosimeters (continued)

Manufacturer Model	Detector	Range	Power	Accuracy	Radiation Detected	Leakage	Location
Nuclear Associates 05-100, 153, 205 06-502	GM tube	up to 10 R/h depending on model	batteries		γ , x rays		RAD-DOS Nuclear Associates 3,4,5
Nucleus Inc. 06-002, 050, 099	Data same as given for Nuclear Associates instrument with same model number						RAD-DOS Nucleus Inc.
Nucleus Inc 06-884, 886	GM tube	20 mrad to 600 rads	use charger	10-20% de- pending on radiation type	γ , neutrons		RAD-DOS Nucleus Inc. 2
Radiation Detection Film badge service	Film badge	0.005 to 50R 0.01 to 1000R 0.01 to 1000 rem			β , γ , x ray		RAD-DOS Radiation Detection
Radiation Detection TLD reader	TLD	10 mR to 50 KR for LiF	ac line	$\pm 10\%$, $\pm 5\%$ above 1R			RAD-DOS Radiation Detection 2
Radiation Management Corp. TLD reader	TLD	1 mR to 1000R	ac line		γ , x ray		RAD-DOS Radiation Management Corp.
Technical Associates PDR-1b, 1c PM-1	GM tube	Different chirp rate depending on model	battery		γ , x ray		RAD-DOS Technical Associates 1,2
Teledyne Isotopes TLD service 8300 reader	TLD			$\pm 10\%$ or ± 8 mR which- ever is larger	γ , x ray		RAD-DOS Teledyne Isotopes
Victoreen 541R, L, 687D	Ionization chamber	0-40 to 0-500 mR depending on model	external charger	10% of full scale	γ , x ray	2% per day	RAD-DOS Victoreen 1,2

(continued)

Dosimeters (continued)

Manufacturer Model	Detector	Range	Power	Accuracy	Radiation Detected	Leakage	Location
Victoreen TLD	LiF, CaF ₂ :Mn	1 mR to 10 ⁵ R depending on material			γ, x ray, neutrons		RAD-DOS Victoreen 3
XETEX 403B, 404A 409A-1,2, 415A, 416A	GM tube	various	batteries	+15%	γ, x ray		RAD-DOS XETEX 1,2,3,4

Stationary Personnel Monitors

Manufacturer Model	Type of Monitor	Range	Power	Response time	Radiation Detected	Detection Limits	Location
Eberline HFM-4A HFM-5A	Hand and foot	0-999 counts (preset level)	ac line		α, β (HFM-4A) β (HFM-5A)		RAD-BET, GAM Eberline 3
Eberline PMC-4B/PMP-4C	Portal	160 to 7000 cpm alarm settings	ac line		β, γ		RAD-BET, GAM Eberline 3
Johnson and Associates UM-4	Hand and shoe	0-100K cps	ac line		β, γ		RAD-BET, GAM Johnson and Associates
Nuclear Associates 05-696	Hand, shoe, clothing	three channels with 1-10 ⁴ counts/cycle log scale	ac line	10 sec count cycle	γ, x-ray		RAD-GAM, XRA Nuclear Associates 2
Nuclear Enterprises HFM6	Both hands	0 - 5 working limits	ac line	2 - 20 sec	α, β		RAD-ALP, BET, GAM, XRA Nuclear Enterprises
Nuclear Research Corp. HFA-200A	Hand and foot	0-500, 5K, 50K, cpm	ac line		α, β, γ		RAD-ALP, BET, GAM, XRA Nuclear Research Corp. 2

(continued)

Stationary Personnel Monitors (continued)

Manufacturer Model	Type of Monitor	Range	Power	Response time	Radiation Detected	Detection Limits	Location
Nuclear Research Corp. HFM-2	Hand and foot	0-99,999 counts	ac line		β, γ		RAD-BET, GAM Nuclear Research Corp.
Technical Associates HSM 10A, 10AM, 10BS, 10G	Hand and shoes		ac line		β, γ - 10A, 10AM, α, β, γ - 10BS, 10G		RAD-ALP, BET, GAM, XRA Technical Associates 2
Technical Associates PPM-21, 21A, 25P	Portal	presettable alarm 10-990 counts in preset 1-99 sec	ac line		β, γ		RAD-BET, GAM Technical Associates 4,5

Instruments Designed for Monitoring Medical Radioisotopes

Manufacturer Model	Detector	Energy Resolution	Features	Radiation Detected	Location
Beckman 4000	NaI(Tl) scintillator	<12% at 0.663 MeV (3 inch crystal)	Energy windows 0-80 KeV, 0-2 MeV; capacity for 200 or 400 samples	γ	RAD-GAM, XRA Beckman
Beckman 7000, 8000, 9000	NaI(Tl) scintillator	<15% at 0.663 MeV	Energy ranges selectable 0.5, 1, 2 MeV; capacity for 300 samples	γ	RAD-GAM, XRA Beckman 2
Capintec CRC-5, 10, 30			Geometry independent response; CRC 30 produces hard copy for patient dose card		RAD-GAM, XRA Capintec

(continued)

Instruments Designed for Monitoring Medical Radioisotopes (continued)

Manufacturer Model	Detector	Energy Resolution	Features	Radiation Detected	Location
Packard 5130	NaI(Tl) scintillator		Adjustable pulse height analyser, four energy range scales	γ	RAD-GAM, XRA Packard
Packard Prias PGD			Can accept tubes 10 to 16 mm in dia.; 240, 408 or 504 sample capacity		RAD-GAM, XRA Packard 2
Picker 632-507			Calibration feature allows correction factor to be applied for a large number of radioisotopes		RAD-GAM, XRA Picker
Tracor Analytic 1185	NaI(Tl) scintillator		Well type NaI(Tl) scintillation crystal with nearly 4π geometry		RAD-GAM, XRA Tracor Analytic

Detectors

Manufacturer Model	Types	Sizes	Operating Voltage	Radiation Detected	Location
Amperex	GM tube	Various lengths and diameters	Various	α, β, γ	RAD-ALP, BET, GAM, XRA Amperex
Baird 988-10C	NaI(Tl) scintillator	1 3/4" dia., 2" thick well size = 2 1/32" diam. 1 35/64" deep		γ	RAD-GAM, XRA Baird 2
Canberra 1701, 1702	NaI(Tl) scintillator	1" diameter, 1 mm thick		x ray	RAD-GAM, XRA Canberra 4
Canberra 7229P/7229	Ge and Ge(Li)	Area: up to 28 cm ² ; depth: up to 25 mm		γ	RAD-GAM, XRA Canberra
Canberra 7313, 33, 83	Si(Li)	Area: 10 to 80 mm ² on model Depth: 3 mm		x ray	RAD-GAM, XRA Canberra 3
(continued)

Detectors (continued)

Manufacturer Model	Types	Sizes	Operating Voltage	Radiation Detected	Location
Canberra 802 series	NaI(Tl) scintillator	3.8 - 7.6 cm dia., 2.5 - 7.6 cm thick depending on model		γ	RAD-GAM, XRA Canberra 2
Canberra Ge detector	Ge	Area: 25-2000 mm ² depth: 5-20 mm dia.: up to 50 mm		low energy photons	RAD-GAM, XRA Canberra 5
Fisher Controls NE 014	Combination β chamber γ chamber	25.4 cm H by 25.4 cm D		β, γ	RAD-BET, GAM Fisher Controls
Harshaw	Phoswich detectors	3/4" to 5" diameter		α, β, γ	RAD-ALP, BET, GAM, XRA Harshaw 2
Harshaw DJ-Si detector	Si	Area: 100-700 mm ² depth: 100-200 μ m	100v positive bias	α	RAD-ALP Harshaw 2
Harshaw DN 105	Si diode	2" x 0.19"		neutrons	RAD-NEU Harshaw
LND	GM and proportional	Various lengths and diameters	Various	α, β, γ	RAD-ALP, BET, GAM, XRA LND
LND 200 series	³ He and BF ₃ tubes, ¹⁰ B counter	various	600 to 4800v depending on model	neutrons	RAD-NEU LND
LND 400-454 series	Proportional counter			β, γ , x ray	RAD-GAM, XRA LND 2
LND 456 series	Proportional counter	Dia.: 2.54-6.35 cm	1200 - 1900v depending on model	x ray	RAD-GAM, XRA LND
Ludlum 42-1, 2, 4	ZnS(Ag) or ⁶ LiI(Eu)	Dia.: 1", 2" or 4 x 4 mm depending on model		neutrons	RAD-NEU Ludlum 2
Ludlum 42-5	LiI(Eu) surrounded by moderator	Dia.: 2, 3, 5, 8, 10 and 12 inch	900v	neutrons	RAD-NEU Ludlum 3

(continued)

Detectors (continued)

Manufacturer Model	Types	Sizes	Operating Voltage	Radiation Detected	Location
N. Wood D-10 through D-20 series	GM counters	Dia. from 1/2" to 2" length from 3" to 34" depending on model	1000v to 2200v depending on model	γ	RAD-GAM,XRA N. Wood
N. Wood MN and G series	BF ₃	Dia.: 1/4" to 2", active length: 1/4" to 38" depending on model	1100v to 2300v depending on model	neutrons	RAD-NEU N. Wood
Nuclear Enterprises	NaI(Tl)	Range from 1/2" x 1/2" to 5" x 6"		γ	RAD-GAM,XRA Nuclear Enterprises
Ortec 7000 series	Si(Li)	Dia.: 4, 6, 10, 16 mm		γ, x ray	RAD-GAM,XRA Ortec
Ortec Series A, B,C,D,E,R	Si surface barrier	Area: 25-2000 mm ² depth: 10-5000 μm		α,β	RAD-ALP Ortec
Princeton Gamma-Tech	Si surface barrier	Area: 25, 50, 100, 150, 200, 300, 400 mm ² depth: 100 to 2000 μm		α,β	RAD-ALP Princeton Gamma-Tech
Princeton Gamma-Tech LGC, LGW, IGP	Ge(Li)	Various; includes coaxial, well and semi planar de- tectors		γ, x ray	RAD-GAM,XRA Princeton Gamma-Tech
Princeton Gamma-Tech PGT	Ge(Li)	Well detector Dia.: 1 cm, length; 4 cm		γ, x ray	RAD-GAM,XRA Princeton Gamma-Tech 2
Quartz Products	NaI(Tl)	Dia.: 32 - 292 mm height: 25 - 102 mm depending on model		γ, x ray	RAD-GAM,XRA Quartz Products
Reuter-Stokes RS-C4-1606 series	Ionization chamber		2000v dc max.	γ	RAD-GAM,XRA Reuter-Stokes 2

(continued)

INSTRUMENTATION
FOR ENVIRONMENTAL
MONITORING

RAD-ALP, BET, GAM, XRA
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Detectors (continued)

Manufacturer Model	Types	Sizes	Operating Voltage	Radiation Detected	Location
Reuter-Stokes RS-P1-P7 and RS-C1-C6 series	Proportional counters and ion chambers	Dia.: 0.68 - 8.02 cm, length: 5 - 318 cm depending on model	25 - 3900v depending on model	neutrons	RAD-NEU Reuter-Stokes 2
TGM Detectors A, K, N and TGM series	GM tube	Dia.: 3.18 to 44.5 mm active length: 7.9 to 305 mm depending on model	600v to 1400v depending on model	γ	RAD-GAM, XRA TGM Detector
Technical Associates PNS-19, 20 PUG-IN	^6Li in ZnS(Ag)	Dia.: 2", length: 8" moderator dia.: 5", length: 6"		neutrons	RAD-NEU Technical Associates 2
Tennelec TD series	NaI(Tl)	Dia.: 25 - 127 mm, thickness: 1.5 - 127 mm, well type also available		γ , x ray	RAD-GAM, XRA Tennelec 2
Texas Nuclear 9300 series	Ionization chamber	4", 6", or 12"	900 - 1000v or 1900v depending on model	neutrons	RAD-NEU Texas Nuclear 1,2
Texas Nuclear 9361	GM tube	Dia.: 2.54 cm, active length: 15.2 cm		γ	RAD-GAM, XRA Texas Nuclear
UKAEA SRD20	Si surface barrier	Area: 700 mm ² (smaller area available) depth: 60 μm		α	RAD-ALP UKAEA

Instruments Designed to Monitor Air Particulates

Manufacturer Model	Function	Features	Location
Eberline AMS-3	Beta, gamma air monitor	Particulate is trapped on filter; activity detected by GM tube. Range is logarithmic from 10 to 100K cpm	RAD-BET, GAM Eberline 2
Eberline Alpha-3	Alpha air monitor	Blower pulls air through filter. Silicon diffused junction-type detector	RAD-ALP Eberline
Nuclear Measurements AM-2A, 3A AM-2P, 3P AM-22P, 33P	Alpha or beta count air particulates trapped on filter	Monitors moving or stationary filters. Counts α or α and β depending on model.	RAD-ALP Nuclear Measurements
Nuclear Research Corp. NRC Lab Map-60	Airborne particulates monitor	Different detectors available for different airborne nuclides; flow 0-10 cfm	RAD-ALP, BET, GAM, XRA Nuclear Research Corp.
Nuclear Research Corp. Map-1000	Alpha air monitor	Automatic; sample for 30 min and count. Can program delay for Rn counting.	RAD-ALP Nuclear Research Corp.
RADCO GM-222	Beta, gamma air monitor	Particulate is trapped on filter; activity detected by GM tube - 1K, 10K, 100K cpm	RAD-BET, GAM RADCO
RADCO/SAI 444, 442A	Alpha air monitors, select energy region	A surface barrier detector is used to select an energy window (i.e., 0.1 to 1.0 MeV or 0 to 10 MeV)	RAD-ALP RADCO/SAI
Technical Associates BAM-3A, 3B, 3H	Air monitors	Blower pulls air through filter. Various ranges depending on model.	RAD-ALP, BET, GAM, XRA Technical Associates 5
Technical Associates SAAM-3	Alpha air monitor	Moving filter, delay for Rn counting	RAD-ALP Technical Associates
(continued)

Instruments Designed to Monitor Air Particulates (continued)

Manufacturer Model	Function	Features	Location
Technical Associates FM-5 ABN	Beta, gamma air monitor	Particulate is trapped on filter; activity detected by GM tube 500, 5K, 50K, 500K cpm	RAD-BET, GAM Technical Associates 3
UKAEA 36190	Alpha count filters, select energy region	Threshold and channel width can be set in the range 0.1 - 8 MeV	RAD-ALP UKAEA 2

Special Instruments

Manufacturer Model	Function	Features	Location
<u>RATE METERS</u>			
Eberline PRS-1, PRS-1P, PRS-2, PRS-2P	Rate meter or scaler	Can be used with α , β , γ or neutron probes. Battery power.	RAD-ALP, BET GAM, XRA Eberline
Elscint GRM-1	Count rate meter	Can be used with many different probes; audible alarm, includes power supply for GM tubes	RAD-ALP, BET GAM, XRA Elscint 2
Johnson and Associates LS-5	Scaler	Can be used with Geiger, scintillation or proportional counters; can use battery or line power.	RAD-ALP, BET GAM, XRA Johnson and Associates 3
Nuclear Enterprises RM2, RM5, PCM5	Rate meters	Can be used with α scintillation and β , γ GM probes	RAD-ALP, BET GAM, XRA Nuclear Enterprises 2, 3
Technical Associates SML-1, SML-2, FS-8 PUG-1, PUG-1AB, PUG-1E	Rate meters	Can be used with GM tube or scintillation detector. Battery power.	RAD-ALP, BET, GAM, XRA Technical Associates 4
<u>SPECTROMETERS</u>			
Fisher Control 0928	Monitor gamma radiation	Energy windows: 0.05-1 MeV, 1-2 MeV, 1.25-2.25 MeV; 2-3 MeV	RAD-GAM, XRA Fisher Control 2
Harshaw 272	Alpha count samples of < 1 1/4" O.D. x 1/8" thick	Diffused silicon junction detector, pulse height analyser, battery powered.	RAD-ALP Harshaw
Ortec 525-780, 782	Determine neutron intensity and energy	Uses a ^6LiF or ^3He sandwich. Determines energy by biasing detectors or either side of material in which neutrons produce particle pairs.	RAD-NEU Ortec
Ortec 576	Alpha count filters or planchets; determine energy spectrum	A surface barrier detector is used to obtain resolution of ≤ 30 KeV at 5.5 MeV	RAD-ALP Ortec 2
(continued)

Special Instruments (continued)

Manufacturer Model	Function	Features	Location
Ortec 8000 Series	Determine x-ray intensity and energy	Energy resolution: 180-500 eV at 5.9 KeV, 485-750 eV at 122 KeV depending on model	RAD-GAM, XRA Ortec 2
Peabody Scientific FNS-1	Determine neutron energy and intensity	Energy resolution achieved by monitoring energy of products of the reaction $^3\text{He} + n$ $= {}^1_0\text{H} + {}^3_1\text{H}$	RAD-NEU Peabody Scientific
Reuter-Stokes RS-P3-1605-261 RS-P3-1605-285	Determine x-ray intensity and energy	Energy resolution -10% at 14.4 KeV, 17% at 5.98 KeV	RAD-GAM, XRA Reuter-Stokes
United Scientific 101	Determine x-ray intensity and energy	Resolution varies between 144 to 290 eV at 5.9 KeV depending on detector	RAD-GAM, XRA United Scientific
AREA MONITORS			
Capintec Diarad III, Therarad I Tomorad I, 192	Sample discrete field in radia- tion therapy	Air ionization chamber detector, digital display	RAD-GAM, XRA Capintec 2,3,4
Dosimeter Corporation 1015 Series	Verify proper operation of x-ray equipment and fluoroscopes	Ionization chamber detector, digital display	RAD-GAM, XRA Dosimeter Corporation
Eberline RM-16, NRD-1	Monitor level of neutrons near reactor	Monitors thermal to 10 MeV neutrons; range: 1-10K mrem/h	RAD-NEU Eberline
Eberline SPI-2	Measure stray radiation from teletherapy machines	Flash warning light if in radiation field; alarm when exposure rate exceeds threshold.	RAD-GAM, XRA Eberline
Fisher Controls GM-700	Measure level of radiation field	Alarm level adjustments, recorder output	RAD-GAM, XRA Fisher Controls
Johnson and Associates TVX-1	Detect x-ray leakage from TV receivers	Large scanning area (20" and 5")	RAD-GAM, XRA Johnson and Associates
MDH Industries 2025AC	Monitor output of x-ray ma- chines	Uses several different chambers for different ranges	RAD-GAM, XRA MDH Industries Inc.2
(continued)			

Special Instruments (continued)

Manufacturer Model	Function	Features	Location
MDH Industries 1015C	Verify proper operation of diagnostic x-ray machines	Uses digital processing techniques to measure pulse intensity, probe rate and pulse width	RAD-GAM,XRA MDH Industries Inc.
Nuclear Associates 05-433,437	Monitor radiation level of therapy machines	Alarm from 1 mR/h to 32 mR/h depending on model	RAD-GAM,XRA Nuclear Associates 3,4
Nuclear Research Corp. TA-900	Area radiation monitor	Range: 0.01 to 10 ⁷ mR/h external alarm	RAD-GAM ,XRA Nuclear Research Corp.
Reactor Experiments 710,700 and 500 series	Monitor neutron level	Activity induced into various foils is used to determine levels of neutron exposure	RAD-NEU Reactor Experiments 1,2,3
Reuter-Stokes 1011	Monitor neutron and gamma levels	Remote electronics station, water proof ionization chamber detector	RAD-NEU Reuter-Stokes
Technical Associates FML-5A,5B	Sound alarm if radiation level exceeds preset value	Measurement of level from 0.01 mR/h to 10 R/h in 6 decades	RAD-GAM,XRA Technical Associates
Technical Associates WRAM	Monitor and record beta, gamma activity over a wide area.	Portable unit, including strip chart recorder, yields a record of distribution of activity.	RAD-BET,GAM Technical Associates 2
UKAEA 0467	Give alarm if gamma activity exceeds preset level	GM tube mounted internally or up to 10m external	RAD-GAM,XRA UKAEA
Victoreen 570	Monitor diagnostic and therapeutic x rays.	Provided with a variety of probes that cover a range of energy and exposure	RAD-BET,GAM Victoreen 2
Victoreen 692D	Monitor gamma and x-ray levels	Measures direct beam or scatter radiation from radioactive source, portable.	RAD-BET,GAM Victoreen
Victoreen 808D	Monitor gamma levels in area containing radioactive sources.	Fail safe alarms, visual and audible	RAD-BET,GAM Victoreen
XETEX 302B,303A	Monitor high radiation levels near accelerators	Range: 0.01 to 1000 R/h depending on model	RAD-GAM,XRA XETEX 2

(continued)

INSTRUMENTATION
FOR ENVIRONMENTAL
MONITORING

RAD-ALP, BET, GAM, XRA
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Special Instruments (continued)

Manufacturer Model	Function	Features	Location
XETEX 501A-1,2,3	Monitor level of ionizing radiation; give alarm	Range 0.01 to 990 mR/h depending on model; settable alarm	RAD-GAM,XRA XETEX
<u>OTHER SPECIAL INSTRUMENTS</u>			
Fisher Controls 0593	Uranium prospecting	Very rugged, light weight, audible alarm, liquid crystal display	RAD-GAM,XRA Fisher Control 3
Fisher Controls NE-148A	Gamma monitor under severe climatic conditions	Can be used underwater to a depth of 5 ft.	RAD-GAM,XRA Fisher Controls 4
Harshaw Phoswich Scintillation detector	Count alpha and beta particles in the presence of a gamma background	Consists of a thin CaF ₂ (Eu) and a thick NaI(Tl) scintillator. Only events that occur in the CaF ₂ (Eu) layer are counted	RAD-ALP Harshaw 3
Harshaw TASC-21	Field monitoring of low levels of transuranics	Designed for rough handling; optimized for 15-100 KeV photons	RAD-GAM,XRA Harshaw
Harshaw TASC-4	Real time gamma monitor for air or water effluent	Air range: 1 to 10 μ R/h Water sensitivity: 10 pCi/l	RAD-GAM,XRA Harshaw 2
National Nuclear Corp. LMS-1	Alpha count liquids	A scintillating compound is added to the liquid and concentration determined by monitoring light output	RAD-ALP National Nuclear Corp.
Nuclear Associates 05-582	Measure "transport index" of packages of radioactive material	Three foot telescoping spacing rod facilitates measurement of levels for "transport index"	RAD-GAM,XRA Nuclear Associates 5
Reuter-Stokes RSS-111	Measure gamma activity in the outside environment	Weatherproof construction; magnetic tape output	RAD-GAM,XRA Reuter-Stokes 3
Technical Associates CP-MU	Underwater monitoring of reactor spent fuel or hot cell	Measures levels of 0-10 ³ , 10 ⁴ , 10 ⁵ , 10 ⁶ R/h	RAD-GAM,XRA Technical Associates 2
Teledyne Isotopes Gamma Systems	Provide NaI(Tl) crystals and a variety of photon spectrometry systems	Wide variety of crystal sizes and shapes available	RAD-GAM,XRA Teledyne Isotopes

INSTRUMENTATION
FOR ENVIRONMENTAL
MONITORING

RAD-ALP, BET, GAM, XRA
Detectors
Amperex
May 1980

Radiation Counter Tubes
Amperex Electronic Corporation

Class	Detector
Principle of Operation	Geiger-Muller detectors used to count alpha, beta, gamma and x-ray
References	Manufacturer's Specifications
Address	Amperex Electronic Corporation 230 Duffy Avenue Hicksville, NY 11802 (516) 931-6200

Cylindrical Counters

Model	Outside Diameter (mm)	Diameter (in.)	Overall Length (mm)	Length (in.)	Operating Voltage (V)	General Application	Price (\$)
ZP1300/18529	7	0.276	27	1.063	500-600	γ -Count or current $\bar{>}$ 1000 r/h, $\beta > 0.5$ MeV	41.00
ZP1310/18509	7	0.276	38	1.496	500-650	γ -Count or current $\bar{<}$ 300 r/h, $\beta > 0.5$ MeV	36.75
ZP1320/18550	10	0.394	52	2.047	500-650	$\beta > 0.25$ MeV; γ -Count or current	39.50
ZP1200/18503	17	0.667	55	2.165	375-600	γ	53.75
ZP1210/18520	22.2	0.874	170	6.693	375-475	γ	94.00
ZP1220/18545	22.2	0.874	270	10.63	380-480	γ	105.00
85NB3	15.9	5/8	98.4	3 7/8	850-1050	γ, β	24.00
75NB3	15.9	5/8	109.5	4 5/16	850-1050	γ	21.50
90NB3	15.9	5/8	141.3	5 9/16	875-1075	β, γ	27.50
76NB3	15.9	5/8	192.5	7 9/16	875-1075	γ	18.00
912NB3	15.9	5/8	285.4	11 1/4	875-1075	β, γ	42.25
78L	15.9	5/8	279.4	11	450-700	γ	39.50
79L	15.9	5/8	362	14 1/4	450-700	γ	57.00

End Window Counters

Model	Outside Diameter (mm)	Diameter (in.)	Overall Length (mm)	Length (in.)	Operating Voltage (V)	General Application	Price (\$)
ZP1400/18504	17	0.669	55	2.17	375-600	β, γ	56.00
ZP1410/18505	25.9	1.02	57	2.24	450-700	α, β, γ	124.00
ZP1430/18526	34	1.339	57	2.24	450-750	α, β, γ	147.50
150N	25.4	1	152.4	6	875-1075	β, γ	68.00
200NB	34.9	1 3/8	110.3	4 11/32	875-1075	α, β, γ	75.00
200HB	34.9	1 3/8	110.3	4 11/32	1200-1450	α, β, γ	78.50
120NB	58.7	2 5/16	146.1	5 3/4	875-1075	β, γ	79.00

Automatic Planchet Counting System

Baird Model 912-000

Class	Laboratory
Principle of Operation	Gas flow proportional detector, automatic planchet sample changer - - spectrometer, and printer
Sensitivity	Detector efficiency: ALPHA - 35-40% BETA - 25% Typical Background: ALPHA - 1 cpm BETA - 25 cpm
Sampling	Batch-up to 100 samples automatically
Performance	Detector: 2" dia. with thin window (0.9 mg/cm ²) or windowless operation plateau; 200-300V slope; 2% per 100V Planchet Changer: sampling capacity, 100 samples (planchet sizes, 1", 1 1/4", and 2", mixable) total sample changing time, ~12 seconds Spectrometer: resolving time, 1 μ sec pulse pairs maximum continuous count rate, 1 megahertz count capacity; 999,999 preset counts; all decades presettable 0 to 9 Time Ranges: 0.1 to 999.9 seconds or minutes Time Capacity: 4 decades presettable 0 to 9 Current Input Mode: maximum gain, 10V/ μ a; input sensitivity, 1 μ a (FS); input impedance, 50 to 100 Ω . Voltage Input Mode; maximum gain, 1000, input sensitivity, 10 mV (FS); input impedance, 10,000 Ω ; voltage range 0 to 10V High Voltage Supply: outputs: 200 - 1600 V DC at 5 ma. maximum; 1600-3000V DC at 5 ma. maximum; polarity, positive
Requirements	Power: 110 VAC, 60 Hz Size: Changer Model 912-600, cm(17"Wx19"Hx21"D) Spectrometer Model 987-514, cm(17 1/2"Wx7"Hx15"D) Printer Model 987-516, cm(8 1/2"Wx7"Hx15"D) Weight: Changer; kg (137 lbs) Spectrometer; kg (45 lbs) Printer; kg (36 lbs)
Features	Lead shielding - can be used with G.M., Scintillation or gas flow proportional detectors Counting gas required, but not included
References	Manufacturer's Specifications
Cost	\$13,650 with matrix printer 14,350 with teletype ASR 33
Address	Baird Corporation 125 Middelsex Turnpike Bedford, MA 01730 (617) 276-6204

Laboratory Monitor

Baird Model 450

Class	Stationary
Principle of Operation	Uses GM detector, gas flow counters, or scintillation detectors, amplifier, high voltage supply, meter readout, red alarm light, speaker
Sensitivity and Range	Range: 0-100, 300, 1,000, 3,000; 10K; 30K; 100K, 300K, 1M cpm Input sensitivity: minus 0.5 - 10V, adjustable
Sampling	Continuous
Performance	High voltage supply: 0-1500V, regulated 9.1% per 1% change in line voltage Recorder output: 100 mV Accuracy: better than 3% full scale, all ranges 3% at recorder outputs Time Constants: .1, 1 second; 10 seconds
Requirements	Power: 110V, 60 cycle 10 watts (220/50 optional) Size: 13.1cm H x 28.2cm W x 25.4cm D Weight: 5.9 kg (13 lbs.)
Features	An alarm trip set for 10% to 100% full scale, all ranges; a red light flashes. Battery standby in case of power failure or portable use

INSTRUMENTATION
FOR ENVIRONMENTAL
MONITORING

RAD-ALP, BET, GAM, XRA
Stationary
Baird 2
Page 2

References	Manufacturer's Specifications
Cost	\$1200.00
Address	Baird Corporation 125 Middlesex Turnpike Bedford, MA 01730 (617) 276-6000

Ultra Low Level α/β Counting System

Canberra Model 2200/2200D/2201

Class	Laboratory
Principle of Operation	Two gas flow proportional detectors operating in the anticoincidence mode for gross counting of ultra low level alpha and beta environmental samples. Alphas and Betas are simultaneously counted and separately read out in net counts. Can use automatic sample changer.
Sensitivity and Range	Sensitivity Alpha and beta Background count rate (utilizing a 5.7 cm dia. proportional detector) Alpha ≤ 0.035 cpm Beta ≤ 1.0 cpm (for ^{14}C) Energy Range 10 keV to 10 MeV
Sampling	Up to one hundred 50 mm dia. planchets can be counted without operator intervention.
Performance	Efficiency* Alpha $\geq 30\%$ for ^{210}Po Beta $\geq 30\%$ for ^{14}C , 40% for ^{90}Sr Window density - 80 gm/cm ²
Requirements	Power: 110 - 220 Vac, 50 or 60 Hz, 100 watts Size. 41" H x 42" N x 24" D (104.14 x 106.7 x 60.96 cm) Weight 1300 lbs. (585 Kg) Model 2200 Flow gas 90% argon, 10% methane

*Based on minimum sample to window distance with a standard disk source.
Simultaneous operation efficiency is dependent on the operating point selection.

Features	Simultaneous and automatic counting of low level alpha and beta particles. The system utilizes standard NIM components and a programmable controller to perform the desired operator functions. The 100 sample capacity automatic sample changer mechanism utilizes stepping motors and lead screws for simple and reliable operation. Output is compatible with all standard 20 mill current loop or Rs232 devices. Useful for measuring isotopes such as ^{14}C , ^{89}Sr , ^{90}Sr , ^{226}Ra , ^{228}Ra , ^{210}Po , ^{230}Th , etc.
Reference	Series 2200 product literature
Cost	\$27,000
Address	Canberra Industries, Inc., 70 Gracey Avenue Meriden, CT 06450 (203) 238-2351

Rascal
Ratemeter Scaler
Models PRS-1, PRS-1P, PRS-2 AND PRS-2P

Class	Portable. May also be used as laboratory scaler.	
Principle of Operation	Operates with most Eberline detector probes to measure alpha, beta, gamma and neutron radiation. Readout units internally selectable.	
Sensitivity and Range	Functions as ratemeter or scaler. In ratemeter mode, integrates to pre-selected total counts and displays rate. Digital readout with six decades. Decimal placement and calibration factor internally adjustable. For use with GM or scintillation detectors, PRS-1P and PRS-2P also may be used with proportional detectors.	
Sampling	Continuous	
Performance	High voltage to probe 500 to 1500 V. for PRS-1 and PRS-2 800 to 2400 V. for PRS-1P and PRS-2P Temperature range 0° to 140°F (-18 to 60°C) No humidity dependence	
Requirements	Power	Rechargeable Gel Cell, 75 hrs. continuous operation/charge
	Size	10 cm W x 24 cm L x 20 cm high (approx. 4 x 9 x 8 inches)
	Weight	2.3 kg (5 pounds)

INSTRUMENTATION
FOR ENVIRONMENTAL
MONITORING

RAD-ALP, BET, GAM, XRA
Portable
ratemeter/scaler
Eberline
Page 2

Features "Smart" electronics integrates to pre-set level of precision desired then displays rate. Calibration factor for each probe and decimal point placement can be quickly set during probe change. Displays digital reading and applicable units.

References See Eberline General Catalog

Cost \$1450 with Gel Cell

Address Eberline Instrument Corporation
P.O. Box 2108
Santa Fe, N.M. 87501
Tele: (505) 471-3232

INSTRUMENTATION
FOR ENVIRONMENTAL
MONITORING

RAD-ALP, BET, GAM, XRA
Survey
Eberline 2
May 1980

Alpha, Beta, Gamma Survey Meters

Eberline Portable Models

E120/HP-270

RO-4

RO-5

RO-3

RO-2

Address

Eberline Instrument Corporation
P. O. Box 2108
Santa Fe, N.M. 87502
Tele: (505) 471-3232

Model	RO-2	RO-3	RO-4	RO-5	E-120/HP-270	E520/HP-270
Radiation Measured	Beta, Gamma X-ray	Beta, Gamma X-ray	Beta, Gamma X-ray	Beta, Gamma X-ray	Beta, Gamma X-ray	Beta, Gamma X-ray
Detector	Juno type Ion Chamber 208 cc volume	Cutie-pie type Ion Chamber 440 cc volume	Juno type Ion Chamber 208 cc volume	Cutie-pie type Ion Chamber 208 cc volume	GM, Energy Compensated	GM, Energy Compensated
Range	0-5 mR/h to 0-5000 mR/h, 4 ranges	0-5 mR/h to 0-5000 mR/h, 4 ranges	0.1 mR/h to 200 R/h 0.1 mR to 2000 mR (integrate range)	0.1 mR/h to 200 R/h 0.1 mR to 2000 mR (integrate range)	0-0.5 mR/h to 0-50 mR/h 3 ranges	0-0.2 mR/h to 0-2000 mR/h 5 ranges
Scale	Linear	Linear	Digital	Digital	Linear	Linear
Meter Accuracy	±5% F/S	±5% F/S	NA	NA	±5% F/S	±15% F/S
Temperature	-40°F to 140°F	-40° to 140°F	32°F to 122°F	32°F to 122°F	-40°F to 140°F	-40° to 140°F
Power	3 NEDA 1604 (9v) batteries	3 NEDA (9v) batteries	3 NEDA 1604 (9v) and 2 NEDA 210 (30v) batteries	3 NEDA 1604 (9v) and 2 NEDA 210 (30v) batteries	Two D size batteries	Two D size batteries
Weight	1.72 kg (3.8 lb)	1.81 kg (4 lb)	1.59 kg (3.5 lb)	1.36 kg (3 lb)	1.36 kg (3 lb)	2 kg (4.4 lb)
Size (WxLxH) Including Handle	10x21x20 cm 4x8x8 inches	10x29x22 cm 4x11x9 inches	10x21x19 cm 4x8x7 inches	11x23x24 cm 4x9x9 inches	9x17x16 cm 4x7x6 inches	10x20x17 cm 4x8x7 inches
Features	Window 7 mg/cm ² mylar, wall 200 mg/cm ² phenolic	Window 3.5 mg/cm ² mylar, wall 200 mg/cm ² phenolic	Digital, measure integrated exposure and exposure rate	Digital, measures integrated exposure and exposure rate	Reads out in mR/h and CPM	Internal GM tube for highest range
Energy responses ±20% or better	12 keV to 6 MeV	12 keV to 6 MeV	12 keV to 6 MeV	12 keV to 6 MeV	40 keV to 1.3 MeV	40 keV to 1.3 MeV
Cost	\$880.00	\$900.00	\$900.00	\$930.00	\$310.00	\$480.00
Sampling	Continuous					
Features	Flat response (±20% or better) vs photon energy for measuring exposure or exposure rate. All also measure beta. RO-instruments can be made sensitive to alpha.

Survey Meter
Elscint Model GSM-1

Class	Portable, handheld
Principle of Operation	G-M detector, amplifier, meter readout, speaker
Sensitivity	Ranges: 0 to 0.5 mR/hr, 0 to 5 mR/hr, 0 to 50 mR/hr, 0 to infinity (logarithmic) Energy Dependence: $\pm 20\%$ from 60 keV to 1.3 MeV
Sampling	Continuous
Performance	Accuracy: $\pm 3\%$ FS Temperature: -35°C to 60°C (-31°F to 140°F)
Requirements	Power: Four 1.5V batteries "C" type, 100 hours continuous service Size: 16.5cm Hx10.2cm Wx19.1cm D (6 1/2"Hx4"Wx7 1/2"D) Weight: 1.1 kg (2.4 lb)
Features	1) In very high radiation fields meter indicates full scale not zero 2) Geiger tube may be operated with a cable at a distance up to 50 meters from the reading instrument.
References	Manufacturer's Specifications
Cost	\$500.00
Address	Elscint, Inc. P.O. Box 832 Hackensack, N.J. 07602 (201) 487-5885 Attn: J. W. Walker

Radiation Monitor
Elscint Model GRM-1

Model GRM - 1

Class	Laboratory
Principle of Operation	GM or scintillator detectors, amplifier, meter readout, speaker, audible alarm
Sensitivity and Range	Energy sensitivity: depends on probe Range: 0 to 300; 1000; 3000; 10,000; 30,000; 100,000; 300,000 cpm Energy dependence: $\pm 20\%$ from 60 keV to 1.3 MeV
Sampling	Continuous
Performance	Accuracy: $\pm 5\%$ FS Temperature: 0°C to 50°C 32°F to 122°F Temperature stability: $\pm 0.1\%/^{\circ}\text{C}$ High voltage range: $+200\text{V}$ to $+1200\text{V}$ Time Constant: short 1 to 2 seconds long 10 to 15 seconds
Requirements	Power: 115V or 230V, 50 or 60 Hz; 10W Size: 11.4cm Hx19.05cm Wx22.86cm D (4 1/2"Hx7 1/2"Wx9"D) Weight: 3 kg (6.6 lbs)
Features	Recorder output: 100 mV FSD
References	Manufacturer's Specifications
Cost	\$585.00 (GRM-1)
Address	Elscint Inc. P.O. Box 832 138-160 Johnson Avenue Hackensack, NJ 07602 (201) 487-5885 Attn: J. W. Walker

INSTRUMENTATION
FOR ENVIRONMENTAL
MONITORING

RAD-ALP, BET, GAM, XRA
Stationary
Gamma Industries
May 1980

Radiation Monitor
Multiple Alarm Radiation System
Catalog No. 831-0010-049

Class	Stationary
Principle of Operation	G-M detector, amplifier, meter readout, audible alarm
Sensitivity and Range	Range: 0-2, 0-20, 0-200 mR/hr
Sampling	Continuous
Performance	
Requirements	Power: 115 VAC Size: 35.6cm H x33.0cm W x20.3cm D (14"H x13"W x8"D) Weight: 6.1 kg (13 1/2 lbs.)
Features	7" high flashing, rotating beacon warns of radiation in excess of the level selected
References	Manufacturer's Specifications
Cost	\$675.00 with visual alarm 755.00 with visual and audible alarm
Address	GAMMA Industries, A Division of Nuclear Systems, Inc. P.O. Box 2543 Baton Rouge, LA 70821

Automatic Alpha/Beta Counting System

Harshaw Model TASC-12

Class Designed for laboratory use.

Principle of Operation Utilizes a CaF_2 (Eu) - NaI(Tl) Phoswich scintillator and pulse shaping electronics to simultaneously and separately measure alpha, beta and gamma radiation. Readout is LCD and printer.

Sensitivity Range and Performance The following specifications are for separate and simultaneous alpha and beta counting.

<u>Radiation</u>	<u>Detector Diameter</u>	
	<u>1"</u>	<u>2"</u>
Alpha background	> 0.05 cpm	> 0.1 cpm
Beta background	> 0.7 cpm	> 2 cpm

Detector Efficiency

<u>Alpha</u>	<u>Beta</u>
^{241}Am < 35%	$^{90}\text{Sr}/^{90}\text{Y}$ < 40%
^{210}Po < 35%	^{137}Cs < 40%
	^{14}C < 40%

Sampling Automatically measures up to 100 planchets with samples.

Requirements Power: 110 V, 60Hz or 220V, 50Hz; 50 watts. Samples changer
Size: 24" x 24" x 24" - sample changer
Weight: 1500 lbs.
Electronics size: 19.75" x 19.75" x 15"H - Electronics
Weight: 93 lbs.

Features	TASC-12 is housed in two cabinets, one is for electronics and the other for the sample changer and lead shielding. Reproducible background - Pushbutton controls - High counting efficiency - Detector uniformity - Large Phoswich scintillator available.	
Reference	Harshaw Technical Literature	
Cost	TASC-12	\$22,500.00 each
Address	Harshaw Chemical Company 6901 Cochran Road Solon, OH 44139 (216) 248-7400

Nuclear Radiation Detectors

Harshaw Phoswich Detectors

Class	Nuclear radiation detector (alpha, alpha/beta, alpha/gamma)
Principle of Operation	A Phoswich detector is a scintillator utilizing a thin primary crystal optically coupled to a thicker crystal of a different scintillation material. For alpha counting applications, $\text{CaF}_2(\text{Eu})$ is employed as the primary scintillator, with a thicker $\text{NaI}(\text{Tl})$ crystal operating as an anticoincidence shield. Signals originating in the crystals are electronically separated by the difference in scintillation decay times of each material. The calcium fluoride crystal is very efficient for alpha particles, yet transmits higher energy gamma ray events to the $\text{NaI}(\text{Tl})$ crystal for electronic rejection.
Resolution	Typically 1 MeV for ^{241}Am
Diameters Available	1.9cm to 12.7cm (3/4" to 5")
Features	Because calcium fluoride is relatively inert and nonhygroscopic, no entrance window is required between the crystal and sample in many applications. Passive shielding requirements are greatly reduced by the anticoincidence operation of the $\text{NaI}(\text{Tl})$ component.
Accessories Available	Low background tube bases, low background shields for table-top use, complete selection of nuclear instrument modules and pulse shape analysis electronics.
References	Manufacturer's Specifications
Cost	Refer to factory
Address	The Harshaw Chemical Company Crystal and Electronic Products Dept. 6801 Cochran Road, Solon, OH 44139 (216) 248-7400

Radiation Survey Meter
Johnson and Associates Model-GSM-5

Class	Portable, Handheld
Principle of Operation	Alphas, betas, gammas, and neutrons are detected by 28 different probes, integrating circuit drives meter.
Sensitivity and Range	Full Scale Ranges: 0.2, 2, and 20 mR/hr for beta-gamma 500, 5000, and 50,000 cpm, linear $\pm 4\%$ Minimum Measurable Energy: Dependent upon the probe used. Approximately 15 keV with GP-200 thin window GM tube Maximum Measurable Flux: Dependent upon the probe used Approximately 1 R/hr with GP-90 X-50
Sampling	Continuous
Performance	Meter Accuracy: $\pm 2\%$ of full scale Response Time: Based on scale range being used. 5 seconds on lowest range 1 second on high range. Input Accessories: Calibration pulser, trans amp, register and timer Output Accessories: Recorder, speaker, alarm
Requirements	Power: Three "D" cells, 200 hours Size: 10cm Wx18cm Lx17cm H (4"Wx7"Lx6.4"H) Weight: 2kg (4 lb)
Features	Battery check circuit, suitcase package, wide selection of detectors.
References	Manufacturer's Specifications
Cost	\$225.00
Address	William B. Johnson and Associates, Inc. Research Park, Boonton Avenue Montville, NJ 07045 (201) 334-9222

SCINTILLATION PROBES FOR
JOHNSON MODEL GSM-5 SURVEY METER

MODEL NUMBER	ASP-2A	ASP-3	GSP-2A
RADIATION MEASURED	Alpha	Alpha	Gamma
DETECTOR	ZnS	ZnS	2"x2" NaI(Tl)
EFFICIENCY	25 - 40%	45%	25%
RANGE	1 to 50,000 counts/min	1 to 50 counts/min	50,000 counts/min
ACCURACY	± 5%	± 5%	± 5%
SENSITIVITY	100 pCi	150 pCi	100 pCi
TEMPERATURE	-40°F to +150°F	-40°F to +150°F	-40°F to +150°F
POWER	Four 6 V Penlight Batteries	0	Four 6 V Penlight Batteries
HOUSING	Aluminum	Aluminum	Aluminum
OVERALL SIZE	22.9 cm long 6.0cm o.d. (9" Long 2 3/8" o.d.)	20.3cm long 6.0cm o.d. (8" Long 2 3/8" o.d.)	30.5cm long 6.0cm o.d. (12" Long 2 3/8" o.d.)
WEIGHT	0.68 kg (1.5 lb)	0.91 kg (2 lb)	0.91 kg (2 lb)
PRICE	\$249.00	\$200.00	\$499.00

NOTE: GSM-5 sells for \$225.00. Add probe cost for complete instrument.

GEIGER MULLER PROBES FOR
JOHNSON MODEL GSM-5 SURVEY METER

MODEL NUMBER	GP-90	GP-200	PPA-2	GP 90 X 50	GP 200 X 50
RADIATION MEASURED	Beta Gamma	Alpha Beta Gamma	Alpha Beta Gamma	Beta Gamma	Alpha Beta Gamma
DETECTOR	GM Tube	GM Tube	GM Tube	GM Tube	GM Tube
WALL THICKNESS	30 mg/cm ²	1.4 mg/cm ²	1.5 mg/cm ²	30 mg/cm ²	1.5 mg/cm ²
RANGE FULL SCALE	0.2, 2, 20 mR/hr	0.2, 2, 20 mR/hr	0.2, 2, 20 mR/hr	10, 100, 1000 mR/hr	10, 100, 1000 mR/hr
ACCURACY	±5% F/S	±5% F/S	±5% F/S	±5% F/S	±5% F/S
TEMPERATURE	-55°F to +140°F	-55°F to +140°F	-30°F to +140°F	-55°F to +140°F	-55°F to +140°F
HOUSING	Stainless Steel	Aluminum Mica	Lead Tungsten Aluminum	Aluminum	Aluminum Mica
	0.20 kg (0.44 lb)	0.17 kg (0.38 lb)	0.57 kg (1.15 lb) 1.93 kg (4.25 lb) with shield	0.059 kg (0.13 lb)	0.059 kg (0.13 lb)
OVERALL SIZE	20.3cm long 2.54cm o.d. (8" Long 1" o.d.)	15.2cm long 3.81cm o.d. (6" Long 1 1/2" o.d.)	16.5 cm long x8.9cm W x9.8cm H (6 1/2"x3 1/2"W x3 7/8" H)	8.9cm long 0.95cm o.d. (3 1/2" Long 3/8" o.d.)	8.9cm long 1.91cm o.d. (3 1/2" Long 3/4" o.d.)
PRICE	\$60.00	\$75.00	\$175.00 Alum. \$275.00	\$100.00	\$100.00

NOTE: GSM-5 sells for \$225.00. Add probe cost for complete instrument.

SCINTILLATION PROBES FOR
JOHNSON MODEL GSM-5 SURVEY METER

MODEL NUMBER	GSP-2	DGSP	FNSP-2A	SNSP-2A
RADIATION MEASURED	Gamma	Low Energy Gamma	Fast Neutron	Slow Neutron
DETECTOR	NaI(Tl)	NaI(Tl) Cleaved Crystal	Hornyak Button	LiI(Eu) Activated
RANGE	0-50,000 counts/min	0-50,000 counts/min	0.2 MeV 20 MeV	Thermal Epithermal
ACCURACY	±5%	±10%	+15% -30%	+15% -15%
TEMPERATURE	-40°F to +150°F	-40°F to +150°F	-40°F to +150°F	-40°F to +150°F
POWER	4 penlite batteries	Four 1.5 V Penlight Batteries	Four 1.5 V Penlight Batteries	Four AA Penlight Batteries
HOUSING	Aluminum	Aluminum	Aluminum	Aluminum
	0.68 kg (1.5 lb)	0.91 kg (2 lb)	0.40 kg (0.875 lb)	0.45 kg (1 lb)
OVERALL SIZE	25.4cm long 0.95cm o.d. 10" Long 3/8" o.d.	25.4cm long 6.35cm o.d. 10" Long 2.5" o.d.	27.9cm long 6.1cm o.d. 11" Long 2.4" o.d.	24.13cm long 5.72cm o.d. 9.5" Long 2.25" o.d.
PRICE	\$499.00	\$450.00	\$350.00	\$450.00

NOTE: GSM-5 sells for \$225.00. Add probe cost for complete instrument.

Radiation Monitor
Johnson Model RML-1A Count Rate Meter

Class	Laboratory
Principle of Operation	Accepts about 30 probes, amplifier, integrator, speaker, recorder, and meter readout
Sensitivity and Range	Energy range depends on probe, see previous note Full scale ranges 500, 5,000, 50,000 cpm Minimum input sensitivity: 10 mV Maximum rate: Unavailable
Sampling	Continuous
Performance	Meter accuracy: $\pm 2\%$ full scale Response time: 1.5 sec or 4.5 sec
Requirements	Power. Size: 28cm x 18cm x 18cm (11"x7"x7") Weight: 3.4 kg (7.5 lbs)
Features	Volume and discriminator control
Cost	\$375.00
Address	William B. Johnson and Associates Inc. Research Park, Boonton Avenue Montville, N.J. 07045 (201) 334-9222

Portable Scaler

Johnson LS-5

Class	Laboratory and field
Principle of Operation	Uses Geiger, scintillation, or proportional counters, high voltage supply, amplifier, L.E.D. readout
Sensitivity and Range	6 L.E.D. ranges: 0- 999,999 counts
Sampling	Pre-set time elapsed count mode
Performance	Variable high voltage: 0-2500 V
Requirements	Power: 115 Volts A/C for line operation, 4 NiCad batteries (rechargeable) for portable operation Size: 30.5cmx27.9cmx12.7cm (12"x11"x5") Weight: 4.1 kg (9 lbs)
Features	Regulated and adjustable high voltage
References	Manufacturer's Specifications
Cost	LS-5 \$975.00
Address	Wm. B. Johnson and Associates, Inc. Research Park, Boonton Avenue Montville, NJ 07045 (201) 334-9222

Alpha, Beta, Gamma Counters

LND Models

End Window Gasfilled Counting Tubes

Model	Maximum Diameter		Overall	Length	Operating voltage	Application	Price
	cm.	in.	cm.	in.			
704	0.87	.344	3.81	1.5	600-800	α β γ	\$135.00
705	0.87	.344	3.81	1.5	600-800	α β γ	588.00
710	0.87	.344	6.99	2.75	600-800	α β γ	83.00
7616	0.91	.359	6.43	2.531	750-950	α β γ	81.00
711	1.27	.5	7.62	3.0	600-800	α β γ	100.00
712	1.52	.600	5.40	2.125	375-600	α β γ	35.00
715	0.87	.344	3.81	1.5	700-850	β γ	72.00
718	5.87	2.312	14.61	5.75	650-850	α β γ	136.00
722/5979	2.54	1	15.24	6	700-850	β γ	53.00
722A	2.54	1	15.24	6	900-110	X-ray	136.00
722K	2.54	1	15.24	6	1175	X-ray	153.00
8204M	2.54	1	15.24	6	660-800	α β γ	145.00
723	3.49	1.375	11.43	4.5	900-1100	α β γ	53.00
7231	3.33	1.312	3.49	1.375	600-800	α β γ	78.00
7232	3.49	1.375	12.38	4.875	900-1100	α β γ	88.00
7233	3.49	1.375	11.43	4.5	600-800	α β γ	88.00
724/7618	2.54	1.0	4.32	1.7	900-1100	β γ	72.00
7242	2.54	1	2.54	1	600-900	β γ	80.00
727/7840	3.49	1.375	11.43	4.5	1300-1600	α β γ	55.00

Pancake Counting Tubes

Model	Maximum Diameter		Overall	Length	Operating Voltage	Application	Price
	cm.	in.	cm.	in.			
730	5.40	2.125	8.09	3.187	900-1100	α β γ	\$180.00
731	5.40	2.125	8.09	3.187	900-1100	α β γ	110.00
7311/8767	5.40	2.125	7.70	3.032	850-1100	α β γ	107.00
732	5.40	2.125	8.09	3.187	900-1100	α β γ	422.00
733	3.76	1.48	6.67	2.625	700-900	α β γ	180.00
734	1.91	.750	4.76	1.875	700-900	α β γ	180.00
7610	0.87	.344	1.91	.75	650-800	α β γ	215.00

Cylindrical Counting Tubes

Model	Maximum Diameter		Overall	Length	Operating Voltage	Application	Price
	cm.	in.	cm.	in.			
713	0.80	.314	5.08	2.0	550-700	β γ	\$ 37.00
714	0.51	.200	3.18	1.25	550-700	γ	37.00
716	0.51	.200	1.91	.750	550-700	γ	35.00
717	0.32	.125	2.54	1.0	650-800	β γ	430.00
719	1.59	.625	27.94	11	900-1100	β γ	64.00
720	1.59	.625	9.84	3.875	900-1100	β γ	36.00
7201	1.59	.625	9.84	3.875	900-1100	β γ	61.00
721	1.59	.625	14.61	5.75	900-1100	β γ	37.00
725	1.91	.750	11.11	4.375	900-1100	β γ	43.00
726/5980	0.87	.344	5.08	2	660-800	γ	35.00
740	0.74	.290	19.69	7.75	675-800	γ	77.00

Medical Counting Tubes

Model	Maximum Diameter		Overall	Length	Operating Voltage	Application	Price
	cm.	in.	cm.	in.			
760	0.87	.344	7.62	3.0	650-800	α β γ	\$230.00
761	0.87	.344	12.78	7	650-800	α β γ	490.00
762	0.87	.344	21.84	8.6	650-800	γ	490.00

Industrial and Cosmic Ray Counting Tubes

Model	Maximum Diameter		Overall	Length	Operating Voltage	Application	Price
	cm.	in.	cm.	in.			
780	2.22	.875	38.1	15	1100-1300	β γ	\$278.00
782	3.49	1.375	84.14	33.125	900-1100	γ	417.00

Gas Sampling Proportional Counters

Model	Maximum Diameter		Overall	Length	Operating Voltage	Application	Price
	cm.	in.	cm.	in.			
495	1.91	0.75	22.86	9.0		low level β	\$905.00
4951	2.22	0.875	25.40	10.0		low level β	905.00
4952	2.54	1.0	35.56	14.0		low level β	905.00
4953	8.89	3.5	59.69	23.5		low level β	1740.00

References Manufacturer's Specifications

Address LND
3230 Lawson Blvd.
Oceanside, NY 11572
(516) 678-6141
Telex 14-4563

Survey Meter
Ludlum Models 2 and 3

Class	Portable, handheld
Principle of Operation	Count rate meter for a variety of G-M and scintillator probes
Sensitivity and Range	Sensitivity: See table, page 3 and 4. Range: Model 2: 0-5 mr/hr, x0.1, x1, x10, linear Model 3: 0-2 mR/hr, x0.1, x1, x10, x100, linear 0-5,000 cpm, optional meter read-out
Sampling	Continuous
Performance	Accuracy: $\pm 5\%$ of full scale Calibration Stability: Less than 5% variance to battery end point
Requirements	Power: Two "D" cells; 300 hr Size: 9 cm x 9 cm x 18 cm (3.4" H x 3.5" W x 7.0" L) exclusive of handle Weight: 1.6 kg (3.5 pounds) less detectors
Features	Pushbutton reset, adjustable high voltage output, speaker output
References	Manufacturer's specifications

Cost Model 2 \$240.00
 Model 3 270.00

Address Ludlum Measurements Inc.
 501 Oak Street
 Sweetwater, TX 79556
 (915) 235-5494

LUDLUM PROBES

Model 43-1
Large Area Alpha Scintillator

Model 43-2
Small Area Alpha Scintillator

Model 43-5
Alpha Scintillator

Model 44-2
High Energy Gamma Scintillator

Model 44-4
End Window Geiger-Mueller Probe

Model 44-6
Thin Wall Geiger-Mueller Probe

Model 44-7
End Window GM

Model 44-9
Pancake GM

Model	Particle	Type	Tube	Window	Counting Area	Efficiency	Size				Weight		Cost
							length	diam.			(kg)	(lb)	
							(cm)	(in.)	(cm)	(in.)			
43-1	Alpha	ZnS(Ag) Scint.	RCA 6199	2 mg/cm ² Al Mylar 1 mg/cm ²	75 cm ² 4" diam.	30% of 2 π 50% of 2 π for source 1/16" window	24.13	9.5	11.43	4.5	0.9	2	\$325.00
43-2	"	"	"	1 mg/cm ² Al Mylar	11.6 cm ² 1.5" diam	50% of 2 π	16.51	6.5	5.08	2	0.4	0.9	175.00
43-5	"	ZnS(Ag) Scint.	"	1 mg/cm ² Al Mylar	50 cm ²		19.81	7.8	5.84	2.3	0.9	2	200.00
44-2	Gamma	NaI(Tl) 1" x 1"				31,000 cpm @ 0.1 mR/hr 226Ra	19.05	7.5	5.08	2	0.4	0.9	225.00
44-3	"	NaI(Tl) 1 mm x 1"		7 mg/cm ² Al 25 mg/cm ² removable			16.51	6.5	5.08	2	0.4	0.9	225.00
44-4	Beta Gamma	End window G-M	Amperex 150N	1.4 to 2 mg/cm ² mica, 25 mg/cm ² Mylar removable	25/32" diam	4,000 cpm when exposed to 226Ra flux of 1 mR/hr	19.05	7.5	3.05	1.2	0.45	1	125.00
44-6	"	Thin wall G-M	LND 725	30 mg/cm ² stainless steel, 1000 mg/cm ² removable		1700 cpm = 1 mR/hr flux of 226Ra	16.51	6.5	3.05	1.2	0.34	0.75	75.00

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RAD-ALP, BET, GAM, XRA
Survey
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Model	Particle	Type	Tube	Window	Counting Area	Efficiency	Size		Weight		Cost		
							length (cm)	(in.)	diam. (cm)	(in.)		(kg)	(lb)
44-7	"	End window G-M	LND 723	1.5 to 2 mg/cm ² mica			13.46	5.3	3.56	1.4	0.23	0.5	75.00
44-9	"	Pancake G-M	LND 725		1.75" diam		27.94	11	7.11	2.8	0.36	0.8	155.00

Survey Meter

Ludlum Models 12, 16

Model 12

Model 16

Class	Portable
Principle of Operation	Power supply for scintillation, proportional, or Geiger-Muller detectors. Model 12 is a count rate meter, Model 16 includes a window discriminator switch for an upper discrimination level similar to a single channel analyzer; meter read-out, audio output.
Sensitivity and Range	Count range: 500 cpm, x1, x10, x100, x1,000 Depends on the detector chosen, see other Ludlum notes.
Detectors	The detector models that may be used with the Model 12 and Model 16 are listed with Ludlum Models 2 and 3
Performance	Accuracy: +5% of full scale Discrimination range: Low discriminator level -- 2 to 60 mV Model 16, high discriminator level -- 2 to 60 mV Response time: Model 12 -- 3 or 11 seconds Model 16 -- 11 sec Output Voltage: 700 to 2,400 Vcd, adjustable
Requirements	Power: Two "D" cells, 100 hours, or optional RM-42R mercury cells, Nickel-Cadmium (rechargeable) cells Size 9 cm H x 9 cm W x 18 cm L exclusive of handle (3.4" H x 3.5" W x 7.0"L) Weight: 1.6 kg (3.5 lb)

INSTRUMENTATION
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Ludlum 2
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Features	Battery test, fluorescent mark to indicate "off" position
Reference	Manufacturer's specifications
Cost	Model 12 \$395.00 Model 16 \$445.00
Address	Ludlum Measurements, Inc. 501 Oak Street Sweetwater, TX 79556 (915) 235-5494

Neutron, Beta, Gamma, and Alpha Count Rate Meter

Ludlum Model 15

Class	Portable, handheld
Principle of Operation	Uses BF_3 detector and associated electronics for count rate meter
Sensitivity and Range	Energy sensitivity: thermal Ranges: 500, 5,000, 50,000, and 500,000 cpm
Sampling	Continuous
Performance	Meter accuracy: $\pm 5\%$ of full scale Response time - Fast: 8 second Slow: 15 seconds
Requirements	Power: - Power: Two "D" cells Size: - 25 cmx9cmx18cm (9.8" Hx3.5" Wx7"L) Weight: - 8 lbs.
Features	High voltage adjustment, calibration adjustment for each scale
References	Manufacturer's Specifications
Cost	\$775.00

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Address Ludlum Measurements, Inc.
 501 Oak Street
 Sweetwater, TX 79556
 (915) 235-5494

Hand and Clothing Monitor

Class	Stationary, Personnel Monitor	
Principle of Operation	Scintillator detectors, amplifier, meter readout, system of lights giving status (safe, wash hands, etc.)	
Specification	Model HFM6 -both hands Range: 0 - 5 derived working limits Response Time: 5 seconds can be varied 2 - 20 sec. Efficiency: α - 11% β - 26% Operating Temperature: 5° to 40°C Accuracy: α : $\pm 12\%$ for one standard deviation at MPL level of 10^{-5} $\mu\text{Ci}/\text{cm}^2$ in five second count time β : $\pm 3\%$ for one standard deviation at MPL of 10^{-4} $\mu\text{Ci}/\text{cm}$	
Requirements	Power: (190 to 260 V or 95 V to 130V, 50 to 60 Hz, at 25 V amp) Size: 75cmW x46cmH x42cmD Weight: Refer to factory	
Features	Independent α and β indication, with separate alarm presets; automatically corrects in high γ background (up to 1.5 mR/hr.)	
Cost	Refer to factory	
Address	Nuclear Enterprises, Inc. 931 Terminal Way San Carlos, CA 94070 (415) 592-8663	Nuclear Enterprises Limited Bath Road, Beenham Reading, RG7 5PR, England (075) 521-2121

Portable Rate Meters

Principle of Operation

Accepts α scintillation and β, γ GM probe, amplifiers, meter readout

Model	RM2	RM5
Range	0-10,000 cps 4 decade log	0.5-5,000 cps 3 decade log
Output Voltage	350-1300V	300-1200V
Minimum Load Impedance	50 M Ω	50 M Ω
Input Sensi- tivity	50 mV negative- going	50 mV
Sampling	Continuous	Continuous
Operating Temperature Range	10°C to 45°C	-10°C to 50°C
Accuracy	linearity better than 5%	$\pm 20\%$ F/S
Requirements Power: Size: Weight:	One 9V Ever- ready type PP9 22.9cmH x12.1 CmW x27.9cm L (9" x4.75" x11"L) 2.7 kg (6 lb)	Two type R-20 cells 24cmL x12cm W x14cmH (9.5"L x4.7"W x5.5"H) 1.6 kg (3.5 lb)

Cost

Refer to factory

Address

Nuclear Enterprises, Inc.
931 Terminal Way
San Carlos, CA 94070
(415) 592-8663

Nuclear Enterprises Limited
Bath Road Beenham
Reading, RG7 5PR, England
(073) 521-2121

Count Rate Meter

Nuclear Enterprises Model PCM 5

Class	Portable, handheld
Principle of Operation	Accepts α scintillation and β, γ GM probe, amplifier, meter readout
Sensitivity	Input Sensitivity: 50 mV Output Voltage: 300-1200 V dc present; minimum load impedance 50 M Ω Ratemeter: 1-10, 10-100, 100-1000; 500-5,000 cps Efficiency: depends on probes (see below)
Performance	Accuracy: $\pm 25\%$ for temperature -10°C to 30°C ; $\pm 35\%$ for -10°C to 50°C Temperature: -10°C to $+50^{\circ}\text{C}$ Probes listed on page 2
Requirements	Power: Type type "D" cells 1.5V; 80 hr at 4 hr/day Size: 24cmL x 12cmW x 10cmH (9.5" x 4.7" x 3.9") Weight: 1.6 kg (3.5 lb)
Features	Loudspeaker, optional power regulator for 50 cps, 220 VAC
References	Manufacturer's Specifications
Cost	With DP2/R probe \$1949.00; without probe \$1200.00
Address	Nuclear Enterprises Ltd. Bath Road, Beenham Reading, RG7 5PR, ENGLAND Tele.: 673-521-2121 Nuclear Enterprises Inc. 931 Terminal Way San Carlos, CA 94070 (415) 593-8663

Type No.		Radiation detected	Efficiency	Area
BP1	Geiger MX 16B	beta/gamma/X-ray	10 cps from 1mR/h 8kev X-ray 40 cps from 1mR/h 20kev X-ray 40 cps from 1mR/h ⁶⁰ Co field	1.36 cm ²
BP3/1	Geiger B12H	beta/gamma	5 cps from 10 ⁴ µCi/cm ² ⁹⁰ Sr/ ⁹⁰ Y 135 cps from 1mR/h ⁶⁰ Co field	-
BP4	Scintillation Anthracene	low energy beta/gamma range 120 keV upwards	15% for ¹⁴ C 30% for ⁹⁰ Sr/ ⁹⁰ Y betas	18 cm ²
BP5	Scintillation Plastic NE 102A	beta/gamma	27.5% for ⁹⁰ Sr/ ⁹⁰ Y 120 cps for 1mR/h ⁶⁰ Co field	49 cm ²
BP6	Scintillation Plastic NE 102A (thin)	low energy beta/gamma	8% for ²⁰⁶ Tl source	100 cm ²
BP7	Scintillation Anthracene	low energy beta/gamma	7% for ¹⁴ C betas	49 cm ²
AP2	Zinc sulphide on perspex	alpha	25% for ²³⁹ Pu alphas	49 cm ²
AP3	Zinc sulphide on perspex	alpha	25% for ²³⁹ Pu alphas	100 cm ²
AP4	Zinc sulphide on perspex	alpha	27.5% for ²³⁹ Pu alphas	18 cm ²
DP2/R	Zinc sulphide on NE 102A	alpha,beta,gamma	22.5% for alpha from ²³⁹ Pu 27.5% for beta from ⁹⁰ Sr/ ⁹⁰ Y 100 cps from 1mR/h ⁶⁰ Co field	49 cm ²
GP6	3/4in x 3/4in (19mm) NaI(Tl) crystal	gamma	850 cps for 1mR/h ⁶⁰ Co field 1150 cps from 1mR/h ²²⁶ Ra field	
GP7	2mm x3/4in NaI(Tl) crystal	low energy gamma and X-ray	3600 cps for 1mR/h 400 cps for 1mR/h ²²⁶ Ra

Airborne Particulates Monitor
NRC Lab Map-60

Class Laboratory

Principle of Shielded sampler, detector, flow indicator and vacuum gauge,
Operation combination ratemeter, power supply and pump-motor

Sensitivity Detectors (see Table below)
and Range

Detector Model No.	Isotope	Time to Reach 100 CPM Net at 10^{-10} uc/cc	Detector Type	Radiation Sensitivity	Background
MD-1B	Sr ⁹⁰ + Y ⁹⁰	15 min	G.M.	β γ	20 cpm
	Co ⁶⁰	26 min			
	Na ²⁴	35 min			
	Cs ¹³⁷	35 min.			
MD-3C	Po ²¹⁰	21 min	Zn S(Ag)	α	5 cpm
	Pu ²³⁹	21 min			
MD-4C	Ti ²⁰⁴	21 min	Plastic Phosphor	β	20 cpm
	C ¹⁴	70 hrs			
MD-5C	I ¹³¹	40 min.	NaI(Tl)	γ	200 cpm

Performance Flow meter 0 $\rightarrow$ 10 cfm

Requirements Power: 115 Vac, single phase
Size: 61 cm W x 105 cm H x 55 cm D (24"W x 41.5" H x 22" D)
Weight: About 90 Kg (200 lbs)

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Features	Ratemeter high level trip alarm
Cost	Price upon request
Address	Nuclear Research Corporation 1105 Industrial Highway Southampton, PA 18966

Hand and Foot Monitor
HFA-200A

Class	Stationary
Principle of Operation	G.M. tube monitoring of high energy alpha, beta and gamma radiation. Background subtraction for operation in 1 mR/hr Co-60 field. Analog readout with individual alarm adjustments.
Sensitivity	Three (3) ranges each section: 0-500, 0-5000, 0-50,000 cpm Alarm adjustable over entire range
Sampling	Selectable 2 seconds, 20 seconds
Performance	Count rate accuracy: $\pm 10\%$ Temperature: 40°F to 120°F
Requirements	Power: 120 VAC, 1 amp Size: 24" W x 56" H x 48" D Weight:
Features	Background subtraction Manual and automatic reset Automatic sequencing
References	
Cost	Price on request
Address	Nuclear Research Corporation 1105 Industrial Highway Southampton, PA 18966

Portable Survey Meters
Nuclear Research Model CP-2

Class	Portable
Principle of Operation	Detector, amplifier, counter, meter readout
Model	<u>CP Series</u>
Range	CP-2: 0.25, 250, 2500 mR/hr CP-5: 0-50, 500, 5000 mR/hr CP-10: 0-100, 1000, 10000 mR/hr
Energy Sensitivity	Alpha from 2.0 MeV Beta from 35 keV Gamma 40 keV to 2 MeV
Energy Dependence	±15% for gamma and x-ray from 40 keV to 2 MeV
Sampling	Continuous
Accuracy	±10% full scale
Requirements	Power: Four 22.5V batteries one 1.3V battery ~200 hr life Size: 23cmH x 29.2L x 9cmW (9"H x 11.5"L x 3.5"W) Weight: 16 kg (3.5 lbs)
Remarks	Designed and engineered to Oak Ridge National Laboratory specifications
Cost	\$595.00
Address	Nuclear Research Corporation 1105 Industrial Highway Southampton, Pennsylvania 18966 (215) 357-5015

Survey Meter
Nucleus Model S-102

Class	Portable, handheld			
Principle or Operation	GM tube, amplifier, meter readout			
Sensitivity and Range	PROBE	RADIATION	RANGE	COST
	X-1	Beta, gamma, X	0.5, 5, 50 mR/hr	\$90.00
	X-2	Alpha, beta, gamma, X	500, 5000, 50,000 cpm	90.00
Sampling	Continuous			
Performance	Accuracy: 5% Response Time: 2 to 10 seconds measured from 0-90% final reading			
Requirements	Power: (2) "D" cells Size: 19cmH x 9cmW x 19cmL (7.3"H x 3.5"W x 7.5"L) Weight: 1.6 kg (3.5 lbs.)			
Features	2 year guarantee, 300 hr. typical battery life with carbon-zinc cells			
Cost	\$185.00 without probe; probe \$90.00			
Address	Nucleus, Inc. P.O. Box R Oak Ridge, TN 37830 (615) 482-4041

Radiation Lab Monitor

Nucleus Model ML

Class	Laboratory		
Principle of Operation	End or side window GM tube detector, amplifier, meter readout, speaker, audible alarm		
Sensitivity and Range	Energy Sensitivity: (see probe) Ranges. 500 to 50,000 cpm full scale, linear		
	Probe:		
	<u>End Window (X-2)</u>	<u>Side Window (X-1)</u>	<u>NaI detector (X-3)</u>
	Alpha over 1 MeV	Beta over 800 keV (30%)	Very sensitive
	Beta over 150 keV (75%)	Beta over 1.7 MeV (75%)	to low energy
	X and Gamma	X and Gamma	gamma
	Price: \$90	Price: \$90	Price \$365.00
Sampling	Continuous		
Performance	Accuracy. $\pm 5\%$ full scale		
	Response Time 2 seconds		
Requirements	Power: 115 VAC, 50-60 Hz, (230 volts optional)		
	Size: 15.2cmHx27.9cmWx15.0cmD (6"Hx11"Wx6 1/4"D)		
	Weight 5 kg (11 lb)		
Features	Variable high voltage adjustment, audio. Alarms when radiation level exceeds preset adjustable value.		
Cost	\$325.00 without probe		
	Probes	X-1	\$ 90
		X-2	\$ 90
		X-3	\$690
Address	Nucleus, Inc. 461 Laboratory Road Oak Ridge, Tenn. 37830 (615) 482-4041

Technical Associates
TBM-3

Class	Survey
Principle of Operation	Built-in pancake (2 inch) GM thin window looks thru hole on bottom of case
Sensitivity and Range	0-500, 5000, 50,000 cpm 0-0.15, 1.5, 15 mr/hr
Sampling	Continuous
Performance	
Requirements	Power: 1 av transistor battery Size: 7.6 cm (3") x 13.3 cm (5 1/4") x 6 cm (2 1/4") 22 oz total weight Sensitive to α, β, γ - small very light one hand operation
References	
Cost	\$295.00
Address	Technical Associates 7051 Eton Avenue Canoga Park, CA 91303 (213) 883-7043

Hand and Shoe Monitors

Technical Associates

HSM-10A, 10AM, 10BS, 10G

Class	Stationary, Personnel Monitor
Principle of Operation	Geiger tubes detect radiation, illuminated readout panel has projected messages that automatically warns user of contamination and contamination region, check incomplete and other messsge (depending on model), counters or ratemeters also indicate count or count rate - alarms also.
Sensitivity and Range	Model HSM-10A: Beta-gamma (LED counters) Model HSM-10AM: Beta-gamma (ratemeters) Model HSM-10BS: Alpha-beta-gamma (LED counters) Model HSM-10G: Alpha and beta-gamma- 1) separately, but simultaneously 2) dynamic digital background subtraction
Sampling	Continuous
Performance	Accuracy, linearity $\pm 5\%$ full scale
Requirements	Power: 105 - 125V; 60 Hz; 300 watts (or 230V 50-60 Hz) Size: 172.1cmH x68.6cmW x70.0cmD (67 3/4"H x27"W x24"D) Weight: 272.2 kg (600 lbs.)
Features	1) Easy to read messages on illuminated panel in addition to count readout 2) Alpha-beta-gamma or beta-gamma (depend on model) 3) Adjustable alarm points 4) Background subtract on some models
References	Manufacturer's Specifications

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Cost	<u>HSM</u>	<u>\$</u>	<u>HSM</u>	<u>\$</u>
	10A:	4490.00	10BS:	5375.00
	10AM:	3850.00	10G:	7380.00
Address	Technical Associates 7051 Eton Ave. Canoga Park, CA 91303 (213) 883-7043

Portable Survey Meters

CP-6

CP-44

ERJ-9

Class	Portable
Principle of Operation	Ionization chamber, MOSFET amplifier, meter readout
Models	See Table on page 2.
References	Manufacturer's Specifications
Address	Technical Associates 7051 Eton ave. Canoga Park, CA 91303 (213) 883-7043

Model	CP-44	CP-5	CP-6	CP-7	*ERJ-9
Type	Cutie-Pie	Cutie-Pie	Cutie-Pie	Cutie-Pie	Juno
Range	0-25, 250, 2500, 25,000 mR/hr 0-500,5000, 50,00 mR/hr (CP-44A)	6 linear ranges from 2.5 mR/hr to 250 R/hr	6 linear ranges from 1 mR/hr to 100 R/hr	6 linear ranges from 0.5 mR/hr to 50 R/hr	6 linear ranges 0-2.5,25,250mR/hr 0-2.5,25,250R/hr
Sampling	Continuous	Continuous	Continuous	Continuous	Continuous
Ionization Chamber Volume	589.9 cc 36 cubic in.	589.9 cc 36 cubic in.	1688.3 cc 103 cubic in	1688.3 cc 103 cubic in	589.6 cc 36 cubic in.
Operating Temperature	-37.2°C to 51.7°C -35°F to +125°F	-37.2°C to 51.7°C -35°F to +125°F	-37.2°C to 51.7°C -35°F to +125°F	-37.2°C to 51.7°C -35°F to +125°F	-37.2°C to 51.7°C -35°F to +125°F
Response time:					
Accuracy	±10% FS full scale mR/hr ranges ±15% FS R/hr ranges	±10% FS (mR/hr) ±15% FS (R/hr)	±10% FS (mR/hr) ±15% FS (R/hr)	±10% FS (mR/hr) ±15% FS (R/hr)	±10% FS (mR/hr) ±15% FS (R/hr)
Drift	<1% per °F at room temperature	<1% per °F at room temperature	< 1% per °F at room temperature	< 1% per °F at room temperature	<1% per °F at room temperature
Power	One "D" cell	One "D" cell	One "D" cell	One "D" cell	One "D" cell
Size Excluding Handle	33cm x16.5cm x10.2cm (13" x6 1/2" x4")	33cm x16.5cm x10.2cm (13" x6 1/2" x4")	33cm x16.5cm x10.2cm (13" x6 1/2" x4")	33cm x16.5cm x10.2cm (13" x6 1/2" x4")	24.1cm x14.6cm x10.2cm (9 1/2" x5 3/4" x4")
Weight	2 kg (4.5 lb)	2 kg (4.5 lb)	2 kg (4.5 lb)	2 kg (4.4 lb)	27 kg/6 lbs)
Remarks	Flat response at all energies	Flat response at all energies	Flat response at all energies	Flat response at all energies	Flat response at all energies
Cost	\$570.00	\$630.00	\$755.00	\$860.00	\$940.00

*ERJ-9-4R Juno same as ERJ-9 except 4 ranges: 0-50, 500, 5000, 50,000 R/m \$780.00.

Portable Rate Meter

PUG-1

FS-8

SML-2

Class	Portable
Principle of Operation	Accepts GM or scintillation detector, amplifier, counter, readout device
Models	See Table on page 2
References	Manufacturer's Specifications
Address	Technical Associates 7051 Eton Ave. Canoga Park, CA 91303 (213) 883-7043

Model	SML-1, SML-2	FS-8 Series	PUG-1	PUG-1AB,	PUG-1E
Type	Portable (rechargeable)	or laboratory (AC operation)	Portable		
Range	0-500,5000,50,000 500,000 cpm	Preset count: 1 to 99,999 Preset time (seconds) 0.1 to 9999.9 Recycling time (seconds) 0.1 to 9999.9	0-500,5000 50,000 cpm	0-500,5000,50,000 cpm	
Sampling	Continuous	Settable on preset count or time with or without recycling	Continuous		
High Voltage	0 to 2000V				
Input Sensitivity	100 mV	100 mV			
Accuracy	±2% FS (meter)	1% (time base)	±5% FS		
Power	Four NiCad C batteries (included) and AC 115V 50-60Hz (or 230V)	Four NiCad D batteries or 115V, 50-60 Hz (or 230V)	Three "D" cells		
Size	cm 20.3cmW x15.2cm x12.7H in. (8"W x6"D x5"H)	24.1cm x26.7cm x8.9cm 9 1/2" x10 1/2" x3 1/2"	10.2cmH x10.8cmW x25.4cmL 4"H x4 1/4"W x10"L		
Weight	1.8 kg (4 lbs.)	3.0 kg (6 1/2 lbs.)	1.8 kg (4 lbs.)		
Remarks	Set table alarm (SML-2) Alarm indicators (SML-2) Meter readout Built-in speaker with volume control	Settable alarm Digital readout Built-in speaker with volume control FS-8W (FS-8 AC line operation only) \$780.00	Zero suppress (PUG-1E) for use with scintillator probes		
Cost	SML-1: \$380.00 SML-2: 485.00	\$870.00	PUG-1:\$215.00 PUG-1E \$395.00 PUG-1AB: \$270.00		
		FS-8A (FS-8 with built-in amplifier-1mV sensitivity) \$950.00 FS-8C (FS-8 with built-in amplifier and single channel analyses - 1mV sensitivity) \$1075.00 FS-8W (FS-8 AC line operation only) \$780.00

Air Monitors

Technical Associates Model BAM-3A,3B,3H

Class	Laboratory		
Principle of Operation	Motor-blower pulls in through filter and open gas chamber, viewed by an end-window GM detector and sodium iodide detector, respectively. Log ratemeter, single channel analyzer, and recorder.		
Sensitivity and Range		BAM3A	BAM3B
		10 to 100,000 cpm	10 to 100,000 cpm
<u>Sensitivity:</u>		(at 5CFM) 6×10^{-10} $\mu\text{Ci/ml/min}$ exposure	3×10^{-9} $\mu\text{Ci/ml/min}$ at 5CFM (iodine channel)
			6×10^{-10} $\mu\text{Ci/ml/min}$ at 5CFM (Beta-Gamma channel)
			BAM3H
			10 to 100,000 cpm
			6×10^{-10} $\mu\text{Ci/ml/min}$ at 5 CFM (Beta Particulates)
			6×10^{-11} $\mu\text{Ci/ml/min}$ at 5CFM (Alpha Particulates)
			6×10^{-7} $\mu\text{Ci/ml}$ (gas)
Sampling	Continuous		
Performance	Accuracy: $\pm 10\%$ except for resolution loss and statistical variation		
	Air flow: Up to 10 CFM

Requirements	Power: 115 VAC, 50-60 Hz Size: BAM3A: 36"Lx24"Wx42"H BAM3B: 36"Lx24"Wx42"H BAM3H: 42"Lx24"Wx42"H Weight: BAM3A: 635 lbs. BAM3B: 640 lbs BAM3H: 960 lbs.
Features	Audible and visual alarms Can add features by adding modules
References	Manufacturer's Specifications
Cost	BAM3A: \$3650.; BAM3B: \$5500.; BAM3H: \$7640.
Address	Technical Associates 7051 Eton Avenue Canoga Park, CA 91303 (213) 883-7043

Radiation Survey Meters

Texas Nuclear Models 2651, 2652 and 2590 Series

2651

2652

2595 and 2596

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Principle of Operation: Ionization chamber, amplifier, oscillator meter readout

Model	2590 Series		2650 Series	
	2595 (low range)	2596 (high range)	2651	2652
Range				
Exposure rate	0-10,100,1000 mR/h	0-10,100,1050 R/hr	0-0.1,0.3,1.0,3.0,10,30 100 mR/hr	
Accumulated exposure	0-0.1,1,10 mR	0-0.1,1,10 R	Rate: 0-150,1500,15000, 150,000 cpm	
Sensitivity	Minimum calculated alpha energy: 1.7 MeV Minimum calculated beta energy: 0.025 MeV			
Sensitive volume	500 cc.	5 cc		
Window thickness	~1 mg/cm ² aluminized mylar		30 mg/cm ²	1.5-2.0 mg/cm ²
Accuracy	+10% full scale calibrated with 1.17 and 1.33 MeV ray ⁶⁰ Co		+10% full scale	
Operating Temperature	0°C to 55°C 32°F to 131°F		0°C to 50°C 32° to 122°F	
Time constants	3 sec	0.5 sec	Selectable 4 or 8 secs. on 0.1, 0.3,1.0mR/hr range - fixed 4 secs. on on 3 and 10mR/hr range - fixed 2 secs. on 30 and 100 mR/hr range	
Requirements				
Power	Three 1.5V "D" cells, 480 hr continuous life		Four 1.5V "D" cells, 300 hr @ 8 hr/day	
Size	37.4 cm L x 10.2 cm W x 21.6cmH (14.75" x 4" x 8.5")		12.7 cm H x 10.2 cm W x 30.5 cm L (5" x 4" x 12")	
Weight	2 kg (4.5 lbs)		2 kg (4.5 lbs)	
Cost	\$1025	\$1025	\$675	\$675

Address: Texas Nuclear
P.O. Box 9267
Austin, Texas 78766
(512) 836-0801 Ext. 214

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Portable Survey
Victoreen
May 1980

Radiation Survey Meters

Victoreen

ENERGY RESPONSE

Address

Victoreen Inc.
10101 Woodland Avenue
Cleveland, OH 44104
(216) 795-8200

MODEL	490 (Thyac III)	491	492	493
Radiation Detected	Alpha, beta gamma depending on probes	Alpha, beta, gamma, x-ray, depending on probes	Gamma, x-ray above 60 keV	Alpha, beta, gamma, x-ray depending on probes
Detector Range	0-800,8000 80,000 c/m & 0-.2, 2,20 mR/hr (489-4 probe)	0-150,450,150,000 cpm in seven overlapping linear ranges	0-10;0-100; 0-1,000 mR/hr	0 to 300,3000,30,000 cpm corresponding to 0-0.5, 5, 50 mR/hr with 493-50 probe
Energy Dependence	Depends on probe	Depends on probe	±20%, 80 keV to 1.0 MeV	Depends on probe
Precision	±10% F/S	±10% F/S	±20% F/S	±20% F/S
Power Requirement	2 "D" cells	4 "D" cells	2 "D" cells	2 "D" cells
Features	Variable response time	Energy correction	Energy and temperature correction	Energy and temperature
Overall Dimensions	22.2cmL x11.4cmW x17.8cm (8 3/4"L x4 1/2"W x7"H)	11.4cmW x22.2cmL x17.8cmH (4 1/2"W x8 3/4"L x7"H)	11.4cmW x22.2cmL x16.8cmH (4 1/2"W x8 3/4"L x6 5/8"H)	11.4cmW x22.2cmL x16.9cmH (4 1/2"W x8 3/4"L x6 5/8"H)
Net weight	1.82 kg (4 pounds)	2.25 kg (5 pounds)	1.6 kg (3.5 pounds)	1.6 kg (3.5 pounds)
Cost	\$350.00 (less probe)	\$425.00 (less probe)	\$300.00	\$235.00 (less probe)

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MODEL	496	498	740F	740G	470A	885
Radiation Detected	Alpha, beta gamma, x-ray depending on probe used	Gamma, X-ray Beta	Alpha, above 3.5 MeV Beta, above 70 keV gamma, above 7 keV; ion chamber	Alpha, Beta, Gamma	Alpha above 8 MeV Beta, above 120 keV gamma and x-ray, above 10 keV; ion chamber	X-Ray Gamma
Detector Range	0-800;0-8,000'; 0-80,000 cpm	0-1;10;100mR/hr 0-1 R/hr	0-25,250,2500, 25,000 mR/hr	0-100,1000,10,000 100,000 mR/hr	Exposure rate mode: 0-3,10,30,100,300,1000 mR/hr or R/hr (rate) Integrate mode: 0-3 10,30,100,300,1000 mR (integrate)	0-999mR
Energy Dependence	Depends on probe	Depends on probe	10% 7 keV to 100 keV +15% 40 keV to 2 MeV	Same	+15% 10 to 300 keV (bare cnamber) +10% 40 keV to 2 MeV (sleeve on)	50 keV to 1 MeV
Precision	±10% F/S	±20% F/S	±10% F/S	±10% F/S	±10% F/S	±15%
Power Requirement	2 "D" cells	2 "D" cells	Four 22.5V batteries	Four 22.5V batteries	Two 1.5V "D" cells and four 22.5V No. 505 cell	One 9 volt batteries
Features	Built-in speaker	High range	Rugged proven design	High range CP	Measures complete energy spectrum over 2π solid angle	
Overall Dimensions	22.2cmL x11.4cmW x17.8cmH (8 3/4"Lx4 1/2"W x7"H)	4 1/2" x8 3/4" x7"H (including probe)	29cmL x9cmW x23cmH 11 1/2"Lx3 1/2"Wx9"H	29cmL x9cmWx 23cmH 11 1/2"Lx3 1/2"Wx9"H	27.9cm x12.1cm x24.1cm 11"x4 3/4"x9 1/2"	2 1/2" x4 1/2" x1"
Net weight	1.8 kg (4 lbs)	1.8 kg (4 lbs)	1.7 kg (3.75 lbs)	1.7 kg (3.75 lbs)	1.8 kg (4 lbs)	6 oz.
Cost	\$310.00	\$340.00	\$625.00	\$650.00	\$850.00	\$190.00

Survey Meters
XETEX Models 306A and 308A

Class	Portable, handheld	
Principle of Operation	Halogen-quenched 1.2 in. diameter GM detector with 1.5 - 2.0 mg cm ⁻² window, digital logic and display (306A), and meter readout (308A)	
Sensitivity	α, β, γ and X rays, 0.002 μCi lower detection limit	
	<u>306A</u>	<u>308A</u>
Range	0 - 9.99 kcpm Overrange indicator	0 - 100, 0 - 1000, and 0 - 10,000 cpm range switch
Sampling	Push button	Continuous
Performance	Accuracy: $\pm 10\%$ FS Response time: 6 seconds Operating temperature: 0 - 50°C	$\pm 10\%$ FS 12 seconds (X1), 3 seconds (X10 and X100) 0 - 50°C
Requirements	Power: 4 AA alkaline penlite batteries, life ~100 hrs Size: 7.1 cm x 15.5 cm x 3.89 cm (2.81" x 6.09" x 1.55") Weight: 0.27 kg (9.5 oz.) w/batteries	One 9V transistor type battery, life ~200 hrs 7.1 cm x 15.5 cm x 4.7 cm (2.81" x 6.09" x 1.87") 0.33 kg (11.5 oz.) w/battery
Features	1. Autoranging, three decade coverage (306A) 2. Zero stable (308A) 3. Battery test 4. LED "counts" lamp	
Reference	Manufacturer's specifications	
Cost	\$330.00 (306A) \$330.00 (308A)	
Address	Xetex, Inc. 660 National Ave. Mountain View, CA 94043 (415) 964-3261

III.B. Alpha-Particle Instrumentation

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1. Introduction

Many different types of instruments are capable of detecting alpha particles. However, more is involved in alpha measurements than a mere alpha detector, both because alphas have a short range in matter and because many applications require measurements of specific activity (curies/gram) or identification of the emitter. In this section, we shall treat several different topics, including both instrumental considerations and techniques for measurements in specific media.

The topics to be covered are:

- Gross-alpha instruments (such as hand-held meters) for radiation survey work and radiological protection. These measure particle emission rate (particles $\text{cm}^{-2} \text{sec}^{-1}$) or, less frequently, dose equivalent (rem).
- Alpha spectrometers for identifying individual alpha energies in a sample that may contain more than one alpha emitter.
- Techniques for measurements of specific activity (pCi/m^3 or pCi/g) in air and water.
- Chemical separation techniques, developed primarily for use with alpha spectrometry. Because they are not treated elsewhere in this volume, chemical separation of polonium (atomic number 84), protactinium (91) neptunium (93), and americium (95) are treated individually.

These categories overlap substantially for many applications, as will be seen in the discussions below.

Alpha-emitting elements treated more fully in their own separate sections elsewhere in this volume are not discussed individually here. These are:

radon (86) and its daughters
radium (88)
uranium (92) and thorium (90)
plutonium (94).

The section on plutonium also treats, to the extent that they are available, methods for monitoring other transuranics, particularly in air.

2. Physical Considerations

An alpha particle is a helium-4 nucleus, with a mass of almost 4 times that of a proton and with charge +2. It can be thought of as 2 protons and 2 neutrons, bound together. Alphas were discovered, named, and studied by Ernest Rutherford in the earliest days of nuclear physics. Emission of alphas is only found among the heavy elements: with one exception (^{147}Sm), naturally occurring alpha emitters are all heavier than lead ($Z = 82$). This section will not go into detail about the characteristic features of alpha emitters, except to discuss those features essential for measurement.

The most characteristic feature of alpha particles is their high specific ionization density as they traverse matter, and their consequent short range. This feature is relied upon in virtually all detection methods. It also limits the sensitivity and capability of alpha-detecting instruments.

Figure 1 (from Ref. 1) shows the range of alpha particles in air. Figure 2 (from Ref. 2) shows the distribution of ionization versus energy for alphas slowing in air. It can be seen both that the range is short, and that the specific ionization is high and peaked at the end (the "Bragg peak"). Figure 3 (from Ref. 1) shows the energies of all of the known heavy alpha-emitters. From the figure, it can be seen that nearly all alpha particles have energies in the range from 4 to 9 MeV.

3. Measurement Considerations

Because of their very short range, the hazard from alpha emitters is almost entirely the result of inhalation or ingestion; skin dose is usually a minor hazard. For many of the important alpha-emitters (e.g., plutonium and radon daughters), inhalation is the most serious hazard. Intake may be direct, as where food, water, or air contains alpha-emitting materials; or indirect, as when for example hands, clothing, or tools pick up active material and later transfer it to the food or water which is consumed.

Alpha survey techniques are often directed at the "indirect" hazard class and hence are designed merely to indi-

Fig. 1. Range-energy curve for alpha particles in air at 15°C and 760 mm Hg.
(Reproduced from K. Siegbahn, Ref. 1, with permission of North-Holland Publishing Co., © 1965).

Fig. 2. Bragg curve of specific ionization by alpha particles in air at standard temperature and pressure.
(Reproduced from H. Cember, Ref. 2, with permission of Pergamon Press, © 1969).

Fig. 3. Alpha-decay energy vs radionuclide mass number. (Reproduced from K. Siegbahn, Ref. 1, with permission of North-Holland Publishing Co., © 1965).

cate the presence of alpha emitting material in an area, particularly on a surface. Survey techniques may also be intended to provide a rapid or convenient measure of the alpha activity in some bulk material or even in air. Except for air monitoring, the short range of alphas often makes direct alpha monitoring in such media difficult. Monitoring may, however, be accomplished by chemical separation of the material (more often a prelude to quantitative measurements of alpha activity than of surveys) or by reliance on gamma or x-ray emission associated with the alpha decay sequence.

Determination of the alpha activity (in pCi per unit volume or mass) may be accomplished in a number of ways for various media (e.g., air, water, soil, food) or for biological samples (e.g., blood, urine, tissue). Except for air, alpha monitoring usually requires separation of the alphas activity from the bulk sample by chemical procedures, or by physical means such as filtering, and preparation of an effectively "thin" sample appropriate to the detector system being employed. To some extent, this difficulty is offset by the possibility of using alpha spectrometry and by an advantage in alpha particle detection: The two salient physical

properties of alphas (relatively high energy and high specific ionization) allow design of detectors with extremely low backgrounds; indeed, backgrounds in the more sensitive detectors are typically dominated by alpha contamination of one sort or another. Another background contribution can come from high-LET (high linear energy transfer) particles associated with cosmic radiation, but this usually is not important. Contamination is usually from either the materials used in the construction of the instrument or from contamination from alpha-active samples, either directly or due to alpha recoils during counting. Care in the choice of construction materials is essential for a low-background alpha detector. An alternative form of background occurs when radionuclides other than the one(s) to be detected yield interfering alpha particles.

An alternative technique for identification of alpha emitters is the use of decay-chain information to isolate one nuclide from a mixed sample. This can be done, for example, by the buildup of a daughter (if the daughter can be measured easily, perhaps by its own decay). Since gamma detection or spectroscopy is relatively easy, it is often used where appropriate. Alterna-

tively, $\alpha - \gamma$ or $\alpha - \beta$ "coincidences" can be used when the daughter's lifetime is extremely short and its γ or β emission can be counted in coincidence with the parent emission. This can be very specific, as well as providing a very low background.

Emission of x rays by the daughter immediately following alpha-emission is a common occurrence for the heavier elements, particularly the transuranic elements. The alpha emission leaves the daughter in an atomic excited state, which de-excites by x-radiation. The γ and x emissions are essentially simultaneous, and in some cases the x rays are a better signature than the alpha emission for elemental identification. This is especially true when samples cannot be prepared thin enough for good alpha counting: the x rays will penetrate further and can be detected instead. A good discussion of the usefulness of x-radiation has been given by Swinth (Ref. 3), who has also tabulated ratios of x to α intensity for a number of important transuranic isotopes (Np-237; Pu-238, 239, 240, 241; Am-241; Cm-242, 244). This technique is commonly used for plutonium measurements, which are discussed in a separate section ("Plutonium") in this volume.

Techniques for measuring gamma and x rays are discussed in their own sections in this volume. Application of these techniques to monitoring of alpha emitters is treated in the sections on individual radionuclides in Part IV.

4. Measurement Techniques

a. Techniques that Measure Particle Flux or Dose Equivalent

There are two common types of measurements in this category:

- A measurement of particles/seconds may suffice, such as when alpha activity needs to be only roughly determined (for example, a spill of alpha-activity).
- A direct measurement of dose equivalent may be required, such as when a worker is unavoidably exposed to an alpha

environment which could result in a significant, measurable dose equivalent.

For spills actual alpha-dose-equivalent is usually not required because prudent practice demands cleaning up the activity rather than permitting exposure to it for any length of time. In either case, it may or may not be possible to ascertain the species of radionuclide responsible. For measurements of dose-equivalent, knowledge of the species (more specifically, the alpha energy) is important but not crucial. For the counting-rate type of measurement, it is less often an important parameter.

Calibration of a ratemeter type of instrument may be, but is not usually, a problem. The most important difficulty in calibrations is assuring a known geometrical set-up. For calibrating a dose-equivalent instrument, a known source can be used, but care must be exercised because of the finite thickness of the source itself: alphas of degraded energy, emerging from below the surface of the standard source, can influence the accuracy of the measurement. The source must be thin enough that none of the alphas are too degraded to register in the counter.

Particle emission rates are commonly measured for two different kinds of media: surfaces with possible contamination, and media that can be specially prepared for measurement (usually by chemical preparations; see also alpha spectrometers, below).

For instruments used for survey, including detection of surface contamination, a primary requirement is a large-area thin window. Typical windows in alpha detectors have thicknesses of a few mg/cm^2 : much thinner windows are occasionally used, but are fragile and usually not necessary, while thicker windows are too opaque. Once an alpha penetrates the window, it can be easily detected since any of several ionization-sensitive instruments can be used (proportional counters, ionization chambers, scintillation counters, semiconductor detectors). Common survey instruments employ air proportional counters or ZnS scintillators. Detector types are discussed below, often as alpha spectrometers, although any of these could also be used for measuring particle flux or gross alpha.

The ZnS technique for survey in air, as well as for measuring small specially-prepared samples, deserves special mention because it is designed especially for alpha detection, i.e., the use of zinc sulfide as a scintillator. Zinc-sulfide crystals with silver impurities (ZnS(Ag)) can be prepared as screens of about 15 mg/cm² thickness and/or applied to a backing as a thin layer. The detection efficiency for alphas should be at least 95% for a properly constructed screen (Ref. 4). The ZnS(Ag) phosphor is viewed by a photomultiplier tube, preferably with low noise. This combination is ideal for use as a ratemeter or even as an analysis device, as in a Lucas cell (see measurements in air). But it is not useful for energy spectrometry, since the energy resolution is so poor (poorer than for any of the systems to be discussed in the next section). However, as noted in the EML Procedure Manual (Ref. 5), the activated zinc sulfide phosphor gives a large alpha pulse, and excellent discrimination against beta and gamma radiation can be obtained. Details of sample analysis techniques have been discussed by Hallden and Harley (Ref. 6). Techniques for measurement of airborne alpha emitters are discussed in Section (c); in their simpler form, these may be used for general surveys. A general requirement for scintillator-phototube systems is, of course, that they be shielded from external light.

Integrating techniques sometimes employ materials that may be examined for total alpha counts. Radon and daughter measurement methods have been developed using thermoluminescent materials and plastics that show alpha tracks after etching. The latter technique has been developed primarily for uranium surveys. See air measurements, below.

b. Alpha Spectrometers

Alpha spectrometry (i.e., the separation and measurement of an energy spectrum) has been used since the earliest days of nuclear physics: indeed, Rutherford and Geiger built a proportional counter for this purpose in 1908. There are two different approaches to spectrometry. The first measures ionization, and the second relies on momentum selection in a magnetic field. The latter will not be discussed here, since it is predominantly used for research. An excellent dis-

cussion of this class of instruments is given by Siegbalm (Ref. 1). We discuss the common alpha spectrometric instruments, which rely on ionization, in brief separate treatments (see also Section I.E in the introduction to this volume):

- (1) pulse ionization chambers
- (2) gas proportional counters
- (3) liquid scintillation counters
- (4) solid scintillation counters
- (5) semiconductor ("solid state") detectors.

Besides these, there are several other ionization techniques which have found occasional use in various applications, but which will not be discussed here. Among them are nuclear emulsions, cloud chambers, and differential range/absorption techniques.

(1) Pulse Ionization Chambers. The pulse ion chamber is widely used because of its excellent resolution. This is an ionization chamber operated so as to count individual alpha particles as "pulses" of ionization. There are two different ways in which the alpha source might be introduced into the counter, either as a gas or (more commonly) as a thin solid film located on or near the anode. A detailed discussion of the techniques for improving resolution of this instrument has been given by Curran and Wilson (Ref. 7). Resolution in an ordinary plane-parallel chamber is limited because the total electronic signal depends on the distance of electrons from the collection plane, so that the signal depends on the position, length, and orientation of the alpha track. A grid inserted between the two planes substantially reduces this effect and hence makes possible much-improved resolution. This type of instrument is known as a "Frisch Grid" ion chamber, after its inventor, O. R. Frisch. Resolution of about 30 to 40 keV (full width, half maximum) at about 5 MeV can now be obtained routinely, and even ~20 keV has been reported (Ref. 8). The most difficult part of this measurement method, and the part which limits its convenience and accuracy (and hence its widespread use), is source preparation. Chemical separation techniques are discussed below.

(2) Gas Proportional Counters. The gas proportional counter

uses an alpha sample similar to that of the pulse ion chamber. The principal advantage over the pulse ionization chamber is the ability to discriminate more completely against background beta and gamma radiation. A drawback is that energy resolution is inherently poorer (no better than ~ 100 keV full-width half-maximum in a typical well-designed system), due to the resolution spread from the charge multiplication process. Moreover, if used for spectroscopy, choice of gas is important.

If, however, the application is alpha counting (as discussed in Air and Water Monitoring), gas type and quality are not critical, although it is important that the density be high enough to stop the alphas before they strike the chamber wall. This contrasts markedly with the case of beta proportional counting, where gas type and quality are critical. In a typical proportional counter for alphas, a pulse-height gain of about 100 is sufficient, since the ionization is dense and the primary ion-pair count is high. Again, this is in contrast to beta-counting, where gains of $\sim 10,000$ are common. Figure 14 (Ref. 2) shows the performance of a typical proportional counter when the activity is about one-half from alpha and one-half from beta. At point A, the alpha pulses are just big enough to trigger the discriminator. The "alpha plateau" occurs when all of the alphas produce counts, but the betas are of insufficient pulse height to trigger the discriminator. At point B, the betas begin to count, and the higher plateau is the sum of (alpha + beta) counts. This relatively clean separation is characteristic of a typical proportional counter.

Fig. 4. Alpha and alpha-beta counting rates as a function of voltage in a proportional counter. (Reproduced from H. Cember, Ref. 2, with permission of Pergamon Press, © 1969).

Curran and Wilson (Ref. 7) discuss the detailed problems and features of gas proportional counters.

(3) Liquid Scintillation Counters. Liquid scintillation counting has been developed primarily for counting and low-energy, low activity beta activity, and its application to alphas has not been widespread. Details on this technique are found in the section on "Tritium" elsewhere in this volume. We here point out aspects that are particular to alpha monitoring.

The basic technique requires incorporation of the material to be monitored into a small amount of liquid scintillator, which is then introduced into the detection system. Using a very small sample (250 μ liters of scintillator solution) viewed by a single phototube, Horrocks (Ref. 9) has achieved energy resolutions of about 6% (full width, half maximum) for 4 to 7 MeV alphas. A typical spectrum is shown in Fig. 5 (from Ref. 9), where 4.8-MeV alphas from ^{233}U are clearly distinguished from the 5.5-MeV alphas from ^{238}Pu . The advantages of a scintillator/photomultiplier technique include ease of operation, particularly in sample preparation; high and well-understood efficiency (essentially 100%); and fast-timing capability. This technique may be particularly appropriate for water sampling or where chemical extraction is required for sample preparation in any case.

Fig. 5. Pulse height spectrum for (a) 4.8-MeV alpha particles of ^{233}U and (b) 5.5-MeV alpha particles of ^{238}Pu in liquid scintillator. (Reproduced from D. L. Horrocks, Ref. 9, with permission of the American Institute of Physics, © 1964).

A principal difficulty with alpha counting in liquid scintillators has been quenching, i.e., decrease or variability in light output due to interference in the scintillation mechanism, or absorption of the emitted light, by the sample itself. McDowell and co-workers (Ref. 10,11) have developed sample preparation techniques that minimize quenching. In addition, pulse shape discrimination techniques for liquid scintillators have been adapted (Ref. 11,12) to alpha counting in order to distinguish alpha-induced pulses from electron-induced pulses originating from cosmic rays or from beta activity in the sample or counter. (These techniques are based on dependence of the pulse shape on the ionization density.) Using these techniques, the minimum background under an alpha peak can be reduced from the 5 counts per minute obtained with commercially available beta liquid scintillator systems to the vicinity of 0.01 cpm (Ref. 11).

A possible disadvantage to this technique is that the light output from alpha scintillation is not energy-proportional because of the saturation of the scintillation process at the high specific ionizations involved. This is shown in Fig. 6 (from Ref. 9): the pulse height is a linear, but not proportional, function of alpha energy.

(4) Solid Scintillation Counters. The main advantages of solid scintillators for alpha spectroscopy seem to be their availability in large

Fig. 6. Pulse height - energy relations for alpha particles and electrons in a liquid scintillator, (a) alpha particles o, (b) electrons . (Reproduced from D. L. Horrocks, Ref. 9, with permission of the American Institute of Physics, © 1964).

sizes, and the feature of particle identification by pulse-shape discrimination. However, solid scintillators are hampered by two problems: (a) The resolution is poorer by a significant factor than for several other techniques, since the required energy of about 300 eV per photoelectron is so much higher than the ~30 eV per ion-pair in gases or the 3 eV per electron/hole pair in semiconductors. Typical alpha particle resolutions for scintillator detectors are in the few-percent range (several hundred keV out of several MeV). (b) Sources must be externally counted since the scintillators ordinarily need to be wrapped or otherwise contained for light-tightness, making the introduction of the alpha-active sample difficult. When these do not present insurmountable difficulties, the scintillator/photo-multiplier system has much to recommend it. See, for example, the discussion of liquid scintillators. Reference 13 discusses the utility and limitations of solid scintillator systems. Presently, they are not often used for alpha spectroscopy. ZnS(Ag) scintillator is often used for gross alpha, as discussed below.

(5) Semiconductor ("Solid-State") Detectors. Commercially available and laboratory spectroscopic systems depend principally on semiconductor (often called solid state) detectors. The introduction of such detectors in the 1960s revolutionized high-resolution particle and photon spectroscopy for many research and practical applications. For heavy charged particles relatively thin detectors are adequate, as opposed to gamma spectroscopy, where detectors with large volumes are often required. For alpha particles, with their relatively short range and high specific ionization, it is necessary that the sensitive region of the detector extend very nearly to the surface where alphas enter the detector, since any "dead" material absorbs energy from alpha particles. Surface barrier and diffused junction silicon detectors have been found useful for detection of alpha particles with energy up to tens of MeV.

Figure 7 shows such detectors schematically; each is based on a silicon wafer. The diffused junction detector typically is formed from p-type material, i.e., material rich in acceptor sites, by diffusing into the surface of the wafer a donor such as phosphorous, thereby causing an n-type region, con-

Fig. 7. Schematic representations of diffused junction (a) and surface-barrier (b) detectors. (Reproduced from W. J. Price, Ref. 14, with permission of McGraw Hill, © 1964.)

taining many free electrons. Application of a reverse bias to the n-p junction depletes a layer of material of free carriers and this depletion region constitutes the sensitive volume of the detector. Holes and electrons produced in it by the ionizing action of incident radiation are collected in the electric field in the depletion layer, thereby producing signals. The surface barrier detector similarly forms a depletion region, but the junction is normally formed by etching n-type thin silicon and depositing on this a thin layer of gold (i.e., about $50 \mu\text{g}/\text{cm}^2$). For both detector types, ionizing particles traversing the depletion region lose energy by producing electron-hole pairs that are rapidly collected at the positive and negative detector electrodes. Discussion of the theory, fabrication, and operation of semiconductor detectors is presented in detail in numerous publications (Ref. 14-15). A brief review of semiconductor detectors is given in Ref. 16. The electronic units required for operation of semiconductor detectors are widely available in modular form and their application is discussed in manufacturer's literature.

Detectors with depleted depths from less than ten microns to several milli-

meters are commercially available. For naturally occurring alpha particles, depths with a minimum of 20 microns (at 4 MeV) to 60 microns (at 9 MeV) are acceptable. Detectors with effective areas up to about 20 cm^2 are available, and their detection efficiency for alphas is virtually 100% for particles entering the detector area. The small physical size of these detectors may be an advantage or a disadvantage, depending on the application.

A semiconductor detector has several distinct advantages over a gas-filled detector:

- For a given amount of energy lost through ionization, the number of charge carriers (corresponding to ion pairs in the gas) is greater by a factor of as much as 10. This decreases the fractional statistical fluctuations in the ion collection process, and provides very good energy resolution.
- With detectors now available, recombination effects are less important than for a gas; this is particularly important when ionization is locally dense, as is the case with alphas (or fission fragments).
- Charge collection times are in the nanosecond range, allowing fast time resolution where required.

Among the difficulties that may be associated with semiconductor detectors are:

- The detectors can be fragile, requiring careful storage and handling.
- The output signals are very small, so that low-noise electronic amplification is required.
- They can become irreversibly contaminated, so that the background may slowly rise. (However, diffused junction detectors are often marketed as decontaminable.)

Figure 8 shows a high-resolution alpha-particle spectrum. Resolutions of 15 keV (full-width-half-maximum) are commonly obtained using junction or surface barrier detectors, but surface barrier detectors generally have slightly thinner entry windows, often

Fig. 8. An early high-resolution spectrum of alpha particles from the decay of ^{212}Bi . (Reproduced from D. E. Watt and D. Ramsden Ref. 17, with permission of the Macmillan Publishing Company, © 1964).

an advantage for high-resolution alpha measurements.

c. Alpha Measurements in Air

Most alpha emitters in air are present in particulate form (an important exception being radon, which is discussed in its own section). For this reason, measurements of alpha activity in air often depend on collection of airborne particulates on a filter whose collected activity is measured. This measurement may be accomplished by several methods, ranging from simple gross-alpha techniques using scintillators to precise alpha spectrometry that distinguishes individual radionuclides.

Integrating techniques may employ alpha-sensitive thermoluminescent material, or even film. The most straightforward techniques for particulate activity involve placing the filter in front of a detector, as indicated in Fig. 9. Ideally, all of the alphas leaving the filter strike the detector. One difficulty in this technique is assuring that a reliable sample is deposited on the filter; large errors may result in dust-laden air (e.g., in mines). Another problem is self-absorption; if the sample to be counted exceeds a few mg/cm^2 in

Fig. 9. Schematic arrangement for alpha detection using a semiconductor detector and an air filter.

thickness, some alphas will not leave the sample without significant energy degradation. Similarly, air between the filter and detector can degrade the alpha energy by amounts that depend on the path length (see Fig. 9). However, for measurements where the alpha energy is not determined, such degradation is unimportant unless the resulting signal is reduced below some system threshold.

Gross-alpha measurements, in the strict sense, do not require energy measurement. Any of the detection techniques discussed above may therefore be applied in relatively simple form, including gas proportional counters, scintillators, and semiconductor detectors. Such a "Tentative Method for Analysis of Gross Alpha Radioactive Content of the Atmosphere" that depends on filter collection followed by counting with any alpha sensitive detector is given by the American Public Health Association's Intersociety Committee in Ref. 18.

For most situations, a predominant contributor to total alpha activity in air is radon (including "thoron") and daughters. These daughters typically attach themselves to particles, so that they are collected by air filter systems. For measurement of total activity other than radon and its daughters discrimination may be effected on two bases. First, the controlling half-lives of radon daughters are relatively short (less than 30 minutes for daughters of radon-222 and up to 10 hours for daughters of the usually less important "thoron", radon-220), so that delaying counting for a correspondingly longer period after collection reduces the degree to which they interfere. Secondly, the alpha energies from radon daughters are larger than 6 MeV, typically greater than those of other alpha

emitters of interest, such as radium, uranium, or plutonium. As a result detectors providing energy information, such as semiconductor detectors, may be employed in configurations that discriminate against radon daughters. Due to poor energy resolution, radon daughter counts often spill over into the low energy region, a difficulty that may be reduced by subtraction techniques employing two energy "windows." More detailed discussion of such methods is given in Section IV.H on plutonium and other transuranics, which have the lowest maximum permissible concentrations in air.

Although most gross-alpha (or gross-alpha except for radon and daughter) instruments monitor activity collected on a filter, a notable exception is the class of devices depending on a "Lucas chamber", a small spherical or cylindrical cell lined with ZnS scintillator viewed by a photomultiplier tube (Ref. 19). The sample, which may be air containing gaseous and/or particulate activity, is introduced into the volume and counted directly. Since the chamber size is small, nearly all of the emitted alphas strike the scintillator on the wall. This method is discussed fully in Section IV.C, "Radon-222 and its Daughters."

Commercially available instruments that measure alpha activity in air are of the more conventional type described above, i.e., based on a collection filter viewed by a proportional counter, scintillator (especially ZnS), or semiconductor detector. The more sophisticated of these, typically employing semiconductor detectors of the diffused junction or surface barrier type, can discriminate to a large extent against radon daughters.

Alpha-integrating techniques may also employ plastics in which etching shows alpha tracks, as well as thermoluminescent materials. Fleisher, et al. (Ref. 20) discuss nuclear tracks in solids. Track-etch materials and associated procedures have been developed for uranium surveys based on detection of radon and its daughters and for detection of these short-lived radionuclides in air. TLDs have also been used in radon and radon daughter monitoring. See Sections IV.G (Uranium and Thorium), IV.E. (Radon and Its Daughters), and III.G. (Personnel Dosimetry).

Semiconductor detectors (and others) may also be employed in a more

complete spectroscopic system, one distinguishing between individual radionuclides. In systems that view the collection filter directly, the presence of radon daughters still provides interference, although the resultant difficulty can be substantially reduced by placing the filter sample in an evacuated counting chamber, thereby improving the energy resolution of the system. Examples of such techniques are given in the sections on plutonium and on radon and daughters (where the purpose is to distinguish between individual daughters). Methods for monitoring other transuranics in air are also discussed in the plutonium section.

Recently developed methods for monitoring long-lived alpha emitters in air, particularly plutonium and other transuranics, depend on the particulate nature of these activities in a more subtle respect than that they may be collected on an air filter. Because the mass of particles affects their behavior in a moving air stream, partial segregation of the activity of interest may be effected by impaction techniques such as those discussed in Section IV.H on plutonium.

An alternative to viewing the filter sample directly is to separate chemically the radionuclides of interest from the filter matrix and deposit them on a sample to be placed in an alpha spectroscopic system. Chemical separation techniques are discussed below.

d. Alpha Measurements in Water; Gross-Alpha

Measurements of alpha activity in water can range in sophistication from simple measurements of gross alpha activity to measurement of individual radionuclide concentration using chemical separation techniques followed by alpha spectroscopy. Chemical separation techniques are discussed separately below and also in sections on individual radionuclides in Part IV of this volume. The present discussion emphasizes measurements in community water systems, primarily of gross-alpha (and gross-beta). Before turning to gross-alpha techniques, we note that alpha spectrometry of water samples may be accomplished using any of the spectrometric techniques discussed above, provided that a thin sample is prepared, as noted under chemical separa-

tion technique (below) and elsewhere (e.g., Ref. 21).

Measurements of gross-alpha and gross-beta levels in water should be a mainstay of public health monitoring systems. If these levels are shown to be small enough, which is usually the case, it is ordinarily unnecessary to identify and measure individual radionuclides using more sophisticated techniques. Interim regulations for alpha activity in drinking water (cf. Section I.D, Appendix D) from the Environmental Protection Agency (Ref. 22) specify maximum levels for radium-226, radium-228, and gross-alpha activity in community water systems as: 5 pCi/l for combined radium-226 and -228; and 15 pCi/l for gross-alpha, including radium-226 but excluding radon and uranium. (The Public Health Service had recommended a radium-226 limit of 3 pCi/l in Ref. 23.) Thus a measured gross-alpha activity of less than 5 pCi/l would preclude the need for measurement of radium activity. (As discussed in the section on radium, the latter measurements may be based either on chemical precipitation or on radon emanation.)

A standard method for gross-alpha and gross-beta is specified by the American Public Health Association in Ref. 24. This method is similar to that specified by the EPA on a tentative basis (Ref. 25). The technique involves depositing a sample extracted from the water on a counting planchet, which is then inserted into the detector volume or placed next to the detector, as appropriate. The sample preparation is the difficult part. Because of the low range of alphas, only a very thin ($\leq 3 \text{ mg/cm}^2$) sample can be used without the need for a significant self-absorption correction. The water sample is dried by oven evaporation just below the boiling temperature, or on a hot plate, then counted. The main systematic error is from alpha-laden particles which might become airborne during the drying process. The recovery of gross-alpha activity in this procedure is in the 85% to 90% range, with reproducibility (precision) of about $\pm 20\%$, at 95% confidence level. If separate determinations of suspended and dissolved matter are desired, a separation using a Gooch crucible or membrane filter is possible.

For counting gross alpha and gross beta an internal gas proportional counter is recommended: it has high-sensi-

tivity, good geometrical efficiency, and reliability. In the presence of both alpha and beta activity, the alpha-beta separation can be accomplished by measuring the counting rate vs. voltage; a typical counter might respond as shown earlier in Fig. 14. As noted previously, the first plateau is from α 's, the second from the sum of $(\alpha + \beta)$.

Alternative methods might employ either a thin-window proportional counter or a Geiger-Muller tube. Both have a lower counting efficiency, and the G-M tube has the added limitation of not separating alpha from beta activity.

This method may be calibrated using standard alpha emitters, including uranium salts, as well as other radionuclides. A recent study (Ref. 26) has been conducted by the Environmental Protection Agency's Analytical Quality Control Service of the APHA method. Unknown samples were distributed to about 15 participating laboratories; the (gross alpha) activities were in the range of about 25 to 100 pCi/liter. Using the Standard Methods from the 12th Edition of the APHA compilation, analyses were performed on each sample. The gross alpha measurements were found to agree quite well (the average activity found was 0.92 ± 0.10 of the known activity). To quote from Ref. 26:

"The results of the technical experiment show that this procedure for gross alpha . . . determination is acceptable as a standard method. As a screening technique, the procedure performed better than expected."

e. Chemical Separation Techniques

Alpha spectroscopy is capable of directly identifying individual alpha emitters only under a restricted set of circumstances. In particular, the short range of alphas often necessitates the use of chemical separation techniques to concentrate the alpha emitter from the bulk material of the main sample. The chief aim of such a separation process is to reduce the sample mass to the microgram range without quantitative loss of the alpha emitters under investigation. The initial sample may be water, soil, biological tissue, or even an air filter. The final sample should be a thin layer deposited on a suitable substrate. In many cases, this is accomplished by

electroplating a fraction of interest onto a metal planchet. In others (such as one of the standard methods for radium), a precipitate collected on a filter may merely be dried before measurement with an alpha detector. (It is also possible, in principle, that the fraction of interest be incorporated into a liquid scintillator sample for spectroscopy as discussed in b.3.)

When an energy spectrometer can be used, it is not generally necessary to separate the alpha emitters from each other (unless there is overlap in the energy spectrum). However, analytical chemical techniques for quantitative separation of individual radionuclides can make spectrometry unnecessary: a "gross-alpha" measurement can be ascribed to one-and-only-one element after adequate chemical separation.

The fact that many of the alpha emitters are members of natural or artificial radioactive series may also be used to advantage. Following buildup or decay after chemical separations have been made may allow a check on the purity of a particular fraction.

Unfortunately, the number and variety of chemical-separation techniques prevents a comprehensive discussion of them here. There are 15 elements and almost 100 alpha-emitting isotopes in the range from bismuth to the trans-uranic elements, and many possess a wide variety of chemical oxidation states which complicate separation.

Several major problems occur in analytical chemical separations:

- The fractional recovery of the individual alpha-emitter must be known to appropriate accuracy. Recoveries above 95% are desirable, since quantitative errors are less likely when recoveries are high. The use of radioactive tracers is common as a means for checking chemical yields, although differences in recovery may occur between tracers and unknowns.
- The alpha-emitting fraction must be separated from the bulk matrix in a way that allows accurate, convenient counting or spectroscopy at the final stage.
- The presence of multiple oxidation states sometimes requires a complicated series of procedures,

especially in samples of unknown or mixed composition. For example, uranium may be found in both hexa- and quadrivalent states in soil samples, and these have completely different chemical properties.

- Sometimes it is not necessary to perform complete separation of all of the elements individually, since alpha spectroscopy may be used for the final discrimination amongst the unknowns. However, certain interferences can be important even in this last state (e.g., the alpha from ^{237}Np has an energy close to those of several of the uraniums).

Several handbooks and compilations contain detailed chemical procedures used at some of the larger radiation laboratories. Among these are the Department of Energy Environmental Measurement Laboratory's Procedures Manual (Ref. 5); the handbook of the EPA National Environmental Research Center (Ref. 27); an older standard USPHS Radiological Health compilation (Ref. 28); the Methods published by the APHA (Ref. 24); and the handbook of the Nevada Applied Ecology Group (Ref. 29). In addition, Sill has described detailed procedures for separations of most of the significant alpha-emitting elements (Ref. 30, 31).

Most routine chemical separations, such as those described below, employ acid leaching to remove the radionuclides of interest directly from the bulk sample. But Sill has described a total dissolution technique "by which virtually all alpha-emitting nuclides can be determined in soil" (Ref. 31) using a few grams of sample. We quote further (underlining in the text is ours):

"The sample is decomposed completely by a combination of potassium fluoride and pyrosulfate fusions with simultaneous volatilization of hydrogen fluoride and silicon tetrafluoride. The cake is dissolved in dilute hydrochloric acid and all alpha emitters are precipitated with barium sulfate. An option is provided by which uranium can be obtained in a separate fraction to eliminate mutual interference with ^{237}Np . The barium sulfate is dissolved in acidic aluminum nitrate and the elements from thorium through plutonium are extracted in-

to Aliquat-336. Acid-deficient aluminum nitrate is then added to the aqueous phase to reduce the acidity, and americium, curium and californium are extracted into fresh Aliquat-336. Thorium is removed by scrubbing the first organic extract with 10 M hydrochloric acid, and protactinium, uranium, neptunium and plutonium are stripped with a mixture of perchloric and oxalic acids. Americium, curium, and californium are stripped from the second organic extract with 8 M nitric acid. Each fraction is electrodeposited and analyzed by alpha spectrometry with few interferences. Overall recovery excluding electrodeposition ranges from 88% for protactinium to 99% for plutonium."

Here we shall merely summarize the techniques which have come to our attention, element-by-element, with references for more details. The elements treated (atomic number in parentheses) are:

- (1) Polonium (84)
- (2) Protactinium (91)
- (3) Neptunium (93)
- (4) Americium (95)

Chemical separation techniques for radium (88), uranium (92) thorium (90), and plutonium (94) are given in the specific sections on these radionuclides in Part IV. Some alpha-emitting elements are not discussed here, either because they do not appear commonly in environmental samples; or because their half-lives are very short; or because procedures for their individual separation have not come to our attention. Among these are: astatine (85), francium (87), actinium (89), and curium (96) and heavier.

(1) Polonium. Polonium (atomic number 84) occurs in isotopic states with atomic weights from 193 to 218. Almost all of the polonium isotopes are alpha-emitters. In environmental media, polonium isotopes occur most importantly as daughters in the uranium-238 decay chain, following radon-222. The radon-daughter isotopes of polonium are:

^{218}Po (E_{α} = 6.00 MeV, $T_{1/2}$ = 3.0 min, known as Radium A)

^{214}Po (E_{α} = 7.69 MeV, $T_{1/2}$ = 164 sec, known as Radium C')

^{210}Po (E_{α} = 5.30 MeV, $T_{1/2}$ = 138 days, known as Radium F)

As progeny of radon-222, these isotopes may appear in the gaseous phase (or more commonly adsorbed on particulates in the air). Measurements of these species in air are discussed in "Radon-222 and Its Daughters" elsewhere in this volume.

The occurrence of polonium in water, urine, or soils can be important when the parent (radium, radon, or a daughter such as Pb-214) is found in these media.

Chemical separation techniques for polonium in water or urine have been described in both the EML Procedures Manual (Ref. 5) and the PHS compilation (Ref. 28). The procedures are also applicable to those organic materials that can be converted to chloride solutions (Ref. 5). The two procedures are similar, each using a strong HCl solution and ascorbic acid to immobilize any iron that might be present. This is followed by plating (spontaneously induced by stirring) on a silver disk (Ref. 28) or nickel disk (Ref. 5). The plating procedure takes 3 to 5 hours, after which the disk is removed and alpha-counted using an internal proportional counter or other alpha spectrometer. We quote from Ref. 5: "This is a very specific procedure and can be carried out directly in the presence of many other radionuclides." A ^{218}Po tracer may be used to determine chemical yield.

A method for determination of ^{210}Po is described in Ref. 27.

(2) Protactinium. Protactinium (atomic number 91) occurs naturally with mass number 231 as a daughter of ^{235}U . ^{231}Pa has a half-life of 3.2×10^4 years, and its alpha decay spectrum consists principally of three major groups within 50 keV of 5.0 MeV. Because this isotope is a likely constituent of uranium tailings, Sill (Ref. 32) has developed a chemical extraction method that is specific to protactinium. As with the radium-to-californium method described above, the

protactinium method involves total dissolution of the sample using potassium fluoride and pyrosulfate fusions. After dissolving of the resulting fusion cake in hydrochloric acid, protactinium is extracted into diisobutylcarbinol either directly or following precipitation with barium sulfate. Extraction may be followed by electrodeposition and alpha spectrometry or by gross alpha counting.

(3) Neptunium. Neptunium (atomic number 93) is the first transuranic element. Its longest-lived isotope, ^{237}Np , has a half-life of 2.14 million years and a complex alpha-decay scheme with energies near 4.6 MeV. Although several neptunium isotopes are alpha-emitters, none except ^{237}Np is of much consequence for environmental monitoring purposes, because of either their short half-lives or their low production rates in processes of concern for environmental surveillance.

A method for neptunium determination in urine, feces, tissue, and air filters is described in the EML Manual (Ref. 5). Reduced Np is coprecipitated with LaF, from HNO_3 solution. Np is extracted into TTA (thenoyl-trifluoroacetone) from dilute HCl, back extracted into more concentrated acid, and electrodeposited for alpha counting or alpha spectroscopy. The procedure takes 6 to 8 hours. A similar method specifically for urine has been developed by Union Carbide at Paducah (Ref. 33).

(4) Americium. Americium (atomic number 95) is the third transuranic element. Its more commonly found isotopes are ^{241}Am ($E_\alpha = 5.3$ MeV, $T_{1/2} = 440$ years) and ^{243}Am ($E_\alpha = 5.3$ MeV, $T_{1/2} = 7950$ years). None of the other isotopes of americium has an alpha-decay mode of significance. We have already described total dissolution chemical separation techniques for americium (Ref. 30, 31). Another typical chemical technique is found in the EML Manual (Ref. 5), from which we quote:

"The actinide elements are concentrated by coprecipitation with lanthanum fluoride. Plutonium, if present, is removed by extraction with thenoyltrifluoroacetone (TTA) by the Hanford plutonium bioassay procedure. Excess sodium

hydroxide is added to the aqueous phase to redissolve aluminum hydroxide, while coprecipitating americium on lanthanum hydroxide. Trivalent americium is oxidized to the americyl ion by silver catalyzed peroxydisulfate. Addition of ammonium fluoride stabilizes the americyl ion, while precipitating the lanthanum and any alpha emitting impurities as fluorides. Following reduction to the trivalent state with manganous ion, the purified americium is collected by coprecipitation with cerium fluoride. Measurement is by alpha counting for Am-241 and Am-243, the usual americium isotopes encountered.

"The method has been tested for urine, tissue, feces and air filters. The procedure gives excellent separation from neptunium, thorium, uranium, plutonium and curium. It is completely compatible with the plutonium procedure described [elsewhere in (Ref. 5).]"

Also discussed in Ref. 5 are a soil method that employs leaching with nitric and hydrochloric acids and a method for water, air filters, and tissue that uses acid leaching.

Americium-241 is an important isotope since it is the daughter of plutonium-241. The 60-keV gamma ray that accompanies 36% of the ^{241}Am alphas is often used for detection. This is discussed fully in Section IV.H on "Plutonium".

5. Calibration Sources

For calibration purposes, the analyst needs known standard sources. To satisfy this need, the U. S. National Bureau of Standards has for many years provided standard materials. The alpha-standard samples "consist of a practically weightless deposit of the nuclide on a thin platinum foil cemented to a monel disk" (Ref. 34). Samples from NBS, available in the activity range from a few to a few thousand nanocuries, can be purchased for the isotopes ^{210}Po ($E_\alpha = 5.305$ MeV), ^{238}Pu ($E_\alpha = 5.50, 5.46$ MeV), ^{241}Am ($E_\alpha = 5.49, 5.44$ MeV), and ^{148}Gd ($E_\alpha = 3.18$ MeV).

6. Summary and Conclusions

In this section we have attempted to outline the various techniques for the detection, measurement, and energy spectrometry of alpha particles in various media. The most salient physical features of alphas (their high energy, dense specific ionization and short range), make their detection relatively easy. Discrimination against other radiations is also easy. The difficulties with alpha detection are a result of these same physical properties. For example, it is often difficult to detect alpha activity if it is accidentally spilled on the floor, especially in cracks and near inaccessible corners. Also, alpha emitters contained deep within a soil sample ("deep" meaning in this sense below 50 microns or so) require chemical analysis prior to counting, as do alphas in water or urine. For this reason, X or gamma techniques are often used, where appropriate, for monitoring of alpha emitters.

These problems have led to a variety of instrumental and chemical techniques, which we have summarized here. For gross-alpha measurements, the standard methods in air and water are sufficiently sensitive and well-understood to provide a satisfactory screening technique. Also, for emission-rate measurements (e.g., to detect spills), instruments now exist which can probe most surfaces and odd geometries.

The area most in need of further developmental work is identification of individual nuclides. Alpha spectrometers with quite good resolution now exist, and their use makes possible the identification of many individual iso-

topic states whenever good samples can be prepared. Proper sample separation seems to be the crux of the problem.

Present wet-chemical techniques, required to isolate alpha-emitting nuclides from background matrices (soil, air filters, water, tissue, urine), are laborious, liable to systematic error, and expensive. The use of tracers to follow chemical-yield fluctuations through a long chain of analytical steps is crucial. However, tracers are not universally available in convenient form, nor is there any way to guarantee their effectiveness, especially when the alpha-emitters themselves are bound in multi-valent states in a variety of organic compounds (e.g., plutonium or uranium in soil).

In conclusion, alpha detection is relatively well developed; however, except in cases where samples are easily prepared, quantitative measurements of specific activity, especially when isotopic identification is required, suffer important drawbacks because of cost, time, and sample-size considerations.

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Berkeley, CA

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B. INSTRUMENT NOTES: ALPHA

Included in the following pages are Instrument Notes of several different kinds. The reader should be aware of the following facts, in order to find all Instrument Notes appropriate for alpha detection and measurement:

- Semiconductor detectors are useful for both alpha and beta counting and spectrometry. Instrument Notes for silicon surface-barrier detectors and silicon lithium-drifted detectors have been incorporated in this section. For more notes on semiconductors, the interested reader is advised to consult the Gamma Spectrometry section - Instrument Notes.
- Many instruments are advertised as sensitive to alphas, betas, gammas, x-rays, and in some cases neutrons. Often these instruments consist of a variety of probes attachable to a central electronic counting unit. These instruments have all been placed in the section entitled "Combination Instruments," with the appropriate mnemonic (e.g., RAD-ALP, BET, GAM, XRA).
- Liquid-scintillation systems can be used for alpha detection and measurements. However, all these instruments have been placed in the Radionuclide Instrument Notes (Section IV.I) in the Specific Radionuclides division.

Alpha Air Monitor

Eberline Alpha-3

Class	Stationary
Principle of Operation	Silicon-diffused junction-type detector (490 mm ² area), 2.54 cm diameter filter, air pump, single-channel analyzer, meter or strip readout.
Sensitivity and Range	Efficiency: About 50% of 2 π (wide energy window) from 2.54 cm (1") ²³⁹ Pu source Background: 400 cpm (wide energy window), with air flow of 30 liters/minute (1.05 ft ³ /minute). This results in ~8 CPM in the window when the instrument is calibrated for ²³⁹ Pu. Range: 0-50, 500, 5000 counts/minute full scale. Response time: about 30 seconds (fast setting), 180 seconds (slow setting), to 90% of final reading.
Sampling	Continuous (air filter)
Performance	Detection Limit: 4 MPC-hours of ²³⁹ Pu can be detected with an average radon-thoron background Temperature: 20°F to 130°F Threshold Discriminator: adjustable from 0-10 MeV energy equivalent (0 to 1.0 volt) Window Discriminator:
Requirements	Power: 105-125 V ac, 60 Hz at 0.1 amp [220 VAC, 50 Hz optional] Size: 21x31x31 cm (8.25"x12.25"x12.25") Weight: 6.5 kg (14.5 lb)

Features	<ol style="list-style-type: none">1) Air pump must be provided separately. Model RAP-1 available as option. Air intake is screen protected. Air flow indicator goes from 10 to 100 liters/minute (0.35 to 3.5 ft³/minute).2) Single channel analyzer has 10-turn adjustments for high and low settings of window.3) Operation modes are gross, pulse-height-analyzing, or pulse-height-analyzing with background subtraction.4) Subtraction circuit provides automatic subtraction of a percentage of pulses rejected by the window discriminator. Subtracted percentage adjustable from 10% to 60%.5) Alarm adjustable, 0 to 100% of full scale. Red alarm light, squealer and remote alarm provided.6) Calibration controls accessible through front panel.7) 1" diameter (25 mm) filter is standard. Optional filter holder for 47-mm filter is available.8) Electronics are solid state. Charge-sensitive amplifier input.9) Silicon detector is cleanable.
Reference	Manufacturer's Specifications
Cost	Alpha-3 \$2100
Address	Eberline Instrument Corporation P. O. Box 2108 Santa Fe, NM 87501 (505) 471-3232

Scintillation Alpha Counter
Eberline Model SAC-4

Class	Laboratory
Principle of Operation	Two-inch detector ZnS(Ag) powder on a plastic light pipe, high voltage power supply, charge sensitive input amplifier, timer, and 6 decade readout.
Sensitivity	Efficiency: 80% of 2π minimum from a 2.54 cm (1 inch) diameter ^{239}Pu source Background: <0.3 counts per min.
Sampling	Maximum sample diameter 5.2cm (2 1/32"); maximum thickness .95 cm (3/8").
Performance	Plateau (detector): at least 200 volts long with a slope less than 1% of 100 volts. Preset timer from 0.1 minute to 50 minutes Temperature: 0°C to + 60°C (32°F to 140°F) with less than ± 50 volts plateau shift
Requirements	Power: 115/230 $\pm$ 10% VAC, switch selectable, 50-60 Hz, 114 amp, 3-wire Size: 29.2 x 15.2 x 35.6 cm (11 1/2" H x 6" W x 14" D) Weight: 7.6 kg (16 3/4 lbs)
Features	Detector and drawer assembly easily decontaminated. Variable high voltage supply.

INSTRUMENTATION
FOR ENVIRONMENTAL
MONITORING

RAD-ALP
Stationary
Eberline 2
Page 2

References	Manufacturer's Specifications
Cost	\$1700.00
Address	Eberline Instrument Corporation P.O. Box 2108 Santa Fe, NM 87501 (505) 471-3232

Alpha Survey Meters
Eberline Portable Models

PAC-4S

PAC-6

PAC-4G

PAC-1SAGA

PAC-7

Class	Portable, handheld
Sampling	Continuous
Features	Some models have earphone and speaker outputs, regulated power supply, individual calibration controls for each range.
Address	Eberline Instrument Corporation P.O. Box 2108 Santa Fe, NM 87501 (505) 471-3232

Model	PRS-1/AC-3	PRS-2/AC-3	PAC 4G-3	PAC-6	PAC-7
Radiation Measured	Alpha	Alpha	Alpha	Alpha	Alpha
Detector	Alpha scintillation probe	Alpha scintillation detector and GM tube	Gas proportional alpha probe	Air proportional alpha counter	Air proportional alpha counter
Range	Six decade	Six decade	0-5 x 10 ⁵ counts/min	0-500,5000, 50,000, and 500,000 cpm	0-1000, 10,000, or 100,000 cpm
Scale	Digital	Digital	Linear	Linear	Linear
Accuracy	Limited only by counting statistics		±10% F/S	± 5% F/S	± 5% F/S
Temperature	-18°C - + 60°C 0°F - +140°F	-18°C - +60°C 0°F - +140°F	-40°C - + 60°C -40°F - +140°F	-18°C to 60°C 0°F to 140°F	-18°C to 49°C (0°F to 120°F)
Power	Rechargeable Gel-Cell battery	Rechargeable Gel-Cell battery	5 "D" cells 200 hours	2 "D" cells	Single 9v alkaline battery
Weight	Inst. 2.2 kg (5 lbs.)	Inst. 2.3 kg (5 lbs.)	4.25 kg (9.6 pounds)	1.9 kg (4.25 pounds)	0.55 kg (1.2 pounds)
Size	20cm H x 10cm W x 24cm L (8" H x 4" W x 9" L)	20cm H x 10cm W x 24cm L (8" H x 4" x 9" L)	22.9cmx10.2cmx25.4cm (9"x4"x10")	10.2cmWx20.3cmLx19.7cmH (4"x8"x7.75")	5cmWx23cmLx13cmH (2"x9"x5")
Features	Ratemeter-Scaler Built in speaker. User selects calibration constant and readout unit.	(See PRS-1/AC-3)	Adjustable high voltage control	Good gamma rejection	Light weight, one-hand operation
Remarks	PRS-1 may be used with many other detectors	PRS-2 may be used with many other detectors	High voltage selector option	Approximately 50% counting efficiency	Removable meter module
Cost	\$1855.00	\$1805.00	\$1080.00	\$765.00	\$545.00

Portable Alpha Monitor

Harshaw Model 272

Class	Portable
Principle of Operation	Diffused junction silicon detector (300mm ²), pulse-height analyzer, four digit LED display
Sensitivity and Range	Range: 0-9999 counts Timer range: 0.02 to 1000 minutes Alpha sensitivity: 2 MeV to 6 MeV
Sampling	Single sample counting
Performance	Efficiency: typically 25% for ²⁴¹ Am alpha and standard sample holder. Background: <1 count per hour Maximum sample size: 3.2cm x 0.32cm (1 1/4" O.D. x 1/8")
Requirements	Power: Rechargeable NiCd batteries + battery recharger (117 VAC) Size: 22.2cmH x 8.3cmW x 27.3cmD (8 3/4" x 3 1/4" x 10 3/4") Weight: 3.3 kg (7 1/4 lbs)
Features	NIM Bin Version Optional Easily decontaminated housing and sample holder
References	Manufacturer's Specifications
Cost	Refer to factory
Address	Harshaw Chemical Company Crystal and Electronic Products Dept. 6801 Cochran Road, Solon, OH 44139 (216) 248-7400

Diffused Junction Silicon Radiation Detectors

Harshaw Series DJ Silicon Radiation Detectors

Accessory—Model 101
Vacuum Chamber

Class	Radiation Detectors (alpha)
Principle of Operation	Diffused junction (p-i-n) silicon detectors
Resolution	Values range from 25 to 60 keV for ^{241}Am (5.47 MeV) depending upon active area
Active Areas Available	100mm ² , 400mm ² , 450mm ² , 490mm ² , 700mm ²
Depletion Depth	Typically 100 μm at 100V positive bias (typically 200 μm for 700mm ² model)
Features	Detector surface may be cleaned using a cotton swab moistened with trichloroethylene, methanol or denatured alcohol. Detectors are available with microdot or BNC connectors
Available Accessories	Model 101 Vacuum Chamber, NV-30 0-300VdC NIM bias supply, NV-32 High Voltage NIM power supply
References	Manufacturer's Specifications
Cost	Refer to Factory
Address	The Harshaw Chemical Company Crystal and Electronic Products Dept. 6801 Cochran Road Solon, OH 44139 (216) 248-7400

Phoswich Scintillation Detector

Harshaw Chemical Company

Class

Laboratory

Principle of Operation

The PHOSWICH detector is designed for high sensitivity radiation counting where low level, low energy radiation is normally obscured by background radiation. The detector consists of a thin primary scintillator and a thick secondary scintillator. Events that occur in the secondary crystal only, and those events that occur in both crystals simultaneously, are electronically rejected. Only the low level radiation that is totally absorbed by the primary scintillator is counted. Events are sorted out using the Harshaw NC-25 Pulse Shape Analyzer which recognizes events in the primary and secondary crystals by their different decay times.

By using a thin CaF_2 (Eu) primary scintillator and a NaI(Tl) secondary scintillator, the Phoswich can detect alpha and beta particles in the presence of a gamma background.

Performance	Can suppress higher-energy backgrounds by factors of 3 to 10, compared to ordinary NaI(Tl) or CsI(Tl) detector. Especially designed for low-energy x-rays from Americium-241, as used to determine plutonium.
Features	Pulse shape analysis system Model TASC-10 also available.
References	Manufacturer's Specifications
Cost	Phoswich Detector: ~\$4500.00 - Depends on size
Address	Harshaw Chemical Company 6801 Cochran Road Solon, OH 44139 (216) 248-7400

Liquid Monitor
National Nuclear Corporation Model LMS-1

Class	Portable
Principle of Operation	Alpha detecting unit, power supplies, amplifier, counter component, light emitting diode display
Sensitivity and Range	Maximum sensitivity: ~10 dpm/ml (depending on characteristic of solution to be monitored) Range: 0-99999 counts
Sampling	Single sample counting
Performance	Accuracy: typical $\pm 20\%$ with optically clear solution with pH 5 to 9 if containing concentrations over 500 ppm Operating Temperature: 4.4°C to 65.6°C (40°F to 150°F)
Requirements	Power: 110 or 220V Size: 12.7cmDia x30.5cmH (5"D x12"H) Weight: {6.8 kg (15 lbs) detector {6.8 kg (15 lbs) electronics
Features	
References	Manufacturer's Specifications
Cost	\$7,500.00
Address	National Nuclear Corporation 1904 Colony Street Mt. View, CA 94043 (415) 962-9225

Air Particulate Monitors

Class	Laboratory
Principle of Operation	A gas flow proportional counter or scintillation detector, stationary or moving filter, air pump, counting ratemeter, meter or strip readout.
Features	<ol style="list-style-type: none">1) Automatic air flow control to ± 0.2 cfm within a flow rate range of 1-4 cfm.2) Alarm adjustable 0% to 100% full scale, red alarm lamp, alarm bell.3) Continuous duty, non-contacting Roots type pump; belt-driven by split phase induction motor at 800 RPM. Temperature rise less than 30°C above ambient.4) Optional digital count ratemeter.
References	Manufacturer's Specifications
Address	Nuclear Measurements Corporation 2460 North Arlington Avenue Indianapolis, Indiana 46218 (317) 546-2415 TWX 810-341-3137

Specifications:

Y						
Model	AM-2A	AM-3A	AM-2P	AM-3P	AM-22P	AM-33P
Radiation detected	α		α or β or α and β		α and β , separately but simultaneously	
Filter	2 1/4" diameter stationary	2" diameter moving	2 1/4" diameter stationary	2" diameter moving	2 1/4" diameter stationary	2" diameter moving
Range	3 cycle log 50-50,000 cpm					
Sensitivity, * count build-up per hour	374 cpm above 10 cpm background	278 cpm above 10 cpm background with filter speed of 1" per hour	692 cpm, α 702, cpm, β	518 cpm, α 525 cpm, β with filter speed of 1" per hour	692 cpm, α 695 cpm, β	518 cpm, α 525 cpm, β with filter speed of 1" per hour
Background	less than 1 cpm before collection		A 0.04 mR/hr background produces less than 0.1 cpm for α and 30 cpm for β . A 10 mR/hr background produced 30 cpm plus 12 cpm/mR/hr for β .			
Efficiency	20%	20% in SA-30 shield	α : 37% β : 37.5%	α : 37% β : 37% in SA-30 shield	α : 37% β : 37%	α : 37% β : 37% in SA-30 shield
Sampling	Continuous					
Response time	Varies inversely with count rate from ~200 seconds at 50 cpm to 1 second at 50,000 cpm					
High voltage	300 V to 1300 V		500 V to 2000 V		500 V to 2000V	
Power	90 - 130 VAC, 60 cycle, 5-7 amp					
Size	53.3cmD x99.1cmH x99.1cmW 21"D x39"H x39"W		30.5cmD x99.1cmH x76.2cmW 12"D x39"H x30"W		53.3cmD x99.1cmH x99.1cmW 21"D x39"H x39"W	
Weight	158.8 kg (350 lb) 249.5 kg (550 lb)			249.5 kg (550 lb)		
Cost	Special Request		\$7542.00	\$9672.00	\$8531.00	\$10,660.00
Requirements	A cylinder of P-10 gas					

*Assuming a 5 cfm regulated air flow and 10^{-10} μ Ci/cc concentration of activity.

Nuclear Research Model MAP-1000
Continuous Alpha Air Monitor

Class	Mobile
Principle of Operation	Model MAP-1000 is a portable (readily transportable) continuous alpha air monitor which features automatic programmed sampling, delay for radon decay, counting, and printout.
Sensitivity and Range	Inlet air flow is such that background count rate does not register more than 36 counts/min/m ³ with counting circuits set so that 200 alpha disintegrations per minute (normal uranium) per cubic meter of air will register 200 ± 20 on the printer. Temperature Range: -18°C to + 55°C
Sampling	
Performance	The air monitor operates automatically in a time sequence as follows: <ul style="list-style-type: none">a) Sample for 30 min, change or transport paper, and repeat. Delay counting for at least 4.5 hr. after sampling periodb) Count 30 min. and print resultc) Average the 30 min. counts for 8-hr intervald) Print 8-hr average, time and date.
Requirements	Power: 120 Vac, 60 Hz, single phase, 5 amperes max. Size: 66 cm max. width, 115 cm depth, 122 cm height Weight: 350 lbs.

INSTRUMENTATION
FOR ENVIRONMENTAL
MONITORING

RAD-ALP
Mobile
Nuclear Research Corp.,
Page 2

Features Designed to Oak Ridge Gaseous Diffusion Plant requirements
for monitoring uranium and other alpha emitters.

References

Cost Price upon request

Address Nuclear Research Corporation
1105 Industrial Highway
Southampton, PA 18966

Alpha Scintillation Survey Meter

Nuclear Research ASM-30

Class	Portable, handheld
Principle of Operation	Scintillation counter, [ZnS (Ag) Detector], counter, meter readout
Sensitivity and Range	Energy sensitivity: alpha 3 MeV or greater Ranges: 10^3 , 10^4 , 10^5 , 10^6 cpm Gamma Interference: above 5 R/hr
Sampling	Continuous
Performance	Meter accuracy: $\pm 10\%$ Temperature: 40°F to 130°F Response time:
Requirements	Power: two "D" cells Size: 21cmL x 11cmW x 15cmH (8.5"L x 4.3"W x 7.1"H) Weight: 3.2 kg (7 lbs)
Features	Earphone jack, battery check circuit. Designed and engineered to U.S. Navy specifications AIV/PDR-56B. Calibration alpha source
Cost	\$895.00 - w/carrying case and probe - addt'l \$245.00
Address	Nuclear Research Corporation 1105 Industrial Highway Southampton, PA 18966 Tele: (215) 357-5015

Silicon Surface-Barrier Detectors

Ortec Series A, B, C, D, E, R

Series R Detectors

Class	Detectors
Principle of Operation	Silicon surface barrier detectors
Features	<ul style="list-style-type: none">(1) Table shows range of prices from poor resolution detectors to best resolution detectors.(2) Various mounts available: ring mount, rear microdot, rear BNC, transmission mount.(3) Front electrode is $40 \mu\text{g}/\text{cm}^2$ gold, except for series R (ruggedized) which has $50 \mu\text{g}/\text{cm}^2$ of aluminum.(4) Rear electrode is $40 \mu\text{g}/\text{cm}^2$ of aluminum.(5) Resolutions are quoted on all detectors, measured at 22°C and 10^{-6} torr.(6) Many specialized detectors are also available.
References	Manufacturer's Specifications
Address	ORTEC, Inc. 100 Midland Road Oak Ridge, TN 37830 (615) 482-4411

Specifications:

Range of Guaranteed Max. Resolution (keV)		Series	Range of Active Areas (mm)	Range of Depletion Depth (μ m)	Description	Range of Prices
α	β					
15-80	6-75	A	7	100 to 300		\$300.00 - \$ 400.00
			25 to 200	100 to 2000	Partially Depleted	95.00 - 3630.00
			600 to 900	100	Partially Depleted	600.00 - 1050.00
			25 to 100	3000 to 5000	Partially Depleted	2000.00 - 4040.00
			450	100 to 1500		300.00 - 2940.00
16-27	7-21	B	25 to 200	150 to 2000	Totally Depleted	190.00 - 3780.00
			300 to 450	150 to 1000	Totally Depleted	440.00 - 2010.00
17-25	9-20	C	50 to 450	100 to 2000	Annular Partially Depleted	230.00 - 2400.00
		D	10	15	Planar Totally Depleted	1365.00
			25	15 to 100	Planar Totally Depleted	1545.00 - 525.00
			50	15 to 100	Planar Totally Depleted	1690.00 - 675.00
			100 to 300	20 to 100	Planar Totally Depleted	2595.00 - 800.00
12-20	4-14	E	7	100 to 300		385.00 - 575.00
			25 to 450	100 to 500	Premium Partially Depleted	365.00 - 1475.00
17-80	8-75	R	50 to 450	100 to 1000	Ruggedized Partially Depleted	210.00 - 2170.00
			600 to 2000	100 to 500	Ruggedized Partially Depleted	900.00 - 3270.00

Alpha Counting and Spectroscopy System
ORTEC 576

Class	Laboratory
Principle of Operation	Two completely independent alpha spectrometry channels are contained in a double-wide NIM module. Each channel has a vacuum chamber with a sample holder and surface barrier detector, a detector bias source, a preamplifier, a shaping and stretching amplifier, a biased amplifier, a test pulser, and a discriminator.
Sensitivity and Range	Energy ranges: 4 to 6 MeV, 5 to 7 MeV, and 6 to 8 MeV Energy output: 0.1 to 10V
Sampling	Individual samples 2 at a time
Performance	System resolution* (for 5.486 MeV ²⁴¹ Am): ≤30 keV FWHM Efficiency:* >25% at 5.486 MeV, 1 1/2 inch diameter sample 1mm spacing Background:* <20 counts/day above 3 MeV Sample size: Up to 1 1/2 in. diameter planchets or filter paper Sampling spacing: Adjustable from 1 to 15mm from detector Nonlinearity: <0.1% of full scale (0-10V) Stability: System, < 0.1% within 10°C temperature change, pulser 50 ppm/°C
Requirements	Power: NIM-BIN Size: Double wide NIM module Weight:
Features	1) Energy zero adjustments on front panel 2) Factory calibrated 3) Interval pulses for quick system checkout 4) NIM logic output from discriminators can be used for either counting or MCA routine 5) Choice of either vertical or horizontal sample holder
References	Manufacturer's Specifications
Cost	576 Models with two 300 mm ² detectors \$3270.00 with two 450 mm ² detectors \$3590.00 with two 600 mm ² detectors \$3900.00 without detectors \$2760.00
Address	ORTEC incorporated 100 Midland Road Oak Ridge, TN 37830 (615) 482-4411

*Based on 576-450R, basic unit with 450mm² Ruggedized™ low background detectors.

Silicon Surface Barrier Detectors

Princeton Gamma-Tech

Class Detectors

Principle of Operation Silicon Surface Barrier Detectors for Alpha, Beta Spectroscopy

Specifications

Fully Depleted Detectors

Active Area (mm ²)	FWHM Resolution (keV)		Depletion Depth (μm)	Price Range
	α	β		
24	19	10	150 to 2000	
50	19	10	150 to 2000	
100	20	13	150 to 2000	
150	20	13	150 to 2000	
200	20	13	150 to 2000	
300	23	18	150 to 1000	
400	28	21	200 to 1000	

Partially Depleted Detectors

Active Area (mm ²)	FWHM Resolution (keV)		Depletion Depth (μm)	Price Range
	α	β		
25	18	12	100 to 2000	
50	18	12	100 to 2000	
100	19	13	100 to 2000	
150	20	15	100 to 2000	
200	24	19	100 to 2000	
300	21	16	100 to 1000	
400	26	21	100 to 1000	

Annular Detectors

Active Area (mm ²)	FWHM Resolution (keV)		Depletion Depth (μm)	Price Range
	α	β		
50	25	20	100 to 2000	
100	30	25	100 to 2000	
150	30	25	100 to 2000	
200	40	35	100 to 2000	
300	40	35	100 to 1000	
400	45	40	100 to 1000

Enviroguard Partially Depleted Detectors (Series E)

Active Area (mm ²)	Resolution (keV)		Depletion Depth (μm)	Price Range
	α	β		
50	19	15	100 to 500	
150	21	16	100 to 500	
300	24	19	100 to 500	

- Features
1. Energy resolution measured with ²⁴¹Am alpha at 5.477 MeV, and ¹³⁷Cs conversion electron at 624 keV.
 2. Entrance window 40 μg/cm² of gold; back window is ^{Al}<40 μg/cm²
 3. Axial, transmission, ring mounts all available.

References Manufacturer's Specifications

Address Princeton Gamma-Tech
Box 641
Princeton, NJ 08540
(609) 924-7310

Air Particulate
Selective Alpha Monitors

Model 442

Class	Stationary
Principle of Operation	Filter, air pumps (extra), diffused junction Si decontaminable detector, amplifier, single channel analyzer, meter or recorder output.
Features	<ol style="list-style-type: none">1) Threshold adjustable audible alarm2) Requires external air mover accessory3) Fail safe alarm4) Alarm delay5) Short/long time constant6) TTL Pulse output for driving remote readout
References	Manufacturer's Specifications
Address	RadCo/SAI 4060 Sorrento Valley Boulevard San Diego, CA 92121 (714) 452-9150 TWX (910) 337-1774

Model	444	442A
Range	50, 100, 500, 1K, and 5 K counts per minute on meter/recorder	
Energy detected	Baseline Window	0.1 to 10 MeV 0.1 to 1.0 MeV
		0 to 1.0 MeV adjustable 0 to 1.0 MeV adjustable
Area of detector	500mm ²	500mm ² or 700mm ²
Alpha efficiency	10% of 4 π geometry for 5.15 α with 1 MeV window	~ 15% of 4 π with 1 MeV window
Background compensation	Digital 1 adjacent window	Digital 1 separate adjustable window
Air flow	0.5 to 3.0 CFM, from panel indicator	1 to 6 CFM
Operating temperature	0°C to 48.9°C 32°F to 120°F	
Filter	47mm diameter membrane type	
Power	110 V, 60 Hz/ 230 V, 50 Hz	
Size	40.6cmW x15.9cmH x27.9cmD 16"W x6 1/4"H x11"D	
Weight	8.2 kg (18 lb)	
Cost	\$2100.00 w/500mm ² \$2500.00 w/700mm ²	\$2400.00 w/500mm ² \$2800.00 w/700mm ²

Alpha Air Monitor (Moving Filter)

Model SAAM-3

Class	Stationary
Principle of Operation	Air is drawn through section of filter tape. Tape with alpha deposits moves to alpha scintillation detector with predetermined time elapsing before count is taken (allows for decay of radon daughters).
Sensitivity and Range	Integrated digital readout within preset time--or digital ratemeter readout and printed record Sensitivity--1% of most restrictive alpha contamination standards
Sampling	Air pumped through filter section for 30 minutes Traversal of filter section to detector takes 4-5 hrs.
Performance	Scaler accuracy for counting: ± 1 count Timer accuracy: $\pm 1\%$
Requirements	Power: 115 VAC, 50-60 Hz Size: 22"W x 36"L x 54"H (on wagon with 6" casters) Weight: 450 lbs.
Features	Automatic subtraction of radon daughter. May be left unattended for periods up to 30 days Digital printer furnishes complete record
References	Manufacturer's Specifications
Cost	\$5950.00
Address	Technical associates 7051 Eton Avenue Canoga Park, CA 91303 (213) 883-7043

Diffused Junction Silicon Radiation Detector

Harwell Type SRD 20

Class	Detector
Principle of Operation	Consists of silicon diode mounted within a stainless steel body and terminating at two wire pairs, which are electrically isolated from the body. Diode mounted flush to form front surface
Sensitivity	Sensitive Area: 700 mm ² , thickness: 0.5 mm (nominal) Reverse current @15 volts: < 2 μ A @20°C Depletion depth @150 volts: 60 microns (nominal)
Sampling	Not applicable without electronic accessories
Performance	Operating bias voltage: 15 volts Total capacitance @15 volts: 1300 pF (typical) 1700 pF (max.) Background: 2 counts/hour at 1.7 MeV, 20°C (typical)
Requirements	Power: Power supply separate Size: Detector dimension: diameter overall 3.3cm; thickness: 7.2mm
Features	The SRD 20 is the largest detector in a series of rugged α detectors from 10mm ² to 700mm ² sensitive area
References	Manufacturer's Specifications
Cost	On application
Address	AERE, Harwell Oxfordshire OX11 0RA England Tel: (0235) 24141 Attn: R. H. W. Sherwood, Bldg 347.3 Electronics and Applied Physics Division Ext. 4766 R. G. Powell, Bldg. 329 Marketing and Sales Department Ext. 2942

Alpha Spectrometer
Harwell Model 36190

Class	Laboratory
Principle of Operation	Uses a silicon radiation detector to measure the α activity of samples such as filter paper used in the Harwell Personal Air Sampler Type 0434 (separate). Includes single channel analyzer.
Sensitivity and Range	4 Decade LED display, with 1, 10, 100 prescale option. For α of 2.5 MeV to 5 MeV, efficiency is 30% for ^{239}Pu . SCA threshold and channel width can be set in range 0.1 - 8 MeV.
Sampling	Discrete timing range: 1 - 9900 seconds - 1" diameter tray for sample holder
Performance	Background: typically, 0.2 c/hr Detector: Harwell type SRD 22
Requirements	Power: 240V AC main lines Size: 8.9cmWx17.8cmHx33.0cm (3.5"x7"x13") Weight: 4 kg (8 1/2 lbs),
Features	The SCA can be operated in the time limit, count limit or digital ratemeter mode. Display can be switched to indicate count time (seconds) or count rate (count per seconds)

References	Manufacturer's literature
Cost	On application
Address	AERE, Harwell Oxfordshire, OX110RA England Tele: (0235) 24131 Attention: F. D. Seymour, Bldg. 347.2 Instrumentation and Applied Physics Division Ext: 4229 R. G. Powell, Bldg. 329 Marketing and Sales Department Ext. 2942

III. C. Beta Particle Instrumentation

1. Introduction
2. Physical Considerations
3. Measurement Considerations
4. Measurement Techniques
 - a. Detectors which Measure Particle Flux or Dose Equivalent
 - b. Beta Spectrometers
 - (1) Gas Proportional Counters
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 - (4) Solid State Detectors
 - c. Gross-Beta Measurements in Water
 - d. Gross-Beta Measurements in Air
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 - (1) Carbon-14
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5. Summary and Conclusions
6. Acknowledgment
7. References

1. Introduction

The necessity for counting beta particles results from the fact that β -emitters can be a health hazard and that a number of these radionuclides emit no gamma or x rays, or emit them at such a low probability per decay that assay by photon detection is impossible. This section covers several different classes of instruments and techniques for beta-particle detection and isotope identification. These include:

- Gross-beta instruments (such as handheld meters) for use in radiological protection and radiation surveys; these instruments may measure absorbed dose (rad) or particle flux (particles $\text{cm}^{-2} \text{sec}^{-1}$).
- Beta spectrometers for identifying individual beta spectra; these are used for isotope analysis in a sample which may contain several beta emitters.
- Gross-beta techniques in air and water, used as screening measurements for public health purposes; these measurements usually determine specific activity (pCi/m^3 , pCi/g , or pCi/liter).
- Chemical separation techniques to prepare samples for beta detection or spectroscopy.

Separate sections elsewhere in this volume have been devoted to some of the more important beta-emitting nuclides found in the environment (see Division IV, "Instrumentation for Specific Radionuclides"). These radionuclides are:

- ^3H , tritium
- ^{85}Kr , krypton-85
- ^{90}Sr , strontium-90
- $^{129,131}\text{I}$, iodine-129, 131.

However, for some of the other important beta-emitters, chemical separation techniques or specific instruments have been developed to the point that some special description is merited. Therefore, separate brief discussions in this section have been devoted to them. These are the following beta-emitters:

- ^{14}C , carbon-14
- $^{103,106}\text{Ru}$, ruthenium-103 and 105

- $^{133,135}\text{Xe}$, xenon-133 and 135
- $^{141,143,144}\text{Ce}$, cerium-141, 143 and 144.

2. Physical Considerations

The term beta-particle was originally used to refer to those particulate emissions of radioactive nuclear decay which had small mass (to distinguish them from alphas) but were charged (to distinguish them from gamma). Although the naturally-occurring beta emitters are sources of negative betas (β^-), artificially-produced isotopes include a large number of positron (β^+) emitters as well.

A beta particle is an electron. This fact was not known when beta-radiation was first discovered by Rutherford in the earliest days of nuclear physics, but as time went on, the properties (mass, charge, interactions) of the two types of particles turned out to be identical. Nevertheless, some doubt as to the exact identity of the two particles existed for a long time. What finally settled the issue were two discoveries: first, the fact that positive β^+ particles annihilated with atomic electrons; and second, the fact that β^- , when captured into atomic orbits, obeys the Pauli exclusion principle just like electrons.

The term β decay embraces all modes of nuclear decay in which the atomic number (Z , i.e., the nuclear charge) changes by one unit while the mass number A remains constant. Beta decay can be thought of as a mechanism that changes the relative numbers of neutrons and protons in a nucleus, bringing the neutron/proton ratio closer to the "region of stability" on the chart of the nuclides. In the beta decay process, the nucleus may: emit a positive electron (changing Z to $Z-1$), emit a negative electron (Z to $Z+1$); or capture an orbital electron (Z to $Z-1$). For practical purposes, negative beta emission can be thought of as the "conversion" of a neutron to a proton within the nucleus. Indeed, the simplest case of nuclear beta decay is the decay of a free neutron into a proton, a negative beta, and an anti-neutrino. Proton-to-neutron conversion produces positive beta particles.

We shall not go into detail here about all of the many features which characterize beta-decay. For such a discussion, the reader is referred to Ref. 1. However, one key feature, essential to design of beta detection instruments, must be discussed. That feature is the experimental fact that beta particles emerge from nuclear decay with a spectrum of energies characteristic of each nuclide. The spectrum has a characteristic bell shape and the average energy is approximately one third of the maximum beta energy. It is now known that all beta-emission is accompanied by the emission of a neutrino (or anti-neutrino), whose symbol is $\bar{\nu}$ (or ν). The neutrino is a massless particle with only very weak interactions as it traverses matter. The two particles (β^- and $\bar{\nu}$, or β^+ and ν) together with the recoiling nucleus share the available kinetic energy. It is this sharing that produces the wide beta energy spectrum. Thus β^- and β^+ emission can be thought of as the following "virtual" processes within a nucleus:

Beta emission is commonly tabulated and referred to by its maximum possible beta energy, $E_{\max}$. The spectrum, however, ranges all the way from $E_{\max}$ to zero energy. The energy distribution reveals details about the dynamic processes (for example, nuclear spin-flip or non-flip) which occur within the nucleus during the decay. In Fig. 1 a large number of beta spectra are shown. These spectra (from Ref. 2) demonstrate the wide variety in the types of spectra which characterize beta-decay.

The maximum energy in beta decay can be very small ($E_{\max} = 18.5$ keV for tritium) or very large ($E_{\max} = 3.53$ MeV for one of the betas from rhodium-106). However, most $E_{\max}$ values are in the energy region between 0.3 and 1.0 MeV. Because electrons of this energy stopping in material do not have the simple straight trajectory typical of heavier particles such as alphas, it is not nearly as meaningful to speak of their "range". Large deviations in the electron path are possible because its mass is equal to that of the orbital electrons with which it is interacting, and a large fraction of its energy can be lost in a single encounter. In addition, electron-nuclear interactions which can abruptly change the electron direction sometimes oc-

cur. However, range-energy curves for betas are in wide use; they usually deal with what is called the maximum range. Normally, the electron range is taken from a plot such as that given in Fig. 2 by extrapolation of the linear portion of the curve. The same curve for beta particles emitted by a radioactive source, because of the continuous distribution in their energy, differs significantly. An example is given in Fig. 3. The initial slope is much greater because the "soft" or low-energy beta particles are readily absorbed even in thin absorbers. As a practical matter, if the detector is sensitive to photons, the plot becomes more complicated for the β -emitters that also emit photons; (see Fig. 11.10 in Ref. 3).

Cember (Ref. 4) notes the useful rule-of-thumb that the absorber thickness which stops one half of the betas is found to be about one-eighth of this "range". Figure 4 (from Cember) shows the range of betas in various materials in centimeters. For all materials, especially with low atomic numbers, the range expressed in mg/cm^2 (see Fig. 5) is nearly independent of the absorber material.

3. Measurement Considerations

The main detection mechanism for betas is ionization energy loss, that is, the interaction with the atomic electrons. Although ionization and nuclear scattering/deflection occur simultaneously for betas traversing through matter, the phenomena are sufficiently different that they can be considered separately.

Like all charged particles, betas lose more energy per unit material traversed as they come to the end of their range and stop. However, the effect does not become very significant until the kinetic energy is below about 100 keV. This can be seen in Fig. 6 (from Ref. 4), which shows the specific ionization (number of ion pairs per cm) in air as a function of beta energy. (The average energy lost per ion pair produced in air is about 34 eV.)

Another feature of electron interactions is bremsstrahlung, which is the emission of photons by the electrons as they decelerate in a medium. The energy distribution of the bremsstrahlung photons emitted by an electron of ener-

Fig. 1. The beta ray spectrum and average beta energy of selected isotopes.
(Reproduced from Mantel, Ref. 2, with permission of the International Journal of Applied Radiation and Isotopes, © 1972).

Fig. 1. (continued)

Fig. 1. (continued)

Fig. 1. (continued)

Fig. 1. (continued)

Fig. 2. Usual method for the determination of electron range.

Fig. 3. Typical range curve for electrons (beta particles) resulting from radioactive decay.

Fig. 4. Range-energy curves for beta-rays in various substances. Note that both scales are logarithmic. (Reproduced from Cember (Ref. 4) with permission of the Pergamon Press, © 1969).

Fig. 5. Range-energy curves for beta particles. The range is expressed in units of density thickness. Note that both scales are logarithmic. (Reproduced from Cember (Ref. 4) with permission of the Pergamon Press, © 1969).

Fig. 6. Relationship between beta particle energy and specific ionization of air. Note that both scales are logarithmic. (Reproduced from Cember, Ref. 4, with permission of the Pergamon Press, © 1969).

gy E is a $1/E$ spectrum, so that most bremsstrahlung from beta particles is soft. However, even behind an absorber thicker than the "maximum range" of the betas, penetrating bremsstrahlung photons will sometimes appear and be detected. This is another reason why the "range" of betas is a concept without exact meaning. Bremsstrahlung also occurs for heavier charged particles, of course, but the probability is proportional to $1/M^2$ (where M = mass of particle); thus it is less probable for alphas than for betas by a factor of $(m_\beta)^2/(m_\alpha)^2 = 1.9 \times 10^{-8}$, a very small number indeed.

The following is quoted from Cember (Ref. 4):

"For purposes of estimating the bremsstrahlung hazard, the following approximate relationship may be used:

$$f = 3.5 \times 10^{-4} ZE,$$

where

f = the fraction of the incident beta energy converted into photons,

Z = atomic number of the absorber,

E = maximum energy of the beta particle, in MeV.

"Because the likelihood of bremsstrahlung production increases with atomic number of the absorber, beta-ray shields are made with material of the minimum practicable atomic number. In practice, beta-ray shields of higher atomic number than 13 (i.e., aluminum) are seldom if ever used."

Another physical feature of β -decay that must be borne in mind when making measurements is backscattering. The fact that electrons often undergo large-angle deflections along their track leads to the possibility that an electron entering one surface of an absorber may undergo sufficient deflection that it may re-emerge from the same surface it entered. Such electrons do not deposit their full energy in the absorbing medium, and can thus have a significant effect on the response of detectors designed to measure the energy of externally incident electrons. Electrons that backscatter in the detector entrance window, or dead layer, will totally escape detection. Backscattering is most pronounced for absorbers with high atomic number and electrons with low incident energy. Figure 7 shows the fraction of monoenergetic electrons that are backscattered when normally incident on the surface of various absorbers. Additional data for materials commonly used as electron detectors are given in Table 1.

Fig. 7. Fraction of normally incident electrons which are backscattered from thick slabs of various materials, as a function of incident energy T . (Reproduced from Tabata, Ito, and Okabe, Ref. 5, with permission of Nuclear Instruments and Methods, © 1971).

Table 1. Fraction of normally incident electrons which are backscattered from various detector surfaces

Scintillator	Electron energy (MeV)				
	0.25	0.50	0.75	1.0	1.25
Plastic	0.08±0.02	0.053±0.010	0.040±0.007	0.032±0.003	0.030±0.005
Anthracene	0.09±0.02	0.051±0.010	0.038±0.004	0.029±0.003	0.026±0.004
NaI(Tl)	0.450±0.045	0.410±0.010	0.391±0.014	0.375±0.008	0.364±0.007
CsI(Tl)	0.49±0.06	0.455±0.023	0.430±0.013	0.419±0.018	0.404±0.016

(From Ref. 6).

Backscattering can also influence the apparent yield from radioisotope sources of beta particles or conversion electrons (Ref. 7). If the source is mounted on a thick backing, electrons which are initially emitted into this backing may backscatter and re-emerge from the surface of the source.

In all of the above discussion, the behavior of β^+ particles (positrons) is essentially identical to that of β^- . Positrons exhibit nuclear scattering, ionization energy loss, backscattering, and bremsstrahlung. In these features, only minor deviations

from the behavior of β^- occur. The one feature in which β^+ differs significantly from its negative counterparts is, of course, annihilation: they essentially always annihilate at rest, and sometimes in flight, by interactions with the atomic electrons they encounter. The annihilation converts the rest energy of the (β^+/e^-) pair into photon energy shared by the resulting 2 and sometimes 3 gamma rays. This total rest energy is 1.022 MeV, twice the 511-keV mass-energy of each particle. This gamma energy can be detected, of course, and can provide a unique signature for a stopping β^+ .

Some beta emitters are "pure", that is there are no other accompanying radiations. Examples of pure beta emitters are tritium, carbon-14, phosphorus-32, sulphur-35, calcium-45, and strontium-90. More commonly, beta-emission is preceded, accompanied, or followed by emission of photons (gamma rays, x rays). Cesium-137, krypton-85, cobalt-60, yttrium-90, and promethium-147 are examples.

Figure 8 shows the principal decay schemes of the isotopes tritium, cesium-137, and cobalt-60. Tritium exemplifies the case of a pure beta-emitter. Cesium-137 has two decay modes into barium-137: for 6.5% of the decays, a β^- ($E_{\max} = 1.176$ MeV) leads directly to the ^{137}Ba ground state; for the remaining 93.5%, a β^- ($E_{\max} = 514$ keV) leads to an isomeric state with half-life 2.55 minutes, which goes to the ground state by emission of a 661-keV gamma. Cobalt-60 is the well-known "gamma emitter", because of the two prominent gammas (1.173 and 1.332 MeV) in the decay chain. However, it can be seen that a β^- always accompanies the decay: for 99.88%, a β^- ($E_{\max} = 319$ keV) precedes the sequential two-gamma emission; for 0.12%, a β^- ($E_{\max} = 1.48$ MeV) leads directly to the intermediate state, followed by emission of the single 1.332-MeV gamma.

4. Measurement Techniques

For isotope identification, the total ionization deposited in stopping a beta is not as useful a signature as it is for gamma or alpha detection, because of the previously mentioned distributed spectrum. This means that absolute isotope identification must rely upon either precise spectral measurements, or upon chemical separation techniques. Since these pose special problems, each has been the subject of much intense research and development effort. We shall discuss them separately below.

In making measurements of either the β radioactivity absorbed dose (rad) or its concentration (pCi/volume or mass), it is useful to bear in mind the relevant radiation protection guides. The dose equivalent limits are discussed in "Radiation Protection Philosophy, Criteria, Guides, and Standards" (Section I.D.). Simply, they are 5

Fig. 8. The dominant transitions in the decays of tritium, cobalt-60, and cesium-137. (Reproduced from Lederer and Shirley, Ref. 8, with permission of Wiley Interscience, © 1978).

rem/year for occupational exposure and 0.5 rem/year for a public individual exposure. Table 2 lists the MPC's for several radionuclides that are of environmental importance.

Table 2. Maximum permissible concentration in air and water
for selected β -emitters (Ref. 9, 10 CFR 20, Appendix B).

Radionuclide		Occupational MPC		Unrestricted MPC	
		Air ^{a,b}	Water ^{a,b}	Air ^{a,b}	Water ^{a,b}
^3H	S	$5 \times 10^{-6} \mu\text{Ci/ml}$	$1 \times 10^{-1} \mu\text{Ci/ml}$	$2 \times 10^{-7} \mu\text{Ci/ml}$	$3 \times 10^{-3} \mu\text{Ci/ml}$
	I	$5 \times 10^{-6} \mu\text{Ci/ml}$	$1 \times 10^{-1} \mu\text{Ci/ml}$	$2 \times 10^{-7} \mu\text{Ci/ml}$	$3 \times 10^{-3} \mu\text{Ci/ml}$
	Sub	2×10^{-3}		4×10^{-5}	
^{14}C	S	4×10^{-6}	2×10^{-2}	1×10^{-7}	8×10^{-4}
	(CO_2) Sub	5×10^{-5}		1×10^{-6}	
^{32}P	S	7×10^{-8}	5×10^{-4}	2×10^{-9}	2×10^{-5}
	I	8×10^{-8}	7×10^{-4}	3×10^{-9}	2×10^{-5}
^{35}S	S	3×10^{-7}	2×10^{-3}	9×10^{-9}	6×10^{-5}
	I	3×10^{-7}	8×10^{-3}	9×10^{-9}	3×10^{-4}
^{85}Kr	Sub	1×10^{-5}		3×10^{-7}	
^{99}Tc	S	2×10^{-6}	8×10^{-2}	7×10^{-8}	3×10^{-4}
	I	6×10^{-8}	1×10^{-2}	2×10^{-9}	2×10^{-4}
^{103}Ru	S	5×10^{-7}	2×10^{-3}	2×10^{-8}	8×10^{-5}
	I	8×10^{-8}	2×10^{-3}	3×10^{-9}	8×10^{-5}
^{106}Ru	S	8×10^{-8}	4×10^{-4}	3×10^{-9}	1×10^{-5}
	I	6×10^{-9}	3×10^{-3}	2×10^{-10}	1×10^{-5}
^{129}I	S	2×10^{-9}	1×10^{-5}	2×10^{-11}	6×10^{-8}
	I	7×10^{-8}	6×10^{-3}	2×10^{-9}	2×10^{-4}
^{133}Xe	Sub	1×10^{-5}		3×10^{-7}	
^{135}Xe	Sub	4×10^{-6}		1×10^{-7}	
^{135}Cs	S	5×10^{-7}	3×10^{-3}	2×10^{-8}	1×10^{-4}
	I	9×10^{-8}	7×10^{-3}	3×10^{-9}	2×10^{-4}
^{141}Ce	S	4×10^{-7}	3×10^{-3}	2×10^{-8}	9×10^{-5}
	I	2×10^{-7}	3×10^{-3}	5×10^{-9}	9×10^{-5}
^{144}Ce	S	1×10^{-8}	3×10^{-4}	3×10^{-10}	1×10^{-5}
	I	6×10^{-9}	3×10^{-4}	2×10^{-10}	1×10^{-5}
^{147}Pm	S	6×10^{-8}	6×10^{-3}	2×10^{-9}	2×10^{-4}
	I	1×10^{-7}	6×10^{-3}	3×10^{-9}	2×10^{-4}
^{151}Sm	S	6×10^{-8}	1×10^{-2}	2×10^{-9}	4×10^{-4}
	I	1×10^{-7}	1×10^{-2}	5×10^{-9}	4×10^{-4}

Table 2 continued

Radionuclide		Occupational MPC		Unrestricted MPC	
		Air ^{a,b}	Water ^{a,b}	Air ^{a,b}	Water ^{a,b}
241Pu	S	9x10 ⁻¹¹	7x10 ⁻³	3x10 ⁻¹²	2x10 ⁻⁴
	I	4x10 ⁻⁸	4x10 ⁻²	1x10 ⁻⁹	1x10 ⁻³

^aTo convert to pCi/cm³ divide numerical value by 10⁻⁶

^bTo convert to µCi/m³ divide numerical value by 10⁻⁶

Detection and measurement are quite different concepts, of course. For many purposes, mere detection is sufficient as is the case when a beta-active material is spilled and needs to be located and cleaned up.

Although the discussion of techniques that follows is divided into several parts, many of these approaches are related to one another. For example, detector types that are discussed as spectrometers may be used to measure the particle flux from a spill or measure beta activity of a chemically separated fraction.

a. Detectors which Measure Particle Flux or Dose Equivalent

There are two types of measurements in this category:

- A measurement of particle flux (particles cm⁻² sec⁻¹) may suffice (e.g., spills).
- A measurement of dose equivalent may be required. This can occur when unavoidable exposure must be quantified for radiation protection purposes. Since the quality factor for betas is unity, dose equivalent (rem) is numerically equal to absorbed dose (rad).

In either of these cases, it is always preferable to ascertain the specific radionuclide(s) responsible for the activity. However, in neither case is it always possible.

For detection of betas on or near a surface, an important requirement for a beta detector is a thin-window or thin wrapping. Typical windows in beta counters have thicknesses of a few mg/cm³. The window thickness of a survey instrument is often determined by the desire to count alpha particles as well; in this case windows of a fraction of a mg/cm³ must be used. Alternatively, it might be desirable to screen out alphas and detect only the betas, in which case thicker windows are called for. Because of the continuous spectrum of betas, there will always be some particles on the low-energy end which are not detected. For survey instruments, especially where the radionuclide responsible might be unknown, the thin window is desirable to minimize this low-energy absorption loss.

Electrons with an energy of about 2 MeV or less are readily absorbed by the surface layers of the body. Since the outer part of the skin is an inert layer it is not affected by radiation; therefore, measurements are required only at depths in the body equal to (or greater than) the thickness of this inert layer. By convention this thickness is taken to be 7 mg/cm² (Ref. 10). The quantity measured at depths greater than this is the absorbed dose or absorbed dose rate. Naturally to measure dose beyond this depth the entrance window must be 7 mg/cm². In some cases only the (underlying) surface dose is required and the detector must not be very thick, since absorption of the electrons within it causes the instrument to measure the average dose (rate) throughout the detector. For this the detector thickness should not exceed 10 mg/cm² (about 10 cm of

air) (Ref. 10). The extension of the detector in the direction perpendicular to the field is limited to prevent averaging the dose over too great a cross section of the field. The base on which the window and detector are mounted is usually thick enough to simulate effectively a human body with respect to absorption and backscattering of the electrons. Hence, these instruments detect only the electrons that arrive from sources in front of the detector window. Electrons arriving nearly tangentially to the entrance window must penetrate a long, slanted path both through the window and the detector and many will be largely or completely absorbed. For these reasons these instruments show a dependence of the instrument response on the direction of the β^- particle radiation as shown in Fig. 9.

Once a beta particle penetrates the insensitive window or wrapping, its detection is possible using any of a number of detector types (ionization chambers, scintillation detectors, solid state detectors, Geiger-Muller tubes, gas proportional counters); many of these are also treated below as spectrometers. For a discussion of these

detection methods the reader is referred to "Basic Radiation Detection Methods" (Section I.E.); also see Ref. 7.

For survey work, the most common instrument is an ionization chamber with a thin window, through which betas penetrate. In a gas-filled detector the chamber is usually filled with gas at slightly more than 1 atmosphere, and the amount of gas is determined by the criterion that a beta has a high probability of producing at least one ion pair. This dictates a path-length of 5 to 10 mm in the gas (Ref. 11) to assure ion-pair probabilities in excess of 99%. These instruments do not normally stop any but the lowest-energy beta particles, and are completely unable to provide spectral information.

Windows on gas-filled detectors today commonly take the form of a plastic film, such as Mylar, with a vacuum-evaporated conducting surface on the inside. For survey use, GM tubes usually employ one of two special mixtures as the counting gas: 98.7% helium + 1.3% butane, or (99.05% helium + 0.95% isobutane. Proportional counters employ either pure methane or a mixture known as P-10 gas (90% argon + 10% methane) (Ref. 11).

When a GM counter is used, it should be pointed out that since it gives an indication of the number of particles rather than their energy, its reading is only approximately proportional to dose. Its chief use should be to detect and establish the relative magnitude of electron fields.

Very thin layers of scintillators, behind a thin entrance window, coupled to photomultipliers whose output current is measured constitute good electron dose meters. Due to the thickness, the total light output and hence the current is small; however, they can measure dose rates above a few mrad per hour. Since a covering opaque to light is necessary to exclude visible light from the scintillator, it may be sufficiently thick to exclude low energy electrons also.

Ionization chambers made especially for electron measurements have a very small ratio of chamber thickness to diameter of the entrance window, for the reasons given earlier. However, measurements of electrons or β^- particles are more often made with chambers designed primarily for measuring x or gamma rays. For small electron sources

Fig. 9. Directional dependence of response of gamma-ray, beta-ray, and neutron instruments. (Reproduced from Ref. 10 with permission of the National Council on Radiation Protection and Measurement, © 1978).

on the axis of the entrance window, the error is minimal, but for extended sources, such as one might find in a contaminated area, the error may be large. Although the walls of an ion chamber should be made of tissue equivalent material, the exact composition is not critical.

For beta absorbed dose measurements, the thin-window ionization chamber is the most frequently used instrument. It is also sensitive to gammas, and "beta/gamma" ion chambers are widely available commercially. These instruments are usually calibrated using gamma sources (e.g., cobalt-60), and have read-outs in absorbed dose (rad) or less commonly in exposure (roentgen); the roentgen is not a sensible unit for expressing beta measurements, indeed it is only defined for x and γ rays.

When a tissue-equivalent ion chamber is used, the measurement is fairly well representative of absorbed dose, provided the wall thickness allows the betas to penetrate. The problem with thin windows is that they may not be thick enough to achieve particle equilibrium by incident gamma rays in the MeV energy range. In situations where mixed-energy, mixed-species beta/gamma activity is present, the use of a series of window thicknesses is advised in order to estimate the true absorbed dose more accurately. This discussion points up a most important fundamental aspect of radiation protection measurements: more important than accurate measurements of some components of a mixed field is the assurance that all components are at least qualitatively identified. This requires attention to possible sources and may require use of more than one instrument. The reader is referred to "Particle Equilibrium", Section I.F.2 in this volume for details.

Absorbed dose should be proportional to the total energy deposited in the detector sensitive volume. As such, the detector properly integrates over the beta energy distribution. Particle flux, on the other hand, merely measures all particles energetic enough to pass through the detector insensitive layer and then produce enough ionization to make a "count". Even if the particular beta-emitting nuclide is known, it is still not easy to calculate absorbed dose from flux, since both the energy spectrum and various geometrical effects (thickness of the source, solid angles) enter into the

calculation. If there is ignorance as to the isotope responsible, it is not possible to determine dose.

Calibrations of ratemeter instruments involve using a known radioactive source in a known geometrical configuration. This last is the most difficult problem in calibrations. Concern of a general nature is the overloading of ratemeters at high instantaneous rates. A well-designed instrument must have a way of letting the operator know that it is being overloaded.

b. Beta Spectrometers

In environmental measurements, beta spectrometry may be used for differentiation between different beta-emitters in a mixed or unknown sample. Precision beta spectrometry has also been one of the most valuable research tools in understanding the detailed features of beta-decay nuclear dynamics; the shape of the spectrum is sensitive to a number of parameters in the quantum-mechanical matrix elements which govern the decay.

The key parameter in any spectrometer system is, of course, energy resolution. Resolution is nearly always balanced against sensitivity (depending on such parameters as solid-angle acceptance, counting rate, source thickness) in the sense that attainment of the very best resolution is usually accompanied by a sacrifice in one or another of these parameters and hence in sensitivity. However, for separation of beta-emitters from each other by spectral analysis, because of the broad energy spectra of the betas, sensitivity is not usually sacrificed for high resolution. Some compromise is usually achieved depending on the type of isotope-differentiation desired. However, if detection of mono-energetic conversion electrons is required, then resolution becomes a crucial parameter.

The very best energy resolution is achieved by spectrometers that use momentum selection in a magnetic field. We will not discuss these here, since they are seldom used in environmental monitoring measurements. Their role in research, however, is indispensable. The reader is referred to Refs. 12, 13, and 14 for discussions, comparisons, and critiques of these spectrometers.

For measurements of the type that concern us, various ionization-de-

detecting spectrometers are common. We shall discuss the following types separately:

- (1) gas proportional counters
- (2) solid scintillation counters
- (3) liquid scintillation counters
- (4) solid state detectors.

Among the ionization-detecting spectrometric techniques that are not discussed here are cloud chambers, photographic emulsion detectors, Cerenkov counters, and differential range/absorption measurements. None of these is used commonly in applications that concern us.

The use of ionization chambers for beta spectroscopy is restricted by the small pulse-height. In the pulse ion-chamber mode, a fundamental limitation is lack of sufficient pulse size. An electron with low enough energy to be completely absorbed in any reasonably-sized chamber does not produce enough primary ionization to yield a pulse above the noise of the preamplifier. With the use of pressurized ion chambers, higher energy betas can be fully stopped so this is less of a restriction. However, the thick walls required to withstand the pressure make external counting impossible and internal sources must be used. A discussion of pulse ion chambers can be found in "Alpha-Particle Instrumentation", Section III.B of this volume. The gas proportional counters (discussed next) are the preferred gas-filled instruments for beta spectrometry.

(1) Gas Proportional Counters. These counters were not widely used for beta counting in the early days of nuclear physics, chiefly because of the difficulty in achieving the high gains and good stabilities required. However with the development of better electronic systems and improved gas-purification, beta counting with proportional counters has become perhaps the most common technique for routine measurements. Today, gains of 10^4 to 10^5 are typical, and even 10^6 is used in some counters for beta spectrometry at low energies.

The gas-filled proportional counter is distinguished from the ionization chamber by the following feature: the primary ionization electrons are accelerated by the high fields to a point where they produce further ionization by collisions in their passage through

the gas to the collector. For beta counting, the main requirement is high gain in order to overcome the low initial ionization of the primary betas. A discussion of the way pulse height varies with applied voltage can be found in many sources; for example, the book by Curran and Craggs (Ref. 15). The operating characteristics of these counters has been well described by Curran and Wilson (Ref. 16).

The choice of gas used for filling is particularly critical for beta counting (in contrast to alpha counting, where gains of ~ 100 are usually sufficient and a wide variety of gases will do). The governing parameter for resolution in a gas counter is the amount of energy required to produce a single ion pair. The choice of counting gas (and its purity) depends on this parameter. A common gas is P-10 (90% argon/10% methane). Quoting from Ref. 16:

"Pure argon is unsuitable, owing to the existence of a long-lived excited state which gives rise to after-pulsing. On the other hand, higher concentrations of methane give much higher operating voltages. The methane may be replaced by nitrogen, carbon dioxide, or other gases without notable loss in performance. Many gases are unsatisfactory for use in proportional counters operating at high gas gain, principally because of their electro-negative character involving the capture of electrons to form negative ions."

The gas impurities which are least tolerable are the halogens. Noble gases are often chosen partly because of the low average amounts of energy required to produce an ion pair (26.4 eV/pair for argon, 24.2 for krypton). Neon can be used, but its value is 36.6 eV/pair and it has a lower stopping power than argon at a given pressure. Methane is the most common mixing gas because of its value of 27.3 eV/pair, and because argon/methane can be easily purified (Ref. 16).

The "maximum range" of a one-MeV beta in air at STP is about 300 cm (Fig. 4); thus it is obvious that to stop betas of this energy in any gas, the vessel must be pressurized. Pressures of a few atmospheres have been commonly employed, and work has even been done on pressures above 150 atmospheres in order to stop betas in a very small geometrical area. The de-

sign of counters to minimize the problems with walls has been well discussed by Curran and Wilson (Ref. 16). These authors also discuss another technique for detecting betas of high energy: the use of a magnetic field to confine the particles to the chamber volume and increase their path length.

As with any of the techniques involving radionuclides, source preparation is a critical step in the measurement process. The problem is not as severe as for alphas, where thicknesses of a few $\mu\text{g}/\text{cm}^2$ are required; however, for low energy beta spectrometry as little as a fraction of a mg/cm^2 is necessary (see Fig. 5). Techniques such as evaporation, sublimation, freeze drying, electrolytic deposition, and lacquering have been used successfully and are discussed by Curran and Wilson (Ref. 16). Another technique, described by Phillips and Kauffman (Ref. 17), uses thin ($\sim\mu\text{g}/\text{cm}^2$) conductive films upon which the sample can be deposited. This enables the sample to be counted with essentially total geometrical efficiency.

(2) Solid Scintillation counters. The advantages of using a scintillator coupled to a photomultiplier tube for beta spectrometry are the fast time response, high gain, and relatively low noise of the system. Another advantage of solid scintillators is that they are available commercially in a variety of sizes and shapes. Since most betas will normally stop in thicknesses of the order of 1 cm of scintillator, detectors of modest size may be used.

Unfortunately, the widespread use of solid scintillators for spectroscopy is hampered by several factors. Perhaps the most important is intrinsic resolution; the ionization energy required to produce one photoelectron at the tube photocathode in these counters is about 300 eV, compared to about 30 eV/ion pair in a gas and only about 3 eV/electron-hole pair in semiconductors. This intrinsic characteristic so limits resolution measurements that can be made with scintillators that they are uncompetitive. However, it is often the case in beta-spectrometry that high-resolution systems are unnecessary because of the wide spectrum of the beta energies. Thus for many applications the intrinsic resolution of scintillators is sufficient.

As mentioned earlier backscattering is a problem with beta-ray measurements. When an ensemble of low-energy electrons impinges on a solid (scintillator) surface, a significant fraction will backscatter out before coming to rest (see Table 1). This is the result of multiple-Coulomb scattering interactions. The effect increases with atomic number. Quoting from Neiler and Bell (Ref. 18):

"About 8% of the β rays entering a flat anthracene surface from a nearby source will scatter out before being stopped and hence will produce smaller pulses than they should. For NaI(Tl), on the other hand, 80% to 90% of the entering electrons will scatter out before stopping. It is consequently quite difficult to obtain a satisfactory β -spectrum using NaI(Tl) with an external source."

Although the scattering out may be reduced by collimating the beta-particles in front of the phosphor, this is still not a sufficient remedy for the higher-Z materials.

One advantage of solid scintillators is their ease of calibration: using a gamma source, an absolute calibration can be relied upon to detect long-term drifts in pulse height due to phototube gain changes. Another advantage is the inherently low background from such detectors compared to gas-filled counters. This is due to their small size, and the ability to use electronic discrimination to bias against any minimum-ionizing high-energy cosmic rays which impinge the scintillator.

As discussed above many beta particles are accompanied by coincident gamma rays or daughter x rays. These can be measured by a two-phosphor coincidence arrangement, and can be used as a unique signature for certain nuclide decays. To take advantage of this property, one can place the unknown beta-emitter between two crystals. The use of these systems is described by Neiler and Bell (Ref. 18). Of course, a problem of beta-spectrometry in these cases is that a single-phosphor detector will have a distorted pulse-height spectrum from those events in which both the beta and the photon register in the same scintillator.

There is one application in which solid scintillators have found some

use: the measurement of gaseous beta-active nuclides (tritium, krypton-85). The section "Tritium" contains a detailed discussion of this application. The problems of backscattering and resolution are just as important, but they may be counterbalanced by the ease of sample preparation and handling.

(3) Liquid Scintillation Counters. The use of liquid scintillation counting for betas is described in detail in the section on "Tritium" elsewhere in this volume. A liquid scintillator is applicable whenever the active nuclide can be dissolved or suitably immersed in the liquid; the detection efficiency is 100% except for the betas escaping from the volume near the edges.

This technique is very sensitive, and is commonly used for carbon-14, krypton-85, and xenon-133 as well as for tritium. An important disadvantage is the cost, since liquid scintillator must be discarded after the sample is dissolved in it and counted; also, the intrinsic energy resolution is not great. The same comments which were made about solid scintillators in the previous paragraphs apply equally well here.

Although the intrinsic resolution is not high, liquid scintillators can be used to distinguish more than one isotope in the same sample. In general, the activities of two isotopes can be measured separately if the β maximum energies differ by a factor of four or more (Ref. 19). To do this, two differential discriminators are used, each one set to select the optimum energy range for one of the isotopes. However, due to the continuous nature of the spectrum, the spectra will overlap as shown in Fig. 10. The higher energy spectrum can be measured with little interference, but the low energy spectrum will contain an appreciable contribution from the high energy one. In this case, the measured result in Region I will have to be corrected for the presence of the other isotope as well as for background.

As with solid scintillators, the liquid systems can yield distorted energy spectra for radionuclides where the beta emission is accompanied by or followed closely by x-ray or gamma emission.

A possible problem which may occur with the introduction of varied or un-

Fig. 10. Spectrum that results from the decay of two beta emitting isotopes.

known samples into a liquid scintillator is quenching, either by coloration of the liquid, or by allowing chemical reactions to compete with light emission as a de-excitation mechanism for the scintillator molecules (Ref. 11).

(4) Semiconductor Detectors. The entire field of nuclear spectroscopy was revolutionized in the early 1960's by the introduction of solid state semiconductor detectors for high-resolution particle detection. The most striking advances were in the field of gamma spectroscopy, where the resolution of previous techniques (such as NaI(Tl) scintillators) was surpassed by large factors. For beta spectroscopy, the advances have been nearly as great, but the importance of the new technique for beta spectral work has been far overshadowed by gamma spectroscopy. Partly, this is due to the lack of need for very precise energy resolution in much of beta spectroscopy research because of the wide energy spectrum. However, for conversion electrons, this technique will provide very high selectivity for the mono-energetic line.

From Fig. 4, it can be seen that the maximum range in aluminum of a 1 MeV beta is 1.4mm. This is sufficiently short that betas can be totally absorbed in solid state detectors of quite modest dimensions. (Silicon is nearly identical to aluminum in its stopping power for betas.) Two problems are important, however. First, some backscattering will occur (as discussed above in the section on solid scintillators), resulting in a degradation of energy resolution on the low-

energy side for a fraction (several percent) of the incident betas. This effect is not as serious for silicon as it would be for a high-Z scintillator like NaI(Tl) because of silicon's low atomic number. Nevertheless, it contributes to the difficulty in measuring low-energy spectra in the presence of high-energy betas.

Second, a premium is placed on making the dead layer at the surface of the detector as thin as possible. For this latter reason, detectors with sensitivity almost directly at the surface have been developed. These are the silicon surface-barrier detectors.

Surface-barrier detectors have been discussed in "Alpha Particle Instrumentation", Section III.B in this volume; for convenience, we shall repeat much of that discussion here. However for a detailed discussion of the theory of operation of these devices, the reader is referred to several excellent discussions in the literature (Ref. 20-23).

A typical silicon surface-barrier detector is a cylindrical diode with a very thin p-type silicon layer on the face of an n-type silicon wafer. Above the p-type layer, a thin surface of gold (thickness ≤ 0.02 microns, ≤ 40 $\mu\text{g}/\text{cm}^2$), provides the negatively biased electrical contact; the positive bias is made through some metal contact (e.g., aluminum) on the back of the n-type silicon. The gold layer also serves as a means of protecting the active elements from contamination. Detectors with depleted depths from about 50 microns to a few millimeters are now commercially available, with surface areas from a few to thousands of square millimeters.

A beta detector using silicon or germanium has distinct advantages over a gas-filled detector. The two primary advantages are:

- For a given amount of energy lost through ionization, the number of charge carriers (corresponding to ion pairs in the gas) is greater by a factor of as much as 10. This decreases the fractional statistical fluctuations in the ion collection process, and is the main reason for the high resolution.
- Collection time in the nanosecond range makes fast time res-

olution possible. This is due to the high carrier mobilities and short distances charge carriers have to travel.

Among the difficulties associated with surface-barrier detectors are:

- The detectors are fragile and require careful storage and handling.
- Because the electrical signal is small, noise in the detector/amplifier system is often the most important characteristic around which a system is designed.
- The detectors can become contaminated, and once this happens decontamination is essentially impossible.

Although the output signals are very small, the needed associated electronics for amplification are readily available.

The detection of betas which enter the sensitive area of the detector is essentially 100%; however, backscatter loss and window absorption will have a degrading effect. Furthermore, backgrounds from external sources (e.g., cosmic rays) or from detection of gamma rays are very small because of the small size of a typical detector.

Energy resolution is the main reason for the choice of a surface-barrier detector. Broadening due to random noise fluctuations in the detector and/or preamplifier is an intrinsic resolution limitation. In addition there is broadening from backscatter losses and from variations in the energy lost in the dead initial layer. Line widths of about ± 5 eV for electrons in the 1 MeV region are typically obtained in the best systems.

c. Gross-Beta Measurements in Water

Measurements of gross-beta in water are now well-standardized. The standard technique is described in the compilation Standard Methods for the Examination of Water and Wastewater, 14th Edition (Ref. 24). (See also Ref. 25 for a tentative reference method suggested by the EPA.) This method

is used as a screening technique throughout the United States and is one of the mainstays of public-health monitoring. If gross-beta levels are shown to be sufficiently small, it is usually unnecessary to isolate and measure individual radionuclides with more complicated techniques.

The standard method deposits the sample on a counting pan by oven or hot-plate drying. Counting is carried out with an internal gas proportional counter.

In the presence of both alpha and beta activity, the separation can be accomplished by measuring the counting rate vs. voltage; a typical counter might respond as shown in Fig. 11. The first plateau is from α 's, the second from the sum of $(\alpha + \beta)$.

In its intercomparison studies program, the EPA found that for participating laboratories intra-laboratory precision for gross beta analysis appears quite satisfactory. However, the accuracy of these measurements is unsatisfactory (Ref. 26). Only 44 percent of the labs were within the control limits. One key problem with gross-beta measurements is that with any instrument there is some effective beta energy below which the betas will be absorbed and not counted; hence gross-beta measurements are intrinsically difficult to interpret.

d. Gross-Beta Measurements in Air

Measurements of gross-beta levels in air have been performed for many years, but only recently has an attempt been made to standardize the method as has been done with the Standard Methods in water. We will discuss the "Tentative Method of Analysis for Gross-Beta Radioactivity Content of the Atmosphere" (Ref. 27), adopted by the Intersociety Committee.

Air particulates collected on a filter paper are counted with an end-window Geiger-Mueller tube, thin-window proportional flow counter, scintillation counter, or solid state detector. The choice of detector depends partially upon whether or not gross-alpha measurements are desired concurrently. In that case, an internal gas proportional counter would probably be the best choice. External counting devices can

Fig. 11. Alpha and alpha + beta counting rates as a function of voltage in a proportional counter. (Reproduced from Cember, Ref. 4, with permission of Pergamon Press, © 1969).

be used to include or exclude alphas depending upon the window thickness.

The sensitivity of the method is proportional to the volume of air moved through the filter. To quote from Ref. 27, "A sample of 10 m^3 of air with a beta particulate concentration of 3 pCi/m^3 ($3 \times 10^{-12} \mu\text{Ci/cc}$) will contain 66.6 dpm, sufficient to produce acceptable precision in ten minutes' counting time."

This method is sensitive to those airborne betas which are attached to particulate material in the air. Naturally-occurring radon and thoron (gases) can provide an interference. The daughters of both radon and thoron are chemically active and attach themselves to particulate matter. Quoting again from Ref. 27,

"Since the effective radioactive half-lives of the progeny of these naturally occurring radionuclides are controlled by relatively short half-life radionuclides, namely 26.8 min ^{214}Pb and 19.7 min ^{214}Bi for radon, and 10.6 hour ^{212}Pb for thoron, their interferences may be nullified by waiting until these daughter activities are negligible before counting. This would require about 3 days for the long-lived thoron progeny."

The accuracy of measurements of this type depends upon several factors, among them the uncertainty in the volume of air sampled, the possible error from self-absorption effects which may not be properly accounted for, and calibration difficulties.

Because beta spectra are so broad and because of the possible energy-de-

pendent response of many counting instruments, it is desirable to perform calibrations with at least two or three beta sources of differing energies. If the dominant nuclide species being measured or sought is unknown, "Cesium-137 is a desirable standard, since its average energy corresponds to that of aged mixed fission products in fallout" (Ref. 27).

The main difficulty in this technique is assuring that a reliable sample is deposited on the filter. Possible large errors may result in sampling of dust-laden air. Another problem (already mentioned) is self-absorption. If the sample to be counted exceeds a few mg/cm² in thickness, a sizeable fraction of the betas may not get out of the sample to be counted without significant energy degradation. This self-absorption is a strong function of the low-energy end of the energy spectrum of the (presumably unknown) beta-emitters being measured.

In a discussion (Ref. 39) of the results of an air filter cross-check program for 1973-75, Easterly, et al. concluded that gross beta was more difficult than gross α for the participating laboratories to analyze. Only 39 percent of the labs were within the control limits for accuracy.

e. Chemical Separation Techniques

In the above discussion, one theme has been reiterated several times: because of the broad energy spectrum of betas from an individual radionuclide, the ability of even the highest-resolution beta spectrometers to identify individual emitters in a mixed sample is limited. Also, the relatively short range of betas, especially at low energies, may require that they be separated from the bulk material of the sample.

For these reasons, chemical separation techniques have been developed to concentrate one particular element (or one class of elements) prior to beta counting or spectroscopy. The chief problem, as with any chemical technique, is the possibility of uncertainties in the yield of the element(s) being sought as successive chemical procedures are performed. Of course, once a truly complete separation has been made, spectrometry is unnecessary;

a "gross-beta" count may be attributed to the radionuclide in question and to it alone.

Another problem that an analyst should keep in mind is the ubiquitous nature of the naturally occurring β -emitter ⁴⁰K. It can cause difficulties in chemical separations because of its presence in the sample being analyzed or in the reagents being used. The analyst should ascertain before using reagents whether ⁴⁰K is present (can be determined by flame photometry or γ counting).

Decay chains and parent-daughter relationships can also be used in gaining greater specificity in an analysis; a good deal of information can be obtained by following the in-growth of a daughter, or the relative quantities of nuclides following build-up and decay of a long chain (such as the chains of alpha and beta emitters among the heavy elements). The in-growth technique is particularly useful when the daughter's presence is more easily detected than the parent's (e.g., strontium-90 determinations by measurements of yttrium-90).

There is such a large number of chemical-separation techniques in the literature that a comprehensive treatment of them here is prohibitive. We shall therefore attempt only to summarize those techniques which appear in some of the major compilations.

The most complete source of information is a series of monographs done by the Subcommittee on Radiochemistry of the Committee on Nuclear Science of the National Academy of Sciences (Ref. 28). For additional information see Ref. 29. Other compilations are the U.S. DOE Environmental Measurements Laboratory Procedures Manual (Ref. 30); the handbook of the EPA Environmental Monitoring and Support Laboratory in Las Vegas (Ref. 31); the U.S. Public Health Service compilation (Ref. 32); and the Tentative Methods published by the Intersociety Committee (Ref. 27).

The beta-emitting radionuclides treated here are:

- (1) carbon-14
- (2) ruthenium-106 and 103
- (3) xenon-133 and 135
- (4) cerium-144 and 141.

Beta-emitting radionuclides treated in their own separate sections (see Di-

vision IV. "Instrumentation for Specific Radionuclides") in this volume are:

Tritium (Hydrogen-3)
Krypton-85
Strontium-89, 90
Iodine-129, 131.

Some important contributors to gross-beta measurements in environmental samples are daughters in the decay chain of radon-222. Among these daughter beta-emitters are lead-214, bismuth-214, lead-210, and bismuth-210. These are treated, but only implicitly, in the "Radon-222 and Its Daughters", section IV.

Many other contributors to gross-beta levels are more easily determined by gamma spectroscopy, and these are not treated here. For many environmental samples, gamma spectroscopy is easier, more accurate, and enables the use of larger samples in the bulk. As an example, in natural rocks, where man's impact has been minimal, one important beta emitter is potassium-40. ^{40}K has an 89% branching ratio to calcium-40 by β^- emission ($E_{\text{max}} = 1.314$ MeV), but in rock-type samples it is almost always preferable to detect the 1.460-MeV gamma which follows the electron capture decay mode (11%) to argon-40.

A majority of the beta-emitting nuclides are not treated in this Survey because they are essentially never found in environmental samples; because their half-lives are very short; because they are gamma as well as beta emitters and are measured as such; or because procedures for their chemical separation have not come to our attention.

(1) Carbon-14. Carbon-14 is one of the most well-known radionuclides; its use in "radiocarbon dating" of archeological artifacts has made it a household word. ^{14}C is a pure β^- emitter, with a very weak beta ($E_{\text{max}} = 155$ keV) which is difficult to detect in many media. The long half-life (5730 years) which makes it useful in dating is a drawback in measurements, since ^{14}C occurs in environmental samples at low activity levels. Carbon-14 is also widely employed in research, such as when organic compounds tagged with ^{14}C are used to trace biochemical processes in living systems.

The primary mechanism for natural ^{14}C production is by cosmic-ray interactions with ^{14}N in the atmosphere. The reaction is $^{14}\text{N}(n,p)^{14}\text{C}$. With its long half-life the radiocarbon ultimately reaches equilibrium with the stable carbon in the biosphere. It is not appropriate here to discuss in much detail what is now known about carbon-14 production and distribution. The interested reader is referred to other sources for more information (Refs. 33,34).

^{14}C is also produced in nuclear reactors via a number of reactions. The emissions expected from power plants and the rest of the fuel cycle constitute one of the most substantial dose contributions from nuclear power, if the ^{14}C half-life is considered.

The problems with measuring ^{14}C are in many respects similar to the problems with tritium, which also has a weak beta. Carbon-14 in biological or environmental samples is not difficult to prepare for measurement, since carbon-based chemistry is one of the most sophisticated disciplines in the world of science. For gas counting, organic samples are usually converted to carbon dioxide by combustion, and inorganic samples by decomposition with acids. The CO_2 can then be used as a gas, or alternatively converted to BaCO_3 for counting as a solid. For liquid scintillation counting, the CO_2 can be trapped in an organic base (for example, phenylethyl-amine or ethanolamine) and then introduced directly into the scintillator solution (Ref. 11).

Liquid scintillation counting is the most common ^{14}C measurement technique today. This method is as sensitive for ^{14}C as it is for tritium; activities as low as ~ 0.2 pCi/cm³ of organic solvent can be measured. Section IV.A on "Tritium" in this volume contains a full description of the various considerations in this technique, which will not be repeated here.

The other common technique for ^{14}C measurements is use of an internal gas counter (proportional, GM) with radiocarbon CO_2 introduced into the counting gas, or as BaCO_3 . Again, the "Tritium" section contains a discussion of similar techniques for tritium.

(2) Ruthenium-106 and 103. Ruthenium (atomic number 44) is one of the important fission product elements which may be emitted by nuclear fuel-reprocessing plants. The ruthenium isotope of most interest in environmental media is ^{106}Ru . Its half-life is 367 days, and it is a pure beta emitter ($E_{\text{max}} = 39.2 \text{ keV}$) leading to a short-lived daughter (Rhodium-106), which itself decays ($T_{1/2} = 30 \text{ seconds}$) to stable palladium-106 by a complicated β^- spectrum dominated by a very energetic beta ($E_{\text{max}} = 3.53 \text{ MeV}$) and several half-MeV-range gammas.

The most common beta detection technique for ^{106}Ru is measurement of the energetic beta from the rhodium daughter. This can be done, after chemical separation, with very little interference. Since the 30-second half-life of the daughter ^{106}Rh is so short, it is an ideal tag for the parent ^{106}Ru , whose own beta is so weak as to be almost undetectable in a mixed sample.

Ruthenium-103 (40 days) is also found in some samples. It has a multiple beta spectrum dominated by an 89% mode with a low-energy beta ($E_{\text{max}} = 200 \text{ keV}$) followed by a 490-keV gamma. It can be determined in a mixture with ^{106}Ru by separate counting with and without a heavy ($\sim 0.3 \text{ g/cm}^2$) aluminum filter: the ^{106}Rh daughter beta of ^{106}Ru beta will penetrate through the filter while the ^{103}Ru beta will not. Since as little as 5 mg/cm^2 of material will screen out the weak ^{106}Ru beta, it is not counted.

Ruthenium-103 and 106 can both be counted with gamma spectroscopy, and in many samples this is the preferred method; one may detect the 490-keV gamma from ^{103}Ru or a 622-keV gamma (11%) from ^{106}Ru . Indeed, the radiochemical technique in the EML Manual (Ref. 30) contains the following note: "This procedure is recommended only for those samples in which the ruthenium activity cannot be directly measured by gamma spectrometric methods."

The EML radiochemical separation technique (Ref. 30) employs ruthenium carrier. The sample is

"transferred to a distillation flask with 1:1 H_2SO_4 , cooled, and KMnO_4 crystals added. The ruthenium is distilled as RuO_4 into a receiver containing 6 N

NaOH . Ruthenium is further purified by successive precipitations as the hydrous oxide and metal."

(3) Xenon-133, 135. In nuclear reactors, radioxenons are among the important isotopes which can be discharged in air effluent; this is especially important in the boiling water reactors, where gas hold-up is short or during an accident. (See "Nuclear Reactors," Section II.B. in this volume.)

Two of the four important radioxenons are β emitters. These are:

- Xenon-133: $T_{1/2} = 5.27 \text{ days}$,
 $E_{\text{max}} = 346 \text{ keV}$.
The beta is followed by an 80-keV gamma leading to stable cesium-133.
- Xenon-135: $T_{1/2} = 9.2 \text{ hours}$,
 $E_{\text{max}} = 920 \text{ keV}$.
The beta is followed by a 250-keV gamma leading to cesium-135, which itself decays ($T_{1/2} = 3 \text{ million years}$) to stable barium-135 by β^- ($E_{\text{max}} = 210 \text{ keV}$).

Two other important radioxenons are gamma emitters. We shall list their properties here for reference:

- Xenon-131m: $T_{1/2} = 11.8 \text{ days}$,
 $E_{\gamma} = 164 \text{ keV}$, leading to stable xenon-131.
- Xenon-133m: $T_{1/2} = 2.26 \text{ days}$,
 $E_{\gamma} = 233 \text{ keV}$, leading to xenon-133, whose β^- decay has just been discussed above.

Xenon is, of course, a noble gas and as such is not amenable to normal chemical manipulation. "Chemical separation" can be accomplished in gaseous samples, if required, by purging the gas of its more reactive components (e.g., the radioiodines).

The beta-emitting radioxenons are usually monitored by techniques very similar to those used for krypton-85. These techniques are either ionization chambers, gas proportional counters, or GM tubes. The reader is referred to "Krypton-85," Section IV.B in this volume for details.

The gamma emissions are also easily detected, and where gamma spectroscopy is available it is often preferred over beta detection.

(4) Cerium-144 and 141.

Cerium-144 is one of the radionuclides most commonly associated with fresh fallout and general mixed fission products. Because of its 284-day half-life, it is sufficiently long-lived to be distributed in a wide band throughout the northern hemisphere whenever large nuclear bomb tests deposit fission products in the stratosphere. A recent NCRP Report (Ref. 35) summarizes the present knowledge of the relevant chemical, as well as physical and biological properties of radio-cerium as a basis for establishing radiation protection guidelines. The radioactive isotopes of cerium of most concern to humans are ^{141}Ce , ^{143}Ce , and ^{144}Ce . These three isotopes, all of which are β^- emitters, are abundant products of fission and have relatively long half-lives.

^{144}Ce is a beta emitter with three principal beta lines leading to praseodymium-144, which itself decays to neodymium-144 by energetic β^- emission ($E_{\text{max}} = 2.99 \text{ MeV}$) with a 17.3-minute half life. The three ^{144}Ce betas have energies and branching ratios as follows:

$E_{\text{max}} = 309 \text{ keV}$, 76%
 $E_{\text{max}} = 240 \text{ keV}$, 5%
 $E_{\text{max}} = 175 \text{ keV}$, 19%.

The frequent occurrence of cerium-141 in the same samples with ^{141}Ce can provide an interference, if a separation is desired. ^{141}Ce decays by beta emission with $T_{1/2} = 33 \text{ days}$; its two betas have $E_{\text{max}} = 582 \text{ keV}$ (30%) and $E_{\text{max}} = 442 \text{ keV}$ (70%). The daughter of ^{141}Ce is stable praseodymium-141.

One common way of separating ^{144}Ce from nearly all other interferences is by counting the energetic decays of its daughter ^{144}Pr ; using filtration with about 0.35 g/cm^2 of aluminum, only this (2.99-MeV maximum) beta survives. Because of the short 17.3-minute half-life of ^{144}Pr , equilibrium is reached very quickly, after which the disintegration rate of ^{144}Pr equals the rate for the parent ^{144}Ce .

If it is necessary to separate ^{141}Ce and ^{144}Ce activities, the ^{144}Ce activity can be measured by de-

termining the ^{144}Pr rate after filtration, as just described. In a mixture of ^{144}Ce and ^{141}Ce , the many intertwined beta spectra are characterized by several maximum energies: 442 and 583 keV from ^{141}Ce ; 175 keV, 240 keV, and 309 keV from ^{144}Ce ; and 2.99 MeV from ^{144}Pr .

The EML Manual (Ref. 30) describes three methods for the radiochemical determination of cerium-141 and 144. In each method, the final sample is counted by the ^{144}Pr method, using a thin-window GM counter. In the first, cerium-144 and a cerium carrier are ignited to insoluble oxides and fused with Na_2CO_3 at 900°C ; redissolved in HF , HNO_3 , and H_2SO_4 acids; and precipitated as ceric iodate for counting. In the second (Ref. 36), cerium is leached from an ashed sample by digestion in a mixture of H_2O_2 and HCl ; the rare earths and alkaline earths are collected as oxalates and ignited to carbonates for collection as hydroxides. Cerium is then precipitated as iodate for counting. In the third, cerium(IV) is extracted into 0.75 M di-2-ethylhexyl phosphoric acid in n-heptane from 1 M potassium bromate/10 M nitric acid solution, then back-extracted with H_2O_2 - HNO_3 , converted to oxalate, and determined gravimetrically before counting. Among these three methods, one can determine cerium in water, urine, soil, feces, vegetation, milk, food, and air filters (Ref. 30).

The second method has been used by Sutton and Dwyer (Ref. 37) to measure ^{144}Ce levels in wheat flour. Levels in the range of 10 to 20 pCi/kg were measured, simultaneously with cesium-137 at levels of about 35 to 125 pCi/kg.

Sill and Willis (Ref. 38) have described a chemical method for separating cerium and lanthanum for counting, by coprecipitating with BaSO_4 from strong H_2SO_4 solutions containing potassium salts and hydrogen peroxide. Recoveries in excess of 99% are reported.

5. Summary and Conclusions

In this section, we have attempted to outline the various techniques for detection, measurement, and energy spectrometry of beta particles in various environmental media.

The current state of the art in beta-particle instrumentation is generally satisfactory. Instruments for detecting beta particles are varied and well-developed, and in the hands of competent personnel they are capable of defining beta radiation fields to sufficient accuracy. However in the area of low energy beta emitters, more development work needs to be done in order to measure low levels in real time for situations where the identify of the β emitter is known.

Beta spectrometry is also well-developed. The requirement for high resolution is generally not severe, because most beta energy spectra are broad. Also, the best instruments (e.g., surface-barrier detectors) have far better resolution than is usually required. The principal exception is for detection at low levels of internal-conversion electrons, where resolution is important, being directly related to the ultimate sensitivity.

The principal problems needing further research and development are in the area of chemical separation techniques. At present, these techniques are often laborious, costly, and liable to error. Unfortunately, there does not seem to be any more promising approach to the identification of low-

level pure beta emitters than the use of chemical separation followed by counting. The continuous beta spectra make it very difficult indeed to separate by spectrometry alone an individual beta emitter from a mixed sample of more than two at widely differing concentrations.

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C. INSTRUMENT NOTES: BETA and BETA-GAMMA

Included in the following pages are Instrument Notes of several different kinds. The reader should be aware of the three following facts, in order to find all Instrument Notes appropriate for beta detection and measurement:

- 1) Silicon surface-barrier detectors and silicon lithium-drifted detectors are useful for both alpha and beta counting and spectrometry. Instrument Notes for all of these detectors have been incorporated in the Alpha section (mnemonic RAD-ALP), and the interested reader is referred to that section. These Notes are not included here.
- 2) Many instruments (ionization chambers, G-M counters, gas proportional counters, etc.) are advertised by their manufacturers as useful for detecting and measuring both betas and gammas. However, many similar instruments, useful for betas, are advertised as if they were only gamma instruments. We have made a somewhat arbitrary decision to place the Instrument Notes according to the manufacturers' claims. Hence both this section (RAD-BET and RAD-BET,GAM) and the section on gamma/x-ray detectors (RAD-GAM and RAD-GAM,XRA) should be consulted together for a fuller overview of instruments of this type.
- 3) Many instruments are advertised as sensitive to alphas, betas, gammas, x-rays, and in some cases neutrons as well. Often these instruments consist of a variety of probes attachable to a central electronic-counting unit. These instruments have all been placed in the section entitled "Combination Instruments", with the appropriate mnemonic (e.g., RAD-ALP,BET,GAM).
- 4) Liquid-scintillation systems are usually used for beta-detection and measurement. However, they have all been placed in the section RAD-NUC, "Radionuclide Instrument Notes."

Soft-Beta Counting Spectrometer
Beckman LS 8000 Series
(LS 7000, LS 7500, LS 8000, LS 8100, LS 9000)

Class	Laboratory, automated sample counter, printer
Principle of Operation	Liquid scintillation system. Detects light from sample by matched multiplier phototubes in coincidence. The total response for the decay product is obtained by summing the outputs of the phototubes
Sensitivity and Range	<p>Virtually all alpha, beta, X-ray, positron and gamma emitters properly prepared can be counted.</p> <p>Range: 0-2 MeV</p> <p>Maximum Count Rate: ^3H, 5.6×10^6 ^{14}C, 5.6×10^6 ^{32}P, 1.0×10^6</p> <p>Maximum Counts: 5.6×10^6</p> <p>Live Time Accuracy: ± 0.5 microseconds</p>
Sampling	Batch sampling, 300 sample capacity
Performance	<p>Operating temperature range: 15°C - 40°C, to 95% humidity</p> <p>Efficiency: ^3H $\geq 60\%$; ^{14}C $\geq 95\%$; ^{14}C above ^3H endpoint $> 70\%$</p> <p>Crosstalk: in ^3H window set at 50%; ^3H efficiency, ^{14}C efficiency $< 10\%$ in ^3H window set at 1% spillover, ^{14}C efficiency $< 20\%$</p> <p>Power failure memory protect - minimum time 3 hours</p> <p>E^2/B; ^3H ≥ 180; ^{14}C ≥ 250; ^{14}C above ^3H > 400</p> <p>Quench monitoring: $\pm 1\%$ of counting efficiency using AQC</p> <p>Coincidence gate resolving time: < 30 nanoseconds</p>
Requirements	<p>Power: < 8 amps at 120V $\pm 10\%$ A.C., 60 Hz</p> <p>Size: 47"Wx30"Dx57"H</p> <p>Weight: 230 lbs.</p>

- Features
1. Two or three channels of analysis
 2. Microprocessor controlled
 3. Ambient temperature
 4. Multiuser capability by use of library programs called up by command towers
 5. Automatic calibration
 6. Quench monitoring by H# or SCR
 7. Automatic quench compensation (AQC)
 8. Automatic data computation (ADC) for % bound applications
 9. Output can be counts, CPM, or % bound
 10. Computer compatibility and many accessories available

References Manufacturer's Specifications

Cost	LS 7000	\$16,900
	LS 7500	20,500 - \$22,300
	LS 8000	22,900 - 27,900
	LS 8100	25,400 - 30,400
	LS 9000	31,800 - 34,100

Address Beckman Instruments, Inc.
Jamboree at Campus Drive
P.O. Box C-19600
Irvine, CA 92713
(714) 833-0751

INSTRUMENTATION
FOR ENVIRONMENTAL
MONITORING

RAD-BET, GAM
Portable Survey
Eberline
May 1980

Beta, Gamma Survey Meters

Eberline Portable Models

E-130G

E-530

6112

PIC-6A

RT-1

Class	Portable, handheld				
Sampling	Continuous				
Model	RT-1A	E-130 A or E-130C	PIC-6A	E-520/HP-270	E-530/HP-270
Radiation Measured	Gamma	Gamma	Gamma	Beta-Gamma	Beta-Gamma
Detector	GM tube	GM tube	Ionization chamber	GM tube	GM tube
Range	Approx. 25 chirps/min in a 10 mR/hr field	A-10,100,1000 mR/hr C-100,1000,10,000 mR/hr	1 mR/hr to 1000 R/hr (6 decades)	0-2000 mR/hr (5) ranges	0-200 mR/hr (4) ranges
Scale	None	Linear	Linear	Linear	Linear
Accuracy		±5% F/S*	±20% of reading	+8%(0.2mR/hr) +15% F/S(0-200 mR/hr) +10% F/S(0-2000mR/hr)	±5% F/S
Temperature	23°C to 54°C (-10°F to +130°F)	-40°C to +60°C (-40°F to +140°F)	-23°C to 60°C (-10°F to 140°F)	-40°C to +60°C (-40°F to +140°F)	-40°C to +60°C (-40°F to +140°F)
Power	9v Neda #1604 Batt. (nominal) 300 hours	2 "D" cells 300 hours (C-Zn cell)	(2) 1604 Neda type battery	2 "D" cells 300 hours for C-Zn cell	2 "D" cells 300 hours for C-Zn cells
Weight	0.17 kg (6 oz)	1.4 kg (3 pounds)	1.48 kg (3 1/4 pounds)	2 kg (4 3/8 pounds)	2 kg (4 3/8 pounds)
Size	3x6x7.9cm (1 3/16" Thk x 2 3/8"Wx3 1/8"H)	17.1x8.6x6.0cm (6 3/4"Lx3 3/8"Wx6"H) including handle	21.3x10.2x14.3cm (8 3/8"Lx4"W x5 5/8"H)	10.2x20.3x17.1cm (4"Wx8"L 6 3/4"xH) handle included	10.2cmWx20.3cmL x17.1cmH(4"Wx8"L x6 3/4"H) handle included
Features	Audible Warning	Variable meter response time	Beta window in bottom	Both mR/hr & CPM scales	Both mR/hr & CPM scales
Remarks	Pocket Monitor	Battery voltage regulated	Weather proof housing	Utilizes 2 detectors	Splash and dustproof
Cost	\$165.00	\$360.00	\$700.00	\$485.00 with HP-270 probe	\$485.00 with HP-270 probe

*F/S - full scale

Eberline Survey Meters

Model	6112	E-120/HP-270	RO-7
Radiation Measured	Gamma	Beta Gamma	Gamma; Mid and High Range Probes also measure beta
Detector	GM Tubes	GM Tubes	Ion Chamber
Range	1 mR/hr-1K R/hr 5 ranges	0.5, 5, 50 mR/hr 600, 6K, 60K	20,000 R/h 200 R/h 2 R/h
Scale	Linear	Linear	Digital
Accuracy	$\pm 10\%$ vs Cs^{137} $\pm 20\%$ of reading	$\pm 5\%$ F/S	N/A
Temperature	-17.7°C to +48.9°C 0°F - 120°F	-40°C to +60°C -40°F to +140°F	-10°C to 55°C 15°F to 130°F
Power	Four 1.5 volt "C" cells 20-60 hrs	2 "D" cells 300 hours for C-Zn cell	3 (9v) NEDA 1604 2 (30v) NEDA 210
Weight	3.18 kg (7 pounds)	1.36 kg (3 pounds)	1.1 kg (2.5 pounds)
Size	91.4cmLx13.3cmW x8.3cmH 36"Lx5 1/4" Wx3 1/4"H (probe retracted)	17.1cmLx8.6cmW x(6.03cmx7.94cm)H [6 3/4"Lx3 3/8"W x(2 3/8"+3 1/8")H	20.3cmLx10.8cmWx24cmH (8"Lx4 1/4"Wx 9 1/2"H)
Features	Telescoping probe To 15 ft 1 in.	Calibration source included	Available with low, medium and high range probes; underwater probe housing and with probe extender for use in air. Medium and high range probes also measure beta absorbed dose rate.
Remarks		E-120E has cpm meter face only	Has exchangeable probes

Eberline survey meters (continued)

Cost	\$2975.00	Less probe \$240.00 with HP-270 \$310.00	Instrument w/o detector \$730.00 Low Range Detector \$510.00 Medium Range Detector \$460.00 High Range Detector \$490.00 Underwater Housing \$455.00
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Features	Some models have earphone and speaker outputs, regulated power supply, individual calibration controls for each range		
Address	Eberline Instrument Corporation P.O. Box 2108 Santa Fe, NM 87501 (505) 471-3232

Stationary Air Monitor
Particulate Air Monitor AMS-3

Class	Stationary
Principle of Operation	Air filter collects particulates. Collected beta and gamma radioactivity is detected and measured by a pancake-type geiger tube. Meter readout, and strip chart recorder. Air pump RAP-1 included.
Sensitivity and Range	1) Counting efficiency for 47mm diameter source: ^{99}Tc (15%), ^{90}Sr ^{90}Y (75%); $\sim 2\pi$ efficiency 2) Background <500 counts/minute when air flow is 1 ft ³ /minute, measured in Santa Fe, NM. Gamma background subtracted automatically by second detector. 3) Range: logarithmic from 10 to 100,000 cpm
Sampling	Continuous
Performance	Accuracy Linearity 10% of reading Temperature: -29°C to 60°C (-20°F to +140°F)
Requirements	Power 105-125 VAC, 50-60 Hz at approximately 0.3 amps Size: 64 x 39 x 39 cm (25"W x 15"W x 15"D) Weight 73 kg (160 lb)

Features

- 1) Model RAP-1 regulated air pump is furnished with AMS-3.
- 2) Can be calibrated for many different isotopes
- 3) Sensitivity 11 cpm for 10^{-12} $\mu\text{Ci/cc}^{90}\text{SrY}$ after four hours operation at 60 lpm air flow rate.
- 4) Geiger-Muller detector is pancake-type, 1.75" diameter, 1.4 to 2.0 mg/cm² mica window. Second detector provided for automatic gamma background subtraction.
- 5) Shield is equivalent to 2" of lead.
- 6) Takes 2" filter paper. A box of 50 filters (Type HV-70) is provided.
- 7) Air intake normally about 1 ft³/minute (28.3 liters/minute). Air flow meter goes from 0 to 100 liters/minute.
- 8) Alarm indicator is flashing red light, bell, and relay contacts for remote alarm. Alarm point adjustable above 10 cpm.
- 9) Strip-chart recorder.

References Eberline Catalog

Cost \$3,145.00

Address Eberline Instrument Corporation
P.O. Box 1208
Santa Fe, New Mexico
(505) 471-3232

INSTRUMENTATION
FOR ENVIRONMENTAL
MONITORING

RAD-BET, GAM
Stationary Portal
Monitor
Eberline 3
May 1980

Hand & Foot, Portal Monitors

Eberline HFM-4A/HFM-5A, PMC-4B/PMP-4C

HFM-4A

HFM-5A

PMC-4B

PMP-4C

Class	Stationary	
Principle of Operation	Geiger-Muller tubes (portal monitor) or gas flow proportional detectors (hand and foot monitor) measure beta/gamma radiation, meter readout. HFM-4A and HFM-5A are hand/foot units, PMC-4B/PMP-4B is portal unit.	
Sensitivity range	HFM-4A: or HFM-5A: 0-999 counts preset level. HFM-4A for alpha and beta. HFM-5A for beta only. PMC-4B/PMP-4C: 160 to 7000 counts/minute alarm settings Counting Times: (HFM-4A or HFM-5A)) 6 sec, 12 sec, 30 sec, 60 sec.	
Sampling	Continuous	
Performance	Accuracy: Linearity $\pm 5\%$ of full scale	
Requirements	<u>PMP-4C</u>	<u>HFM-4A or HFM-5A</u>
	Power: 115 VAC, 60 Hz	115 VAC, 50-60 Hz
	Size: 201cm x61 cm (79"x24")	1.5mx0.7mx1.04m (59.25"Hx27.5"Wx41"D)
	Weight: 17.7 kg (39 lb)	113 kg (250 lb)
	<u>PMC-4B</u>	
	Power: 115 VAC, 50-60 Hz	
	Size: 19.1cmx11.4cmx19.1cm (7.5"Hx4.5"Wx7.5"D)	
	Weight: 4.2 kg (9 1/4 lb)	
Features	<u>PMP-4C Portal</u>	
	(1) Geiger-Muller tubes, 5/8" diameter 11" long with 30 mg/cm ² walls. 7 tubes in frame, 4 smaller tubes along foot channel.	
	(2) Two alarm lights per channel.	
	(3) 10-foot cable to PMC-4B console.	
	<u>PMC-4B Console</u>	
	(1) Response time to 63% of final reading: about 7 seconds	
	(2) Failure indication feature if no pulses counted in 30 seconds	
	<u>HFM-4A or HFM-5A Hand/foot Monitor</u>	
	(1) Gas flow proportional detector, active area of each of 4 hand detection: 50 in ² = 323 cm ² ; active area of each foot detector 78 in ² = 503 cm ²	
	(2) 4 efficiency 15% (HFM 4A only) 4 efficiency 20%	
	(3) HM-4A is available for only hand monitoring.	
References	Manufacturer's Specifications	
Cost	PMC-4B console/PMP-4C portal	\$3080.00
	HFM-4A	7500.00
	HFM-5A	6995.00
Address	Eberline Instruments Corporation P.O. Box 2108 Santa Fe, NM 87501 (505) 471-3232

Ion Chamber
Fisher Controls Model NE 014

Class	Laboratory
Principle of Operation	Combination of re-entrant cylinder, gamma chamber, and a parallel-plate beta chamber containing air at atmospheric pressure
Sensitivity and Range	Energy Sensitivity: (Gamma) 100 keV to 3 MeV
Sampling	Continuous
Performance	Accuracy: Gamma, $\pm 1\%$; Beta, $\pm 3\%$ Response time: Zero
Requirements	Power: 100 pW per pA Size: 25.4cm Hx25.4cmD (widest part) [10"H x10"D] Weight: 4.15 kg (9.1 lbs)
Features	A.E.R.E. Type 1383A
Cost	Refer to manufacturer
Address	Fisher Control Ltd. Measurement and Analytics Division Century Works, Lewisham London, SE13 7LN, England Attn: A.E. Pegg Tele. 01-692 1271 Ext. 260 Telex 22469

Beta-Gamma Survey Meter

Health Physics Instruments Model 1075

Class	Portable, handheld																								
Principle of Operation	Uses two air equivalent ionization chambers, one $\beta\gamma$ and other γ plus ratemeter with amplifier and digital readout																								
Sensitivity and Range	Energy range: 20 keV to 1.3 MeV ($\beta\gamma$, if use energy response curve) 30 keV to 1.3 MeV (γ , if use energy response curve) Dose rate range: 0.001 to 200 rad/h in three digital ranges: 200 rad/h: 0.1 to 199.9 rad/h 20 rad/h: 0.01 to 19.99 rad/h 2 rad/h: 0.001 to 1.999 rad/h																								
Sampling	Continuous																								
Performance	Accuracy: overall, $\pm 15\%$ or two digits, which ever is greater (elevation, barometric corrections must be made) Temperature dependence: zero shift, ± 0.005 , 0-50°C reading, $\pm 10\%$ from 10° to 50°C operating range, 10°C to 50°C storage limits, -20°C to 65°C Pressure dependence: reading dependent on elevation and barometric pressure, detectors are unsealed ionization chambers Exposure rate response: 90% collection efficiency at 200 rad/h Off scale indication: 1- - -; a numeral in the left-hand digit with all other digits blanked. Switching transient: range switching transient; 5 seconds between adjacent ranges Detector volume: 300cc each Detector wall thickness: $\beta\gamma$: 7 mg/cm ² γ : 1 g/cm ² Warm-up time: 30 sec. Response time: 3.5 second on 2 rad/h range; <1 second on other ranges Energy Response: <table><tr><td></td><td colspan="2">Reading/True Dose rate</td></tr><tr><td>keV*</td><td>$\beta\gamma$**</td><td>γ</td></tr><tr><td>22</td><td>.72</td><td>.05</td></tr><tr><td>32</td><td>.82</td><td>.31</td></tr><tr><td>46</td><td>.87</td><td>.56</td></tr><tr><td>72</td><td>.94</td><td>.76</td></tr><tr><td>120</td><td>1</td><td>.88</td></tr><tr><td>Co⁶⁰</td><td>1</td><td>1</td></tr></table>		Reading/True Dose rate		keV*	$\beta\gamma$ **	γ	22	.72	.05	32	.82	.31	46	.87	.56	72	.94	.76	120	1	.88	Co ⁶⁰	1	1
	Reading/True Dose rate																								
keV*	$\beta\gamma$ **	γ																							
22	.72	.05																							
32	.82	.31																							
46	.87	.56																							
72	.94	.76																							
120	1	.88																							
Co ⁶⁰	1	1																							
	* Filtered x-rays (HF series filter) **With 1/8" equilibrium wall added.																								
Requirements	Power: batteries, 4D cells; 1AA (light) Size: 17.9cmHX13.3cmWx30.5cmL (7"x5.25"x12") Weight: 2.5 kg (5.5 lbs.)																								
Features	Carrying case in black analyzed aluminium environmentally sealed Detector removable																								

* Filtered x-rays (HF series filter)

**With 1/8" equilibrium wall added.

INSTRUMENTATION
FOR ENVIRONMENTAL
MONITORING

RAD-BET, GAM
Survey
Health Physics
Instruments
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References	Manufacturer's Specifications
Cost	\$2,400.00
Address	Health Physics Instrument, Inc. 124 Santa Felicia Drive Goleta, CA 93017 (805) 685-2612

Hand and Shoe Monitor

Johnson Model UM-4

Class	Laboratory
Principle of Operation	Uses GM tubes for both hand and foot, amplifier, LED display, audible-visual alarm
Sensitivity and Range	Range: 0-100,000 cps
Sampling	Continuous
Performance	Accuracy: better than $\pm 5\%$ Display response: instant
Requirements	Power: 115 VAC at 0.24 amps Size: Electronic section 43.2cm x38.1cm x21.3cm (17" x15" x 8 3/4") Base pedestal - 45.7cm x71.1cm x7.6cm (18" x28" x3") Weight: 56.7 kg (125 lbs)
Features	Paper roll protects shoe detectors
References	Manufacturer's Specifications
Cost	\$4,200.00
Address	William B. Johnson and Associates, Inc. Research Park, Boonton Ave. Montville, N.J. 07045 (201) 344-9222

Survey Meter
Nuclear Associates 05-700

Class	Portable Handheld
Principle of Operation	GM tube, amplifier, meter readout.
Sensitivity and Range	Energy Sensitivity: Beta: 175 keV minimum Gamma: 20 keV through cobalt-60 γ 's Ranges: 0.5, 5, 50 mR/hr (300, 3000, 30,000 counts/min)
Sampling	Continuous
Performance	Meter Accuracy: $\pm 2\%$ of full scale Temperature Range: 15°C to 75°C Response Time: One to three seconds dependent on scale
Requirements	Power: Four 1.5 V "D" cells Size: 22 cm L x 11 cm W x 18 cm H (8.5" L x 4.5" W x 7" H) Weight: 1.82 kg (4 lbs.)
References	Manufacturer's specifications
Cost	\$250.00 for 05-700 \$ 16.00 for 05-701 (optional headset)
Address	Nuclear Associates 100 Voice Road Carle Place, N.Y. 11514 (516) 741-6360

RAD-BET, GAM
Survey
Nuclear Enterprises
May 1980

Beta/Gamma Dose Ratemeter
Nuclear Enterprises Model 0500

Class	Portable, handheld
Principle of Operation	Ionization chamber (500cc), amplifier, meter readout
Sensitivity	Energy range: 10 keV to 10 MeV Dose Rate Range: XI - 0 to 3, 3 to 10, 10 to 100, 100 to 1000 mrad/hr X10 - 0 to 30, 30 to 100, 100 to 1,000, 1,000 to 10,000 mrad/hr Dose: 0 to 1,000 mrad
Sampling	Continuous
Performance	Temperature range: 10°C to +40°C Energy response: with end cap-gamma response uniform from 50 keV to 10 MeV; without end cap-low energy response is extended down to 10 keV see Fig. above
Requirements	Power: Two 1.5V "D" size cells; Two 22.5V batteries type 15-F-15 Size: 18.7cmHx14.7cmWx28.8cmD (7.4"x5.8"x11.3") Weight: 13.2 kg (7 lbs.)
Features	RF shielded; sealed to eliminate the effects of humidity. Developed in conjunction with UKAEA at Harwell.
References	Manufacturer's Specifications
Cost	Available on request
Address	Nuclear Enterprises, Inc. 931 Terminal Way San Carlos, CA 94047 (415) 592-8663

Hand and Foot Monitor

Nuclear Research Model HFM-2

Class	Stationary
Principle of Operation	Detects and measures beta-gamma contamination on hands, shoes, hair, face and clothing. Seven halogen-quenched GM tubes (4 for hands, 2 for shoes, 1 in external probe); hand and shoe detectors shielded with 0.5" lead, ratemeter amplifier, alarms.
Sensitivity and Range	Energy response: Scaler range: 0 to 99,999 counts (3-four digit registers), one for each hand, and one for shoes.
Performance	Preset time: 1 to 30 secs, variable
Requirements	Power: 105-125 VAC, 50/60 Hz Size: 173cmH x 53cmW x 56cmD, (68 3/8"H x 21 x 22"D shoe deck: 10cmH x 53cmW x 45cmD (4"H x 21"W x 18"D) Weight: 300 kg (650 lbs) shipping.
Features	Allows self-examination for contamination, audible response in proportion to degree of contamination. External survey probe.
References	Manufacturer's Specifications
Cost	Available upon request
Address	Nuclear Research Corp. 1105 Industrial Highway Southampton, PA 18966 (215) 357-5015

Beta-Gamma Air Monitor

RADeCO Model GM-222

Class	Laboratory continuous monitor for airborne beta-gamma emitters	
Principle of Operation	Filter, GM tube, amplifier, meter/recorder readout.	
Sensitivity and Range	Range: 1K, 10K, 100K counts/min Energy range: 150 keV up Air flow: 0.5 to 3 cfm, front panel indicator Probe: Detectors: 15-2, 3-4 or 5-6 mg/cm ² Quench: Halogen Operating Voltage: 900 V dc Shield: 2" lead equivalent, nominal	
Performance	Accuracy: $\pm 10\%$ of calibration Temperature: 40°F - 120°F Filter: 47mm Air flow: $\pm 2\%$ of set level	Beta Efficiency: 90Sr-90Y: 35% 14C: 5% Time constant: continuously variable 3 to 15 seconds
Requirements	Power: 110V, 60 Hz Size: 41.9cm x 22.2cm x 22.9cmD (16 1/2" x 8 3/4" x 9"D) Weight: 31.8 kg (70 lb)	
Features	Threshold adjustable audible alarm; External air mover accessory; stainless steel air path; fail safe indication.	
Reference	Manufacturer's Specifications	
Cost	GM222A with 2" lead shield - \$4400.00 with meter GM222 - 4775.00 with recorder GM222-1 with 3" lead shield - 5100.00 with recorder GM222A-1 with 3" lead shield - 4775.00 with meter

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MONITORING

RAD-BET, GAM
Continuous Air Monitor
RADeCO
Page 2

Address RADeCO
 4060 Sorrento Valley Boulevard
 San Diego, CA 92121
 (714) 452-9983

v

High Range Survey Meter

Model CP-TP-54A
Model CP-TP-54B
Model CP-TP-10K

Class	Portable (handheld)
Principle of Operation	Extendable air ion chamber, MOSFET amplifier, meter readout
Sensitivity and Range	CP-TP-54A: 0 to 0.5, 5, 50, 500 R/hr CP-TP-54B: 0 to 5, 50, 500, 5000 R/hr CP-TP-10K: 0 to 10, 100, 1000, 10,000 R/hr
Sampling	Continuous
Performance	Accuracy: $\pm 5\%$ full scale Time constant less than 2 seconds on all ranges
Requirements	Power: Single "D" cell Size: 13" x 6 1/2" x 4" (excluding handle and with chamber close-coupled) Weight: 4 lbs.
Features	High dose rate capability (to 10,000 R/hr) Interchangeable ion chambers Probe may be close-coupled, extended up to 200" on rod or extended on 25' cable Can monitor around a corner

References	Manufacturer's Specifications
Cost	CP-TP-54A: \$1325.00 CP-TP-54B: \$1270.00 CP-TP-10K: \$1330.00 Includes 40" extension rod Additional 40" extension rods: Upon request
Address	Technical Associates 7051 Eton Avenue Canoga Park, CA 91303 (213) 883-7043

Wide Range Recording Area Monitoring System

Model WRAM

Class	Portable, remote probe
Principle of Operation	Pressurized ion chamber, log response count rate meter, strip chart recorder, alarms
Sensitivity and Range	Range 1: 0.01 mR/hr to 100 mR/hr Range 2: 100 mR/hr to 1000 R/hr
Performance	Time constant automatically varies from less than 1 second at highest level to 60 seconds at lowest level. Up to 300 hrs. continuous running time for PLR-5. 100 hrs. between charges for recorder.
Requirements	Power: PLR-5 ion chamber and electronics: Three 1.4 V mercury cells Strip chart recorder: 12 C-size nicads (built-in charger) Size: 12"W x12"H x12"D excluding handle Weight: 14 lbs.
Features	Remote operation (up to 5000 ft. cable) Portable Permanent monitoring record without need of outside power Unit is dust and fume resistant Remote alarm (Model RAL-4) optional
Cost	WRAM: \$2870.00 RAL-4 remote alarm (optional): \$80.00
Address	Technical Associates 7051 Eton Avenue Canoga Park, CA 91206 (213) 883-7043

Beta-Gamma Particulate Air Monitor

Model FM-5ABN

Class	Laboratory
Principle of Operation	Motor blower, filter paper, 2" Pancake GM detector, count rate meter, strip chart recorder, speaker, alarms
Sensitivity and Range	Range: 0-500, 5000, 50,000, 500,000 cpm Limit of sensitivity: 10^{-4} μc (betas > 0.1 MeV) or 10^{-8} $\mu\text{c/ml}$ minute exposure
Sampling	Continuous
Performance	Accuracy: linearity $\pm 2\%$ full scale Background: typically 10-15 cpm Air flow adjustable to 100 liters/MIN
Requirements	Power: 115 VAC, 50-60 Hz Size: 43.2cmW x 33.0cmH x 34.3cmD (17"W x 13"H x 15 1/2"D) Weight: 18.2 kg (40 lbs.)
Features	Quick change filter Audio and visual alarms
Reference	Manufacturer's Specifications
Cost	\$1770.00
Address	Technical Associates 7051 Eton Avenue Canoga Park, CA 91303 (213) 883-7043

Portal Monitors

Technical Associates Model PPM-21 and PPM-21A

Class	Stationary
Principle of Operation	Geiger Muller Tubes detect beta-gamma. Scaler circuitry sums pulses. Indicator lights for contamination, system failure alarm, location of contamination, safe operation, etc. Timer and light beam switching included in PPM-21A.
Sensitivity and Range	8 channels (11 GM detectors) Detector position light up--console lights indicate on human silhouette location of contamination. Presetable alarm level: 10-990 counts occurring within recycling time (presetable) for PPM-21 or within preset time (1-99 sec) for PPM-21A.
Sampling	PPM-21 Recycling count (digital ratemeter operation) PPM-21A Light beam switching - scaler in preset time
Performance	Scaler accuracy for counting ± 1 count Timer accuracy $\pm 1\%$
Requirements	Power 115 VAC, 50-60 Hz (or 230 V) Size 223.5cmH x63.5cmW x71.1cmD (88"H x25"W x28"D) console 22.9cmH x25.4cmW x33.0cmD (9"H x10"W x13"D) Weight 81.6 kg (180 lbs) total weight
Features	Scaler counting (high accuracy), low level alarm Detector position light up Audio and visual alarms Light beam switch and timer warning system (PPM-21A)

INSTRUMENTATION
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RAD-BET, GAM
Stationary
Technical Associates 4
Page 2

References	Manufacturer's Specifications
Cost	PPM-21: \$2000.00 PPM-21A: \$2650.00
Address	Technical Associates 7051 Eton Avenue Canoga Park, CA 91303 (213) 883-7043

Beta-Gamma Background Suppress Portal Monitor

Model PPM- 25 P

Class	Stationary
Principle of Operation	GM tubes detect radiation above a varying background level. Can activate presetable alarms.
Sensitivity and Range	8 channels (18 Pancake--2" O.D.--GM tubes) Detector position light up--console lights indicate on human silhouette location of contamination
Sampling	Background is constantly monitored and updated except when a person walks within sensor range - then detectors switch to monitoring person. Increased radiation level is monitored for 2 seconds or less
Requirements	Power: 115 VAC, 50-60 Hz Size: 223.5cmH x63.5cmW x71.1cmD (88"H x25"W x28"D) Weight: 81.6 kg (180 lbs) total weight
Features	Very fast contamination checks possible with continuously updated background Audio and visual alarms Detector position light up
Cost	\$5800.00
Address	Technical Associates 7051 Eton Avenue Canoga Park, CA 91303 (213) 883-7043

Radiation Survey Meters

Texas Nuclear Models 9120, 9121, 9122

Class

Portable

Principle of
Operation

A geiger-muller tube, amplifier, and meter

Sensitivity
and Range

	Model 9120	Model 9121	Model 9122
Operating Range (mr/hr)	0.02 to 200	0.2 to 2000	2.0 to 20,000
Minimum Beta Sensitivity	Shield open 160 keV Shield closed 1700 keV	N/A	N/A
Energy Dependence	Compensated over the range of 80 keV to 1.3 MeV		
See instruction manual for response curve			

Sampling

Continuous

Performance Accuracy: $\pm 30\%$ calibrated with 0.66 MeV γ -rays, ^{137}Cs
 Operating temperature: -30°C to $+60^{\circ}\text{C}$
 Response time: To 90% of final reading for step increase
 1st decade about 1 minute
 2nd decade about 10 sec
 3rd decade about 2 sec
 4th decade about 1 sec
 Temperature Dependence: $<12\%$ of reading from 10°C to 40°C

Requirements Power: Three 1.5 Ni-Cd cells, 35 hours continuous operation
 Size: Model 9120 - 40.0cm x 8.9cm x 7.6cm (15.75"L x 3.5"W
 x 3"D)
 Model 9121
 9122 35.9cmL (14 1/8"L)
 Weight: Model 9120 - 1.5 kg (3.3 lb)
 Model 9121
 9122 - 1.4 kg (3.1 lb)

Features Log scale readout, battery check circuit, earphone jack,
 optional recharging base which adapts instrument to area
 monitoring, full scale reading maintained up to 1000 R/hr

References Manufacturer's Specifications

Cost Model 9120 \$675.00
 Model 9121 640.00
 Model 9122 640.00

Address Texas Nuclear Division
 P.O. Box 9267
 Austin, Texas 78766
 (512) 836-0801

High Level Beta-Gamma Doserate Meter

Harwell Types 3247 and 3248

Class	Portable, handheld
Principle of Operation	Uses a GM detector mounted on extendable tube (4 in) for remote detection and survey of localized high activity spills and log ratemeter with visual meter display.
Sensitivity	Energy range (γ) 50 keV to 3 MeV Dose rate range (γ) 0.1 to 100 rads/hr (type 3247) (logarithmic) 5 to 5000 rads/hr (type 3248)
Sampling	Continuous
Performance	Dose rate accuracy $\pm 33 \frac{1}{2}\%$ @ 20°C (type 3247) Operating temperature range -5°C to 45°C Controls function switch, 4 position switch off-batt. check-set zero - on. Set zero, coin operated control for adjustment of electrical zero
Requirements	Power Batteries, 2 U2 type (or equivalent), life 45 hours Size Meter unit, 11.5cmWx10.2cmHx16.5cmL (4 1/2"x4"x6 1/2") plus 3.8cmDx14cm (1 1/2x5 1/2") rear handles, extension rod 76.2cm (2ft.6") long Weight ~ 2 kg (4 1/2 lb) including probe cell
Features	Additional 2m long extension rods available. Failure Indication a small priming source (^{90}Sr) is mounted on inside wall of energy radiation compensating shield to enable the meter to be set to an electrical zero that is offset from the mechanical zero. Thus if the probe should be disconnected or inoperative, the meter pointer will fall back to mechanical zero so that it lies in FAULT sector of the scale.
References	Manufacturer's literature
Cost	On application
Address	AERE, Harwell Oxfordshire OX11 ORA England Tele (0235) 24141 Attn R. H. W. Sherwood, Bldg. 347.3 Electronics and Applied Electronics Ext. 4766 R. G. Powell, Bldg. 329 Marketing and Sales Department Ext. 2942

Radiation Survey Meters

Victoreen

Model	692	808D
Radiation Detected	X-ray and Gamma above 30 keV	Gamma
Range	0-10, 0-100, 0-1,000 mR/hr 0-10,000	0.1 mR/hr to 100 mR/hr (3 decade log scale)
Detector	Ion chamber	GM tube
Energy Dependence	$\pm 10\%$ from 45 keV to 1.3 MeV	$\pm 20\%$ from 80 keV to 2 MeV
Precision	$\pm 15\%$ f/s	$\pm 15\%$ f/s
Power Requirement	2 'D' cells	90-130 VAC, 60 Hz or 12 V dc; 220 V ac available
Features	Specifically designed for non- destructive testing.	Fail safe alarms Visual and audible alarms
Overall Dimensions	30.5cmLx12.1cmWx17.1cmH (12"Lx4-3/4"Wx6-3/4"H)	10.8cmDx35cmWx23.5cmH (4-1/4"Dx13-3/4"W x9-1/4"H)
Net weight	2.25 kg (5 pounds)	5.1 kg (11-1/4 pounds)
Cost	\$625.00	\$1110.00
References	Manufacturer's Specifications	
Address	Victoreen Inc., 10101 Woodland Ave. Cleveland, Ohio 44104 (216) 795-8200

Condenser Roentgen-Meter

Victoreen Model 570

Class	Portable, handheld
Principle of Operation	External ionization chamber, meter readout
Sensitivity and Range	See chart
Requirements	Power: 105 - 130 VAC, 50-60Hz (220 VAC available on request) Size: 17.8cmLx10.2cmWx17.8cmH(7"Lx4"Wx7"H) Weight: 2.7 kg (6 lbs)
Cost	\$1130.00 plus chambers
Address	Victoreen, Inc. 10101 Woodland Avenue Cleveland, Ohio 44104 (216) 795-8200

Model No.	Range (roent)	(R/Sec.) Intensity Limits* Model 570	Energy Response			Volume (Cm ³) Approx.	Nominal Wall Thickness mg/cm ²	Approx. Thimble Dimensions		Beta cut off (MeV)
			Tolerance	(keV eff)	Material			Dia. x Length		
								Inches	cm	
LOW ENERGY CHAMBER										
651	250	90	± 5%	6-35	Mylar	0.20	6.6	5/8 x 5/8	1.59 x 1.59	.066
MEDIUM ENERGY CHAMBERS										
188	0.025	24 x 10 ⁻⁵	± 5%	30-660	Phenolic	1450	212	4 1/2 x 5 5/8	11.4 x 14.3	.62
130	0.25	9 x 10 ⁻³	±10%	42-660	Phenolic	177	212	2 x 5	5.08 x 12.7	.62
227	1	0.195	±10%	30-660	Phenolic	41	212	1 1/4 x 3 1/4	3.18 x 8.25	.62
633	2.5	0.3	±20%	30-660	Phenolic	17	212	1 1/4 x 1 5/8	3.18 x 4.13	.62
228	5	5.1	±20%	30-660	Phenolic	8	212	3/4 x 2	1.91 x 5.08	.62
326	10	5.1	±10%	30-660	Polystyrene	6	170	3/4 x 1 1/4	1.91 x 3.18	.55
70-5	25	45	± 5%	30-660	Nylon	1.83	67	1/2 x 1	1.27 x 2.54	.27
131	100	75	± 5%	30-660	Nylon	0.46	89	3/8 x 7/8	0.95 x 2.22	.335
154	250	120	± 5%	30-660	Nylon	0.183	89	3/8 x 7/8	0.95 x 2.22	.335
HIGH ENERGY CHAMBERS										
552	2.5	0.3	±10%	250-1300	Delrin	9.7	569	1 x 2 3/8	2.54 x 6.03	1.31
553	25	45	± 5%	250-1300	Delrin	1.83	583	3/4 x 1 1/8	1.91 x 2.86	1.31
621	100	75	± 5%	250-1300	Delrin	0.46	576	5/8 x 7/8	1.59 x 2.22	1.32
SKIN EQUIVALENT CHAMBER										
576	0.25	0.009	±10%	30-250	Mylar	170	7.0	1 7/8 x 4 3/4	4.76 x 12.1	.07

*These intensity limits are based on full scale reading and may be increased if readings less than one half full scale are taken. A government report entitled "Measurement of High-Intensity Gamma Radiation" indicates that VICTOREEN Chambers have given valid readings in pulsed gamma fields of as high as 5×10^4 R/sec. Copies of this report are available from VICTOREEN or office of Technical Services, U. S. Dept. of Commerce.

Digital Radiation Monitor
XETEX Models 304A and 305B

Class	Portable, handheld	
Principle of Operation	Uses halogen-quenched filtered GM tubes as detector, digital logic, LED display	
Sensitivity	β - γ	
	<u>304A</u>	<u>305B</u>
Range	0.1 mR/hr to 99.9 mR/hr overrange indicator	0.1 mR/hr to 99.9 R/h overrange indicator
Sampling-		
	304A-1 } 305B-1 }	push button
	304A-2 } 305B-2 }	continuous
Performance	Accuracy: $\pm 15\%$ referred to ^{137}Cs Energy dependence: $\pm 15\%$ from 60 keV to 1.3 MeV betas greater than 0.25 MeV Response time: nominal one second	
Requirements	Power: 305A-1 305B-2 4 AA alkaline cells, 100 hours life

Requirements (cont.)

304A-2

4 AA NiCad cells, 8 hr/charge life

305B-2

Size: 7.1 cm x 15.5 cm x 3.9 cm (2.81" x 6.09" x 1.53")

Weight: 0.33 kg (11.5 oz) w/batteries

Features

1. Autoranging
2. Six decade coverage (305B)
3. Digital display
4. Overrange indication
5. Earphone
6. Charger included for 304A-2 and 305B-2

References

Manufacturer's specifications

Cost

\$298.00 (304A-1)	\$370.00 (304A-2)
\$410.00 (305B-1)	\$498.00 (305B-2)

Address

XETEX, Inc.
660 National Ave.
Mountain View, CA 94043
(415) 964-3261

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1. Introduction

There are a variety of applications in which x-ray and γ -ray fluxes are produced at levels of concern from the point of view of radiation protection. Among these are:

- x-ray tubes, such as those in therapeutic and diagnostic medical usage, and in industrial applications;
- sealed gamma ray sources, used in medical, industrial, and research applications;
- charged particle accelerators, particularly electron accelerators;
- leakage of unintentional x rays from a klystron transmitter, color television receiver, or other high-voltage device.

For a discussion of x-ray sources, the reader is referred to Section II.D, "Sources of X-Radiation." There are great differences in the energy spectra, intensities, and temporal (pulsed) characteristics of the photon fluxes involved, and no unified treatment can be given of ways to measure radiation-hazard levels from all of the various sources mentioned. However, the properties of photon interactions with matter permit a convenient division of the measurement problem into domains of energy, which will be discussed below.

Photons traversing matter predominantly interact in three ways: (a) the photoelectric effect at the lowest energies, (b) Compton scattering at intermediate energies (a few hundred keV to a few MeV); (c) electron-positron pair production above a few MeV. In special cases, photo-induced nuclear reactions such as (γ, n) , (γ, α) , photo-excitation, or photofission may acquire importance. For more discussion on photon interactions the reader is referred to Section I.E. "Basic Radiation Detection Methods" and references mentioned therein. The relative effectiveness of each of the above processes, either for attenuation or for usefulness in measurement, depends upon target material as well as energy. Any discussion of measurement techniques must, therefore, take place in the context of the properties of photon interactions with matter. There exist a number of summary articles and reports about radiation protection and measure-

ments (Refs. 1-14), shielding design (Refs. 15,16) and the physics of photon interactions (Refs. 17,18). No comprehensive treatment of either of the last two shall be attempted here; the interested reader is referred instead to the cited works. However, an abbreviated treatment of some aspects of the problem will be introduced here where appropriate.

This section shall specifically exclude instrumentation for the following types of radiations: (a) gammas from decay of induced radioactivity (say, in beam pipes around accelerators, mentioned in Section II.A, "Particle Accelerators"), (b) photons below about 5 keV; (c) gamma spectroscopy as used for radionuclide identification, discussed in Section III.E, "Gamma Spectroscopy" and in Part IV; (d) the measurement by such techniques as film badges of individual doses received by radiation workers, discussed in "Personnel Dosimetry", Section II.G.

2. Radiation Protection Guidelines

In the general case of occupational exposure to external radiation, the relevant Maximum Permissible exposure guideline is a basic limit to absorbed dose-equivalent of 5 rem/year, with no more than 3 rem to be accumulated in any 13-week quarter and no lifetime accumulation up to age N years in excess of 5 (N-18) rem. For individual members of the general public, the relevant limits are 10% of those for occupational exposure, or a basic limit of 500 mrem/year. These matters are discussed in more detail in the introductory part of this volume. It is important to note that for photons, the dose equivalent (in rem) is numerically equal to the absorbed dose (in rad).

3. Instrumentation - General Considerations

From the point of view of radiation protection instrumentation, it is useful to divide up the photon energy spectrum into three broad regions, delineated approximately in the following way:

- (a) below 150 keV
- (b) 150 keV to 3 MeV
- (c) above 3 MeV.

Our division near 150 keV in energy is for the following reason: at energies much below 250 keV the number of different types of useful instruments is extremely limited, while at higher energies a much wider variety is available. The reason has to do with the wall thickness of various detectors, and is related to the idea of particle equilibrium, which is one of the most generally applicable concepts in radiation instrumentation and is discussed in the introductory section of this volume.

An introductory section (I.B., "Units of Radiation & Radioactivity") contains a discussion of the unit of exposure (the roentgen) and its relationship to the unit of absorbed dose (the rad) which for the case of photons is numerically equal to dose equivalent (given in rem). It was pointed out that for energies below about 3 MeV, the measurement of exposure with an air-equivalent ionization chamber is nearly equivalent to the measurement of absorbed dose with a tissue-equivalent ion chamber or almost any other device designed to measure absorbed dose. This is the energy region in which the roentgen has been defined: in which ionization in air has a meaningful relationship to ionization in tissue, and in which the ranges of the charged secondaries from photon interactions are short.

Above about 3 MeV, the pair-production cross-section becomes increasingly important, and its Z^2 dependence on the atomic number (Z) requires that any ionization detector match closely the composition of tissue. This leads to the need for tissue-equivalent instruments for measurements at higher energies. This concept has been discussed in the introductory section of this volume in I.F., "General Analytical Considerations".

Another general consideration is the difference between dose and dose rate. For many applications (such as at pulsed accelerators or medical x-ray machines) the relevant radiation occurs in extremely short bursts, from a few milliseconds down even to a fraction of a microsecond in duration. Many instruments cannot respond in such short times, and will seriously underestimate the intensity of the radiation field if used in these situations. There are two ways to circumvent this problem. The first is to alter the source appropriately: thus one can often decrease the intensity and lengthen the time of

a medical x-ray pulse until the measuring instrument can respond properly. The second method is to use a measurement technique which is intrinsically an integrating device (such as a photographic film, a TLD chip, an ion chamber in the charge mode assuming the bias is such that recombination does not occur, or other electronic instruments with an electronic band width sufficient to prevent electronic saturation operated in current-integration rather than instantaneous mode). With this in mind, it can be seen that one extremely valuable feature is the flexibility to allow optional use of such an electronic instrument in either pulse-counting or current-integration mode.

Still another general consideration is the directional sensitivity of response and the sensitive aperture of any instrument.

A standard practice should be that any commercial instrument must quote specifications for these two parameters; unfortunately, this is not always the case. The importance of the directionality of response is obvious. Fortunately, in most gamma and x-ray measurement situations the direction from which the radiation is emerging is known. The aperture-size consideration is more complex. For example, suppose an ion chamber with 10 cm² effective aperture is calibrated so that a given output indication corresponds to 1 mR/hour, in a general diffuse photon radiation field of a size much larger than 10 cm². Now suppose instead that a focussed beam of diameter only 1 cm² is being measured. If the instrument response is the same, then it is easy to see that within the beam the radiation field actually corresponds to an absorbed dose rate of 10 mR/hour, since the amount of ionization produced in the instrument is identical in the two cases.

For this reason, both the effective sensitive aperture of the instrument and the effective size of the radiation field must be understood, and the instrument aperture should be the smaller of the two.

Another comment about these instruments concerns stability of calibration. The effect of ambient temperature is one important consideration, another is humidity. The reader is referred to Section I.F., "General Analytical Considerations" for a brief

discussion of these environmental factors.

The final general comment about the electronic instruments concerns the electronic circuitry. Most ionization chambers are attached to and read out through electrometers. The electrometer tube was a standard device for many years, but it suffers from several drawbacks, perhaps the most important of which are shock sensitivity and poor stability, both long and short term. However, the output of ionization chambers may now be measured using the vibrating reed or MOSFET (metal oxide semiconductor field effect transistor) electrometer. The MOSFET's advantages are low cost, shock insensitivity, and high reliability; its disadvantages are that it has high noise and can sometimes be burned out by overloading. The vibrating reed, while expensive, has low noise and excellent resistance to overloading.

Another common electronic circuit is the count rate meter, used with G-M tubes, proportional counters, scintillators, and some ion chambers. The typical circuit connects the detector output through a discriminator, into a pulse-shaping network (usually a multi-vibrator), and then into the rate meter. Perhaps the most common source of difficulty in these instruments is the high voltage power supply, which must be well regulated because of the extreme dependence upon high voltage of detector sensitivity and of dead time (in the case of G-M tubes and proportional counters). Another problem is saturation, or jamming. At very high count rates, zero output may result; this is true especially for G-M tubes, which stop pulsing at high rates, or produce pulses too small to fire the discriminator.

Dead time is not usually an electronic problem with most commercial rate meters, except when attached to G-M tubes, which can have detector dead times of as much as 200 microseconds. In this latter use, corrections may be important at the higher counting rate levels; the user should be aware of the need for corrections, usually applied according to instructions supplied by the manufacturer.

4. X and Gamma Radiation Below About 150 keV

Three general references for discussion of instrumentation in this energy region are ICRU Report No. 17 (Ref. 2), NCRP Report No. 57 (Ref. 13) and IAEA Technical Report Series No. 150 (Ref. 19). In addition, the proceedings of two Conferences [on Color Television X-Radiation Measurement (Ref. 12) and on Radiation Safety in X-Ray Diffraction and Spectrometry (Ref. 20)] contain good summaries of work in these areas.

There are two quite different measurement situations in the low energy region: (a) the measurement of exposure levels from short, high-intensity bursts of radiation, such as from various sealed sources (such as the 17 keV x-radiation from decay of plutonium) and from unintentional x-ray leakage of color T.V. receivers, klystrons, and other high-voltage sources. In the former case, the total absorbed dose can be significant (a few mrad to a few rad) but the pulse is usually short (5 to perhaps 500 msec); the "signal" is therefore large enough to detect easily, and measurement problems are due to energy dependent response and inadequate temporal response in the instruments. In the latter case, the signal is extremely small (accurate readings of 0.1 mrad/hour may be required), but the pulsed nature of the source is less troublesome if only because of the weakness of the signal. These two areas shall be dealt with separately. If no other radiation types are present (as is the case in many medical or industrial applications) the instrument reading of exposure (roentgens) is sufficiently close to absorbed dose (rads), and therefore to dose equivalent (rem), that they may be used interchangeably for most radiation protection purposes (see Section I.B, "Units of Radiation and Radioactivity").

There are many different detection techniques which are sensitive to low-energy photons. Some of these are:

1. Ionization chambers
2. Geiger-Muller counters
3. Scintillators
4. Film and Emulsion
5. Solid state detector (thermoluminescence, flouroluminescence, induced conductivity in semiconductors).

Three critical parameters must be examined: the energy-dependence of the response, time characteristics, and the limit of sensitivity.

To gain insight into the measurement problems with air ionization chambers, consider that the exposure of 1 roentgen brings about the release of about 3×10^{-10} coulombs of charge from 1 cm³ of air. A rate of 1 mR/hour thus yields an ion current of about 10^{-16} amperes per cm³ (Ref. 21). The only way to increase the current is to increase the sensitivity volume (for a non-pressured chamber), which introduces other problems, such as variations of sensitivity in the volume. The ultimate sensitivity of an ion chamber depends on electronic circuitry capable of detecting these extremely small currents or charges.

In a cavity ionization chamber, when the majority of initial interactions occurs in the chamber wall, this produces a basic limitation to low-energy sensitivity. Chambers with walls as thin as 5 mg/cm² have been constructed (Ref. 3), in order to overcome this problem. Figure 1 (from Ref. 21) shows a typical energy response for a well-designed ion chamber, whose walls are approximately air equivalent; the fall-off at low energies is due to wall thickness and the rise above unity response is due to the fact that there is an energy region in which the effective atomic number of the walls exceeds that of air.

ICRU Report No. 17 (Ref. 2) discusses in detail the use of free air ionization chambers below 150 keV, along with the corrections needed due to attenuation and scattering of photons in air, recombination of ions, and various other effects. A standard instrument for absolute measurement of exposure (roentgens) within about 1/2% is described, as well as the relationship of exposure to absorbed dose.

Of the other possible detection means, the most common is Geiger-Muller tubes. G-M counters can detect a single ionization event, since the avalanche resulting from very great multiplication of the initial ionization is detected as a pulse. The sensitivity is about two orders of magnitude greater than that of a cavity ion chamber with comparable dimensions. However, there is a very significant energy dependence at low energies, because a single x ray at low energies will give one 'count' even though its energy

Fig. 1. Variation of response with photon energy of a typical cavity ionization chamber whose wall material is approximately air equivalent. (Reproduced from Bates, Ref. 21).

Fig. 2. Variation of response of a G-M tube with photon energy for various wall materials. (Reproduced from Sinclair, Ref. 22).

deposition is vastly less than a single photon of high energy. Figure 2 (from Ref. 22) shows the variation in response of typical G-M tubes of various wall materials.

Scintillators have also been found useful. They are the most sensitive of all detectors: a NaI(Tl) crystal of even modest size will detect individually every low-energy x ray incident upon it. Due to this fact, the detector is a significant perturbation on the radiation field, and in some situations large detectors are thus ruled out. The energy dependence of the inorganic scintillators [such as NaI(Tl)] for measuring roentgen or rad is very non-uniform, since the effective atomic number differs considerably from that of air or tissue. On the other hand, the organic plastic scintillators are composed essentially of $(CH)_n$, which is reasonably tissue-equivalent. It is the high-Z atoms (such as iodine) which make the difference here: the photoelectric cross-section is proportional to Z^5 and in high-Z materials it predominates at low energies in a way quite different from the correct tissue-equivalent behavior. The inorganic scintillator/photomultiplier combinations must be used with a spectrum analyzer if one desires a measurement of absorbed-dose, at least unless the energy spectrum is already known.

Film and emulsion are also sensitive to low energy x-radiation. (Indeed, W. K. Roentgen discovered x-radi-

ation in 1895 by the darkening of film, and film is the most commonly used technique in medical diagnostic applications). Up to at least a few roentgens of exposure, the linearity of response of film is quite good (density vs. exposure). As an example, Fig. 3 (from Ref. 23) shows this linearity for two kinds of Kodak film. The spatial resolution of film is excellent, and far exceeds the requirement of any typical radiation-protection application. Its drawbacks are two-fold: first, the processing of film must be done with great care if absolute exposure values are desired from absolute density measurements. Second, due to fading the time delay (between film exposure and processing plus analysis) sometimes makes its use inappropriate for many applications...thus, it finds its most common use in attempts to quantify intense, spatially complex x-ray sources (for example, slit leakage from the sides of tubes or voids in shielding) or in long-term, integrating dosimetry (for example, placed on a wall and developed only periodically). It is seldom used for rapid surveys if only because of the inconvenience and cost.

Although the use of solid state detectors in the low energy region is hampered by the fact that the atomic

Fig. 3. Optical density against roentgen exposure plotted on linear grid for Commercial film and for Type M films in Ready Packs. (Reproduced from Stewart, Ref. 23).

numbers of the detectors are very different from those in tissue, producing too rapid a variation of response with energy, they are used for surveys of exposure due to diagnostic radiology. To quote from ICRU Report No. 17 (Ref. 2):

"Two thermoluminescent materials, lithium fluoride and lithium borate may prove to be a value for dosimetry at low energies. The effective atomic number of the former is higher than that of soft tissue whereas that of lithium borate is lower. Attempts have been made to produce a dosimeter with tissue-like x-ray absorption characteristics by mixing them in a suitable proportion (see Kirk, et al. Ref. 24)".

In Fig. 4, from Ref. 25, the energy-dependence on TLD-700 material is shown. It can be seen that it is quite satisfactory down to perhaps 25 keV, below which its response varies quite significantly. With appropriate calibration it can be used below this energy.

As far as the application of TLD's in the x-ray domain is concerned, many of the same comments made about film apply: it is most useful as an integrated dosimeter (say, on a wall for weekly information) and is not often used for real time survey work.

Fig. 4. Energy response of TLD-700 thermoluminescent material. (Reproduced from Endres, Ref. 25).

a) Medical and Industrial X Rays

In this section we shall deal with those photons with energies below about 150 keV which occur mainly from pulsed sources. We have in mind primarily medical (and dental) x-ray machines (for therapeutic and diagnostic applications) and industrial x-ray machines, but it is not intended that this section apply only to these categories; thus, low-duty-cycle, high-voltage klystrons may emit radiation similar in quality to that considered here. Indeed, any device with high voltage is capable of being a source of x rays. We shall explicitly deal primarily with the medical x-ray problem...the other situations are implied throughout. Of course, as pointed out in the section on "Sources of X-Radiation", the medical use of x rays accounts for more than 90% of all human exposure to man-made sources of ionizing radiation.

The important properties of the radiation under consideration are the following: a) energy between about 5 and 150 keV; b) pulsed time structure; c) large intensities in each pulse (say, at least a few mR); d) the radiation is intended to be directed only at a (relatively small) target; (although the x-ray diffraction and spectroscopy applications do not commonly involve pulsed radiation, they are also treated here).

When one considers the radiation-protection aspects of x-radiation from industrial or medical usage, there are several sides to the problem:

- Proper functioning of the x-ray machine itself;
- Proper design of collimation systems, beam-stops, and shielding;
- Proper operation of the machine by the operator (proper alignment, correct exposure setting, removal of impediments from beam);
- Proper radiation-protection procedures (ensuring that accidental exposures do not occur; routine surveying of direct and stray levels; appropriate personnel monitoring and record-keeping; interlock systems; etc.)

We shall deal here only with the task of monitoring direct and stray radiation; and shall limit the treatment

by excluding both personnel monitoring of occupational exposures and patient dosimetry in medical therapy. Thus, this discussion deals only with measurements of direct and stray radiation for the purposes of a radiation protection survey. The interested reader is referred to several basic sources for guidance on shielding and interlock design and other general considerations on x-radiation (Refs. 2,15,16,26).

There are two aspects of the measurement problem: first, the measurement of possible stray radiation levels far outside of the main beam; and second, the problem of an unintended 'halo' around the intended beam.

Of the many types of instruments that are sensitive in this region, only two seem to be in common use for this type of work: ionization chambers and integrating devices (film, TLD's). The latter are used mainly in two applications: film is used for determining the exact pattern of a beam profile and halo, and both film and TLD's are used for long-term integration of levels where expected exposure rates are exceedingly small (e.g., as film badges).

For measurement of absorbed dose around x-ray machines, ionization chambers of various kinds are the predominant instrument. The parameters of interest in the choice of an instrument are four in number; first, the energy-dependence of the chamber to its exposure; second, the lower limit of sensitivity, together with the time required to obtain a reading at that limit; third, the stability of the instrument against drift; and fourth, the sensitive area of the instrument. RF sensitivity, can often be another factor [Ref. 27].

On the first point (energy-dependence) most of the Instrument Notes on ionization chambers contain energy response curves, in many cases taken from the work of Storm at Los Alamos, (Ref. 28). Remembering that medical x-radiation occurs predominantly between about 10 and 100 keV it can be seen that for many of the instruments there is a very large variation in instrument response in this region. Of those tested by Storm, only the Victoreen 440, 440RF, 444 and 666 have reasonably flat response ($\pm 30\%$, say) down to 50 keV. Since the work of Storm, many new chambers with flatter responses have been developed. Others have a serious fall-off in sensitivity in this 50-100 keV

region; still others have too much sensitivity. However, if the approximate energy spectrum is known, many of them can be used without gross error in this region. It is important to remember that no beam is really monochromatic; there is always a low-energy tail, making low-energy response a desirable feature.

The low limit of sensitivity is an important parameter mainly for measurements at a distance from the main beam, because the intensity in and near the beam is usually high enough to provide plenty of signal. However, the low duty cycle of many x-ray machines can be a problem in stray-radiation measurements or behind shielding. Many sources cannot produce a radiation pulse more often than one every several seconds, so that ionization chambers which attempt to read in dose rate (mR/hr, say) will have difficulties in this situation. The solution is an instrument which can integrate several short bursts, and read total exposure (mR). Ionization chambers exist with this feature.

The importance of knowing the sensitive area of any instrument has been discussed. For ionization chambers this can be a problem: although large volumes are required to obtain greater overall sensitivity, they are of no help at all if a narrow-cross-section beam is being monitored. Indeed, large size is a handicap if the beam is smaller than the instrument volume, since corrections are then required. For small enough beams, even rather large intensities may produce low (or null) readings, which induces a dangerous false sense of security. Further, large size prevents access to some inconvenient locations where survey information might be desired. This compromise must be borne in mind. One solution is to use another type of instrument (e.g., a small G-M tube or scintillation counter) for this kind of work.

For use in rapid searches for hot-spots, it is not necessary to have an instrument with response proportional to absorbed dose. In this application, devices such as G-M tubes, proportional counters, and scintillation counters are useful. Indeed these instruments are preferable to the ion chamber when searching for small cracks, since they can be intrinsically more sensitive by as much as two orders of magnitude and because their response times are much faster. If one then needs absorbed

dose information, an ion chamber can be called upon.

This fact was brought out in a striking way during the Conference on Radiation Safety in X-Ray Diffraction and Spectrometry (Ref. 20). We quote Dr. J. W. Thomas in the conference proceedings, discussing the use of G-M tubes for survey work around x-ray diffraction units:

"In most cases of protection surveys about diffraction equipment we really are doing a go/no-go type of survey. We are not trying to measure the radiation level in the direct beam because we really don't care what the direct beam is, only where it is. So, if you are approaching apparatus from a position where there is essentially no radiation, all of this group of instruments would start to respond at about the same point in a radiation field. This point is low enough as to read mR/hr to serve our yes/no survey needs. If the meter responds but doesn't go banging off the lower scale, the region in question is safe enough for any reasonable and necessary operations. If the meter goes banging off the most sensitive scale and perhaps the next most sensitive scale too, you really want to turn off the x-ray unit and figure out what's wrong before going on with the work. This means to me that the instruments can serve a valuable function, and they don't require highly skilled radiation measurements men to do the job. You have to remember that when the meter dial says mR/hr, that may be wrong; you may need a correction factor of 2, of 10 or 30 or more if you really want to know mR/hr. But for an awful lot of protection work you don't really care about mR/hr; you want an indication of where there may be a problem and help in locating and correcting the source of a problem. This is the very great value of these meters in operational radiation safety such as we are discussing."

One of the main goals of a radiation protection survey around a medical or industrial set-up is to insure the radiation safety of the system as routinely operated by a technician. The problem of the integrity of shielding is particularly difficult. In many installations, workers in adjacent rooms may be exposed to radiation far above

permissible level, because of poor design or improper maintenance of shielding. It is essential, as standard operating procedure, to perform an adequate test of shielding and maintain its integrity against the inroads of convenience or expediency. One of the most common failings of shielding design (Ref. 29) is that the shielding is only over-designed in the region where the main beam is intended to be stopped; other areas are shielded by the minimum thicknesses required for small stray radiation levels. The beam set-up may then change, or a leak may develop in the housing, and the shielding is no longer sufficient. The New York City Department of Health now recommends shielding tests by directing the main x-ray beam at areas which ordinarily cannot be expected to receive it; shielding improvements are then suggested until the levels are small even under those circumstances (Ref. 28).

The detection of pin-holes and cracks in shielding may be quite difficult. Since many industrial and medical x-ray machines cannot put out bursts at high repetition rates, the task of scanning for cracks is time-consuming beyond the patience of many surveys. The recommended practice here (Ref. 28) is the use of a strong radioactive x-ray source (such as Technetium 99m with 140 keV x rays, 6 hour half-life), placed in the correct location. A survey with any sensitive rate meter (e.g., a G-M or proportional counter) will detect the presence of leaks without any trouble...in this case, one wants to fix the leak, not quantify it.

As already mentioned another problem of radiation protection is a possible halo around the main beam. This halo can serve as a secondary source itself if it strikes objects along the path. The way to measure this type of effect is to quantify the actual main beam size with exposure of film, and then to search near the beam edges with a sensitive electronic instrument of small sensitive area. An important problem occurs when beam halo from a medical application strikes a patient outside of the intended area of exposure. The seriousness of this problem cannot be over-emphasized: greater than 90% of all exposure to man-made radiation comes from medical x-radiation; yet it is estimated that dramatic decreases in total exposure can easily be made by proper collimation of those machines which still do not possess adequate collimation systems (Ref. 30).

See discussion of Section II.D,
"Sources of X-Radiation."

In this regard, the U. S. Bureau of Radiological Health has published requirements for x-ray equipment manufacturers, governing machine operation at time of sale (Ref. 31). One of the BRH's functions is in fact to conduct an electronic product control program to protect the public health and safety. They have as a part of this role issued performance standards for ionizing radiation emitting products, specifically for cold-cathode gas discharge tubes, diagnostic x-ray systems and their major components, radiographic equipment, fluoroscopic equipment, and cabinet x-ray systems (Ref. 32).

Another function BRH has performed is to insure that adequate instrumentation exists for determining compliance with the Federal Performance Standard for Diagnostic X-ray Source Assembly and Their Major Components. BRH performed a preliminary design of the needed instrument and contracted for it to undergo final design and packaging in the commercial sector (Ref. 33). The instrument represents the culmination of work on the development of charge digitizers, a form of electrometer which is inherently digital. By exploiting the speed of response of such devices, timing information as well as radiation information is derived from ion chamber currents. This accomplishes additional testing with a single instrument and eliminates the need for a separate instrument to evaluate timer performances.

b) Continuous, Low-Level Leakage

In this section, emissions from gamma sources, color TV receivers, klystrons, and other low level, low energy (< 150 keV) x-radiation sources will be considered. The industrial application of x-ray diffraction is also treated here, although the intensity levels are in this case very high.

Emissions of this kind generally have a known energy spectrum: for isotope sources this is certainly true, and for leakage radiation at least the maximum energy is known, since x-radiation significantly above the accelerating voltage does not occur. This simplifies the measurement problem enormously. As an example of the detailed problems of measurements from

leakage sources, the particular case of color TV receivers will be treated. This choice is due primarily to the large amount of attention which the color TV problem has received, and to the existence of the TV X-Radiation Measurement Conference referred to previously (Ref. 12). More stringent regulation and increasing use of solid state circuitry have reduced the x-ray emission from recently built color TVs to negligible levels under conditions of normal operations and proper servicing (Ref. 34, 32, 35). The requirements for monitoring leakage from color televisions is similar to that for monitoring leakage from other sources. Most x-radiation emitted from the active elements within a typical color TV set is limited to the energy region up to 45-50 keV, and in the past it tended to emerge into the open air through narrow cracks and slits (Refs. 36, 37). This is no longer so, now it is the face of the cathode ray tube (CRT) (Ref. 38). The levels which must be detected here are very low: the recommended NCRP maximum emission level and BRH performance standard is 0.5 mR/hr over a 10 cm^2 area, at a distance of 5 cm from the receiver surface (Refs. 32, 39, 40).

Figure 5, from Ref. 37, shows the energy spectra of the three important x-ray sources (picture tube, rectifier, and from a typical 25" rectangular color TV receiver (manufacturer undisclosed), operating at 25 kV. It can be seen that the x rays are peaked at about the energy of the tube, with a secondary peak at about 12 keV from the shunt regulator, which has not been used for a decade. Figure 6 (from Ref. 37) shows the spatial distribution from a typical receiver when de-tuned and adjusted so that the radiation level is exactly at the 0.5 mR/hr maximum recommended by NCRP.

In order to measure this type of radiation, several specific instruments have been designed, which also have usefulness for other x-radiation leakage monitoring. One of the earliest, built in 1949, is a flat, cylindrical ion chamber run either in condenser mode or attached to an electrometer. It has 0.1 mm windows of nylon, coated on the inside with aquadag to make them conducting. This instrument has been described by Braestrup (Ref. 41). Its principal disadvantage is that the measurement is time-consuming (several minutes for 0.1 mR/hr levels) and represents an average over a large (50

Fig. 5. Typical energy distribution TV receiver HV at 25 KV. (Reproduced from Wang, Ref. 37).

Fig. 6. Spatial distribution from a typical 25" color receiver (full raster) when adjusted to 500 R/hr level at 5 cm from tube front. (Reproduced from Wang, Ref. 37).

cm²) area. This latter problem is very serious when searching for narrow-beam leaks, as already pointed out.

In 1968, the U.S.P.H.S. National Center for Radiological Health (now the Bureau of Radiological Health) designed a TV-set x-ray Survey Meter (Ref. 38) using six aluminum 1B85 G-M tubes connected so that the meter circuit responds only to the signal from that G-M tube with the highest momentary output. The most sensitive scale responds to a product of exposure rate and detector area of 0.5 mR/hr over 13 cm²; more intense exposure levels over smaller areas are integrated over the 13 cm² effective area. Its energy dependence was found to be acceptable down to about 20 keV (Ref. 42); in Fig. 7 the energy dependence is shown explicitly compared to several other instruments.

Another Bureau of Radiological Health study (Ref. 23) examined the relative merits of film (Kodak Royal Blue in Ready Pack), an end-window-G-M

Fig. 7. Response vs. voltage of tube for four detectors, normalized to 1 mR/hour as read on a Victoreen 440RF ionization chamber. (Reproduced from Savic, Ref. 44).

tube (Eberline RM-3A), and an ion chamber (Victoreen 440RF), when used to test leakage from an ensemble of about 100 misaligned Type GEF4 shunt voltage regulator tubes. The size of the leakage spot varied from a few mm² to a few cm², and the effective energy was 10 to 16 keV. It was found that, using the film as the standard, the two electronic instruments were both unable to distinguish the intense spot-like characteristic of the leakage beam from a diffuse source of lower intensity over a bigger area. The average "area correction factor" for the two cases was a factor of 18.5 for the G-M tube and 30 for the ionization chamber. The conclusion of the study was that the data "suggest a need for additional investigation and development of instrumentation for surveying small-area x-ray beams which are also very intense and soft" (Ref. 23).

The following summary is taken directly from Ref. 23:

"Based on the work conducted at the X-ray Exposure Control Laboratory, several statements can be made and verified by experimental data concerning the calibration of radiation detection devices used in measuring x-ray emissions from color television receiver components.

"Film has been found to be the best detector for permanently recording and measuring in detail the area distribution of x-ray exposures, especially for small x-ray beams such as found from the components investigated. Calibration of film for each quality tested is necessary for accurate results, but the quality-dependence over the range 20 to 30 kv, 0.1 to 0.5 mm Al HVL, is at most 30 percent, so only the extremes need be calibrated and an average conversion curve used for the purpose of surveying high voltage shunt regulator tubes and the like.

"End-window G-M tubes are useful in probing for narrow x-ray beams in that they can locate rapidly with relative ease the x rays emitted from television components and from sets. A drawback, however, to the ordinary large-aperture end-window probe is that only a small fraction (5 percent on the average for our data) of the actual intensity

will be read when surveying narrow x-ray beams.

"Ion chamber survey meters are considered the best for accurate exposure measurements of broad beams that uniformly irradiate the entire volume of the ion chamber and are of high enough exposure rate to give statistically valid readings. In narrow beam use the resulting readings were too low by an average factor of 30."

This conclusion is a fundamental one for usage in any leakage application: for narrow beam, high intensity studies, an electronic instrument with as small as possible a sensitive volume should be used. These instruments have recently become available.

Several other techniques have been proposed and the use of a small scintillator shows promise for measuring slit-like sources.

The use of a NaI(Tl) scintillation detector, in the Eberline Model PRM-5 with PG-1 probe, has been described by Rechen and Williams (Ref. 43). The PRM-5 unit houses a battery-operated high voltage supply for the phototube, a double pulse-height discriminator, and a count rate meter. The discriminator must be adjusted to a pulse height 'window' of interest. The detector, a thin (1 mm) chip of NaI of area 8 cm², has a >80% efficiency from 10 to 70 keV. The sensitivity is adequate: 0.5 mR/hr produces 110,000 cpm. It thus has extremely large dynamic range, making it "very useful for plotting approximate iso-exposure rates around television receivers and other electronic products" (Ref. 43). The instrument is easily calibrated, (Ref. 43), and convenient to use. However, the small detector area makes a survey for minute cracks difficult, and it should therefore be used in conjunction with a larger-area detector. Reference 43 contains a complete analysis of this instrument's performance, and should be consulted for details.

For the problem of sources distributed broadly in space, a very useful study has been made at Zenith Radio Corporation by Savic et al (Ref. 44), who tested the following five instruments:

Victoreen 440RF shielded ionization chamber;

SPECTRAL DEPENDENCE - 239A and 208A MINOMETER CHAMBERS

Fig. 8. Energy response of Victoreen 239A and 208A ionization chambers. (Reproduced from Bellian, Ref. 45).

Victoreen Thyac III GM Counter with
a 489-35 α, β, γ probe;

Victoreen 1B85 bare GM tube (no
longer available from manufac-
turer).

Victoreen Scintillator " γ probe"
which was used with the Thyac III
count rate meter.

Eberline Model PRM-5 pulse rate
meter having single channel analy-
zer capability in conjunction with
a sodium iodide (NaI:Tl) crystal
Plutonium Gamma Probe (Model PG 1).

Although these instruments may not
be the instruments of best choice today
or may even be superseded, a discussion
of their characteristics serves to il-
lustrate basic concepts in such instru-
mentation. In this study (Ref. 44) the
Victoreen 440 RF ionization chamber was
used as a standard, and Fig. 7 shows
the energy dependence of the other four
instruments compared to it. Here the
rate of emission is 1 mR/hr of approxi-
mately monochromatic radiation with the
energy on the abscissa, and the data
are taken 18" from the front of a color
TV picture tube. It can be seen that
all except the Thyac III (with gamma
probe) can be used in this application
with only minor corrections required
for energy dependence in the 25-35 keV

energy region. It should be noted that
the Victoreen 1B85 G-M tube is the de-
tector used in the NCRH (BRH) instru-
ment described above (Ref. 42).

Throughout the literature on this
subject, the Victoreen 440RF and 440RFC
ionization chamber is sometimes con-
sidered to be the reference instrument.
Victoreen's 208A/239A Minometer chamber
has also been a common reference labora-
tory standard. The responses of these
two instruments are shown in Figs. 8
and 9, taken from Victoreen's own data
(Ref. 45).

Fig. 9. Energy response of Victoreen 440
RF ionization chamber. (Reproduced
from Bellian, Ref. 45).

To summarize the section on leakage radiation from color TV receivers, klystrons, and other similar sources, it can be seen that while instrumentation does exist for the measurement task, it has difficulties which must be understood by the user. For measurements of broad-area, low-level radiation the ionization chamber technique is certainly the most reliable; of course, chambers with slowly-varying energy dependence at low energies must be used. The use of G-M counters is recommended only when the energy spectrum is understood. For studies of slit-like, high-intensity leakage the scintillation/photomultiplier technique is appropriate, because its sensitive area can be made small. Recently ionization chambers with small volume (down to .2 cm³) have become available and can be used also. Alternatively, an instrument like that described in Ref. 42 (with 6 G-M tubes, connected so that only the one with the greatest instantaneous count rate is read out) can be useful. The use of TLD's and film or emulsion is very satisfactory, but only in those situations where an instantaneous answer on exposure levels is not required.

5. X and Gamma Radiation from About 150 keV to Several MeV

Photons in this energy region arise principally from two types of sources, particle accelerators and radionuclide sources. While only a very few radionuclide sources produce gamma rays at energies above 2 MeV, particle accelerators can yield photons of much higher energies; these last will be discussed separately ("high-energy gamma rays"). An important distinction is that whereas accelerator photons often have an unknown energy spectrum, and can be accompanied by other species of particles (principally neutrons), neither of these problems is present for most applications with radionuclide sources.

The basic references on radiation protection in this energy region are NCRP Reports 33, 35, 36, 40, and 49, (Refs. 15, 9, 10, 8, 16) ICRU Report 14 (Ref. 1), and especially NCRP Report 25 (Ref. 11). These reports discuss fundamental considerations (such as shielding, handling, protection instrumentation) and should be incorporated into the basic operating philosophy of any worker in this field.

There is a wide range of instrumentation available for measurements in this energy region. However, the instrument probably most suited to hand-held survey measurements (and most commonly used) is the cavity ionization chamber, in one of several designs. The main reason for this is that if absorbed dose is desired the other kinds of common instruments (Geiger-Muller counters, gas proportional counters, scintillation detectors) have serious energy-dependent sensitivity in this region.

Also, ionization chambers with rather small sensitivity to neutrons can be constructed. The use of TLD's, film, and emulsion for these purposes is discussed elsewhere in this volume (see Section III.F, "Personnel Dosimetry").

As mentioned, this is the energy region in which air and tissue are nearly equivalent in terms of ionization (ergs per gram) deposited by a given incident photon. Thus, the use of air equivalent, tissue-equivalent, and a variety of other materials has been common, and all work fairly well. However, sensitivity to neutrons varies from one material to another; in particular, the response of hydrogenous materials to neutrons is often more than an order-of-magnitude greater than that of non-hydrogenous materials. This latter fact makes possible the independent measurement of gamma and neutron absorbed doses by the comparison of response to two differently-constructed chambers.

To illustrate this, consider the comparison (Ref. 11) between a tissue-equivalent chamber and a chamber with conducting-teflon lining and CO₂ gas. Exposure of each of these chambers to 1 R of gamma radiation at about 1 MeV produced a reading of 0.97 rad. However, 8 to 10 rem were required to produce a reading of 1 R in the tissue-equivalent chamber; its response was nearly rad-proportional, since 8 to 10 rem is equal to 1 rad for fast neutrons. For the teflon-CO₂ chamber, 30 to 60 rem (depending on the energy) were required for response of 1 R, and below about 100 keV, the neutron response was completely negligible.

In this energy region most of the comments made in the previous section about lower energies still apply. One basic difference is due to the properties of the photons at these higher energies: they penetrate mater-

ial (especially low-Z) to significant depths, and Compton scattering (γ -electron scattering) overtakes the photoelectric effect as the dominant interaction process. An additional complication for radiation protection is that these gammas can backscatter, since the Compton process has such a wide angular distribution (Refs. 17,18). Thus, the energy spectrum reaching a detector from a sealed gamma source can contain components from many directions besides the source itself, and with the wide energy spectrum characteristic of the Compton process in the continuum. The instrumentation required for the measurement of such a spectrum needs a wide, flat energy response...the same comments apply as for the lower energies.

Fortunately, at these higher energies the wall thickness of the cavity detectors becomes less critical. Also, the uniformity of energy response is better for G-M and proportional counters, especially above about 1 MeV, so that these instruments become useful for rough absorbed-dose measurements.

6. High Energy Gamma Rays

Photons with energies above a few (say 10) MeV almost never occur except at high-energy accelerators (see Section II.A, "Particle Accelerators"). Even in these installations, the radiation fields outside of the shielding walls tend to be dominated by neutrons (Refs. 46, 47). Even when high-energy components are present, ionization chambers are the most commonly used instrument; this despite the fact that their response is not proportional to absorbed dose beyond a few MeV. The reason is as follows:

Consider the situation where very high-energy (even GeV-level) electromagnetic showers are developing in the shielding. The primary energy is successively degraded into more and more particles of lower and lower energy; eventually the process reaches a peak (with regard to the number of particles in the shower). The absorption processes of photons in shielding are such that most low-energy photons are absorbed in rather short distances after their production; above about 5 or 10 MeV, however, the absorption coefficients are both small and relatively constant. When one of these higher-energy photons does interact, however, energy is then lost quite quickly due

to production of lower energy secondaries of shorter range. Thus, once equilibrium is reached, the fall-off of intensity with increasing penetration is slow, and governed by the attenuation probabilities near the minimum in the absorption curve, despite the fact that at any point the dominant particle flux (and absorbed dose) are from low-energy particles.

This concept is crucial to an understanding of measurement of fluxes such as those from high-energy photons. This is why ionization chambers, whose sensitivity is poor above a few MeV, are nevertheless adequate instruments for absorbed doses in equilibrium photon radiation fields.

If there is a suspicion that non-equilibrium high-energy components are present, the best check is to add absorber around the ion chamber in stages, up to about 30g/cm². In equilibrium fields, there should be a small, monotonic decrease in instrument response, while in non-equilibrium situations an actual build-up of response may occur as mass is added.

7. Instrument Calibrations

Many of the general problems with the calibration of instruments have been discussed in the introductory part (Section I.F., "General Analytical Considerations") of this volume. Here some of the considerations specific to x and gamma-ray instruments will be given a more detailed treatment.

There are two common sources of photons for calibration purposes: x-ray tubes and radioactive gamma emitters. Since the response of many instruments is a strong function of energy, particularly at the lower energies, there is a high premium on the use of a source as mono-energetic as possible.

In the case of radioactive sources, the intrinsically mono-energetic gamma energy can be degraded either by scattering or by absorption (in the source itself, its container, or the air). Below a few tens of keV absorption is dominant, and care should be taken to account for it in the calibration procedure. This is particularly important because one normally wants the detector to be far enough away that the finite instrument dimensions (depth and area) are not the major uncertainty in

the measurement. Since knowing the effective 'center' of a detector is often difficult to better than perhaps one third of the relevant dimension, one usually needs to locate the detector several detector-diameters away from a source to reduce the $1/r^2$ uncertainties to 10%. Deviations from inverse-square fall-off of flux can be studied with a detector whose size is much smaller than the distances involved.

Scattering is very important in the MeV range, because of the broad angular distribution of the Compton scattering cross-section (Ref. 18). This makes it desirable to locate the calibration set-up as far as practical from large scattering surfaces such as walls.

Beams from x-ray tubes are useful mainly in the lower-energy region. The main problem here is the energy spectrum of the source, which usually has a broad spread even as it emerges from the tube. The low-energy tail on an x-ray spectrum can be greatly reduced in magnitude by the use of absorbers. To quote from ICRU Report No. 20 (Ref. 3):

"Care should be taken that secondary peaks are not introduced from the radiation produced by fluorescence of the filter material. For example, with a lead filter it may prove necessary to follow the lead with a tin filter to absorb the K radiation from the lead. Likewise, a copper filter may be required to absorb the K radiation from tin; an aluminum filter may be used to

absorb the K radiation from copper. Normally high purity materials should be used as absorbers.For measurements of energy dependence below 100 keV, the K-fluorescence radiation of different elements can be used as an almost monoenergetic source of radiation (Refs. 48, 49). For most radiation protection applications the better energy definition obtained by this method is not required."

Table 1 (from Ref. 3) shows some common gamma-ray sources. Note that sources are available even in the low-energy region where x-ray tubes are also used. The last column, taken from Nachtrigall (Ref. 50), shows the specific gamma-ray constant in units of roentgen-meter²/ curie-hour, which specifies numerically the exposure rate (R/hr) at 1 meter distance from 1 curie of activity. However, these values may not be accurate enough for calibration purposes in the presence of scattering and absorption problems.

Two geometrical effects are particularly important: first, the angle of incidence of the radiation can affect the sensitivity of the detector, and the effect can be dependent on energy. Second, backscatter from a human body can significantly affect the response of an instrument held in the hand or worn on the body. This can be checked by using a phantom positioned appropriately.

Table 1. Some gamma-ray calibration sources

Radionuclide	Effective Photon Energy (keV)	Half-life (Physical)	Specific Gamma-Ray Constant SI Units (R m ² /Ci hr)
Am-241	60	458 years	0.016
In-114m	192	50 days	0.043
Hg-203	279	47 days	0.12
Au-198	412	2.7 days	0.23
Cs-137	662	29.9 years	0.32
Co-60	1170, 1330	5.23 years	1.30
Ra-226*	- -	1608 years	0.825
Na-24	1370, 2750	15 hours	1.83

*0.5 mm Pt filtration, decay products included.

The effect of humidity, temperature, and pressure have been discussed above and in the introductory material to this volume. Here we note again that the ambient conditions during calibration should be included with the calibration information.

A detailed discussion of absolute national standards for instrument calibration is beyond the scope of this work. The National Bureau of Standards has for some years performed calibrations of x and gamma-ray instruments against its own national standard (Ref. 41). However, the need for a larger number of calibration facilities has prompted the American Association of Physicists in Medicine (AAPM) to establish two certification centers. The primary motivation for these centers is for purposes of precise dosimetry in radiation therapy, but their use for radiation protection instrumentation will also be important. The centers, with secondary standards calibrated against the NBS national standard, would perform routine calibrations of instruments for users. Information about the effort is available from the AAPM office (Ref. 52). Two regional centers have been certified: M.D. Anderson Hospital in Houston and Memorial Hospital in New York City.

8. Summary

Monitoring of x and gamma-radiation has three distinct classes: (a) surveys where the presence of ionizing radiation is generally not intended (e.g., klystrons and TV receivers); (b) industrial and medical x-radiation, usually at energies of ~150 keV; and (c) higher-energy photons, usually from accelerators or sealed radioactive sources.

Two important guidelines should be kept in mind. First, it is essential that detailed radiation protection surveys be performed whenever an installation begins operation or makes substantial changes in operating conditions. This survey should identify and quantify the important sources of radiation, thus leading the way to reducing unnecessary exposures wherever possible. Second, routine surveillance is essential for maintaining the integrity of any operation.

The routine surveillance should have two goals: The quantification of absorbed-dose (and dose-equivalent) levels, and the determination of possible changes in operating conditions which might demand remedial action (e.g., the search for cracks in shielding). Instruments suitable for each of these tasks have been described in the text.

The ionization chamber is the basic survey instrument for x and gamma-ray absorbed dose measurements; the properties of and applicability of various ionization chambers have been discussed. For survey use where absorbed dose is not required, a variety of other instruments is available: Geiger-Muller tubes, gas proportional counters, scintillation detectors and in some cases film and thermoluminescent detectors are appropriate. We have indicated guidelines to aid in the selection of the class of instrument which will best perform a particular function. Also, we have attempted to give a brief discussion of some instruments and techniques which are not generally commercially available, but are worthy of widespread availability. The reader is referred to the original references for more details.

Finally, a comment about the electronics circuit design is necessary. Still some monitoring instruments have not as yet incorporated the most recent advances in electronic technology. Properly exploited, these advances lead to many improvements: An increase in stability and reliability; a reduction in frequency of maintenance; a decrease in size, weight and cost; and a better means of data read-out.

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D. INSTRUMENT NOTES: GAMMA

Included in the following pages are Instrument Notes for those instruments useful for gamma and x-ray monitoring. The reader should be aware of the following facts, in order to find all Instrument Notes appropriate for gamma detection and measurement:

- 1) Instruments suitable for gamma spectrometry are also included here.
- 2) The filing system lists Instrument Notes alphabetically by manufacturer.
- 3) In the filing system, no distinction has been made between gamma, gamma/x-ray, and x-ray instruments. The choice of mnemonic (GAM, GAM-XRA, or XRA) has usually been made based upon the manufacturer's advertised specifications.
- 4) Many gamma instruments are also useful for beta radiation, and some for alpha radiation as well. If a manufacturer claims beta/gamma sensitivity, the instrument has been included in the "Beta" section with mnemonic RAD-BET, GAM. If alpha/beta/gamma sensitivity is claimed, the instrument has been included in the "Combination Instruments" section with mnemonic RAD-ALP, BET, GAM. The reader should therefore consult those other sections together with this section for a fuller overview of instruments of this type.

Survey Monitor
Baird Model 904-121

Class	Portable, handheld
Principle of Operation	GM detector, amplifier, meter readout
Sensitivity and Range	Range: 1-100 mR/hr 500-50,000 cpm
Performance	Accuracy: $\pm 10\%$ full scale
Requirements	Power: Five batteries Size: 19.1cmL x10.2cmW x17.8cmD (7.5"L x4"Wx 7"D) Weight:2.7 kg (6 lbs)
Features	Includes radium source for calibration Model 904-122 detects alpha also
References	Manufacturer's Specifications
Cost	\$461.00
Address	Baird Corporation 125 Middlesex Turnpike Bedford, MA 01730 (617) 276-6000

Shielded Well Scintillation Counter

Baird Model 988-10C

Class	Laboratory
Principle of Operation	Well-type NaI (Tl) scintillation detector system 1 3/4"Dx2"thick NaI (Tl) crystal. Well size: 21/32"ID x1-35/64"D
Sensitivity and Range	Efficiency: 35% for Iodine-131 Background: 250 cpm; 20 cpm when used with spectrometer
Sampling	Batch
Performance	Energy dependence: Accuracy: Operating temperature: meets minimum requirements
Requirements	Power: 110VAC, 60 Hz (220 VAC, 50 Hz optimal) Size: 45.7cmH x32.4cmD (18"H x12 3/4"D) Weight: 131.5 kg (290 lbs)
Features	Lead shield, tube mount, Model 810LX does not include detector
References	Manufacturer's Specifications
Cost	\$2350.00
Address	Baird Corporation 125 Middlesex Turnpike Bedford, MA 01730 (617) 276-6000

Gamma Counter

Beckman Gamma 4000 Counting System

Class	Laboratory, automated sample counter, printer		
Principle of Operation	Utilizes an inorganic scintillating material, sodium iodide crystal, as a gamma ray detector. A multiplier phototube collects the impulses generated which are then amplified, scaled, and presented as counts and time.		
Sensitivity and Range	Range: 0-2 MeV		
		<u>2 Inch Crystal</u>	<u>3 Inch Crystal</u>
	Background:	0-2 MeV < 400	< 250
		0-80 KeV(¹²⁵ I Window) < 50	< 50
	Maximum counts:	1 x 10 ⁶ counts	
	Maximum count rate:	6 x 10 ⁵ cpm	
Sampling	Batch sampling, 200 or 400 sample capacity		
Performance	Operating temperature range: 15°C to 40°C, 95% humidity		
	Stability: Less than 3% channel drift in any 8 hr. period for operating temperatures held within 5°C		
	Sample changer time:	Samples in same tray - 9 seconds	
		Sames in adjacent tray - 11 seconds	
		<u>2 Inch Crystal</u>	<u>3 Inch Crystal</u>
	Efficiency (in wide open window)		
	¹³⁷ Cs	≥ 30%	≥ 40%
	¹²⁵ I	≥ 72%	≥ 74%
	Resolution: ¹³⁷ Cs	< 14%	< 12%
	Volume Independence (column height)	2.2cm	3.2cm

INSTRUMENTATION
FOR ENVIRONMENTAL
MONITORING

RAD-GAM, XRA
Laboratory
Beckman
Page 2

Requirements	Power: >2 amps at 120 volts ($\pm 10\%$), 60 Hz. Size: 32 1/2"Hx22"Wx25"D (or 34 1/2"D) Weight: 150 lbs.
Features	<ol style="list-style-type: none">1. Ambient temperature, tray sampling system2. One or two channels of analysis3. 2 inch or 3 inch overhead crystal detector4. Accommodates up to 16mm x 87mm tubes5. Plug-in discriminator modules (preset or variable)6. Single gain calibration7. Output in counts or % bound8. Many accessories available.
References	Manufacturer's Specifications
Cost	Gamma 4000 \$10,900 - \$16,200
Address	Beckman Instruments, Inc. Jamboree at Campus Drive P.O. Box C-19600 Irvine, CA 92713 (714) 833-0751

Gamma Counter

Beckman Gamma 7000, 8000, 9000 Counting Systems

Class	Laboratory, automated sample counter, printer
Principles of Operation	Utilizes as inorganic scintillating material, a sodium iodide crystal, as a gamma ray detector. A multiplier phototube collects the impulses generated which are then amplified, scaled and presented as counts and time.
Sensitivity and Range	Energy Ranges (selectable): 0.5 MeV, 1 MeV, 2 MeV Background 2 MeV Energy Range: Wide open window: < 200 cpm 125I window < 50 cpm Maximum count rates: 125I - 1.0×10^6 cpm 137Cs - 6.0×10^6 cpm Maximum counts: 5.0×10^6
Sampling	Batch sampling, 300 sample capacity
Performance	Operating temperature range: 15°C - 40°C, 95% humidity Stability: $\pm 3\%$ in 8 hours for temperature variation not exceeding 5°C Sample changer time: < 15 sec. Efficiency: 125I > 70% wide open window 0.5 MeV energy range 137Cs > 38% wide open window 1 or 2 MeV energy range Resolution: < 15% at 137Cs photopeak (0.663 MeV) Volume Independence: Precipitate to 30 ml (in 25mmx60mm vial) Power Failure Memory Protect: Minimum 3 hours
Requirements	Power: 8 amps at 120V $\pm 10\%$ A.C., 60 Hz Size: 47"Wx30"Dx60"H Weight: 1750 lbs.

- Features
1. Two or three channels of analysis
 2. Three inch crystal detector
 3. Ambient temperature
 4. Microprocessor controlled
 5. Multi-user capability by use of library programs called up by command towers
 6. ^{125}I efficiency calculation program
 7. Energy spectrum scan program
 8. Automatic calibration
 9. Output in counts, cpm or % bound single or dual label
 10. Computer compatability and many accessories available

References Manufacturer's Specifications

Cost

Gamma 7000:	\$19,300 - \$23,000
Gamma 8000:	20,900 - 24,900
Gamma 9000:	32,300 - 33,800

Address

Beckman Instruments, Inc.
Jamboree at Campus Drive
P.O. Box C-19600
Irvine, CA 92713
(714) 833-0751

Germanium Gamma Spectrometry Detectors

Canberra Models 7229P/7229

CLOSED-END COAXIAL DETECTOR

Class

Laboratory

Principle of
Operation

Pure Ge and Ge(Li) semiconductor detectors used for gamma spectrometry.

Specifications

	Model 7229P	Model 7229
Type	Pure Ge	Ge(Li)
Shape	Closed End Coaxial	Closed End Coaxial
Active Area (cm ²)	Up to 28	Up to 28
Volume (cm ³)	10 to 150	10 to 150
Drifted Depth (mm) (Depletion Depth)	Up to 25	Up to 25
Resolution, FWHM (keV) for 1.33 MeV (eV) for 122 keV (eV) for 5.9 keV	1.7 to 2.3 475 to 1300 145 to 370	1.7 to 2.1
Efficiency*	Up to 30%	Up to 30%
Peak-to-Compton for 1.33 MeV	Up to 50:1	Up to 50:1

*Efficiency measured relative to 3" x 3" NaI(Tl), at 1.33 MeV and 25" distance.

	<hr/>	
	Price Range	\$8,000 - \$35,000 \$5,000 - \$25,000
<hr/>		
Features	1) Cryostats and associated electronics also available. 2) Model 1427 electronic unit available for fast-timing applications.	
References	Manufacturer's specifications	
Address	Canberra Industries 45 Gracey Avenue Meriden, CT 06450 (203) 238-2351

NaI(Tl) Gamma Spectrometry Systems

Canberra Model 802

Class

Laboratory

Principle of Operation

NaI(Tl) crystal, photomultiplier tube, and tube base voltage divider assembled into a unit for gamma spectrometry. *

Specifications

Designation	Crystal Diameter		Crystal Thickness		Tube Size		Price for Assembly
	(cm)	(in.)	(cm)	(in.)	(cm)	(in.)	
802-1	3.8	1-1/2	2.5	1	5.1	2	\$ 295
802-3	5.1	2	5.1	2	5.1	2	395
A802-3W (well detector)	5.1 (1.59)	2 (.625)	5.1 (3.65)	2 (1.438)	5.1	2	450
802-4	7.6	3	7.6	3	5.1	2	1060
B802-4W (well detector)	7.6 (1.67)*	3 (.655)	7.6 (5.24)	3 (2.063)	5.1	2	1180

*Size of well in the crystal

Features

- 1) Entrance window 0.019" aluminum
- 2) Aluminum housing
- 3) Auxiliary electronics also available

References

Manufacturer's specifications

Address

Canberra Industries
45 Gracey Avenue
Meriden, CT 06450
(203) 238-2351

X-Ray Detectors
Canberra Industries

Class

Laboratory

Principle of
Operation

Si(Li) detectors used for x-ray spectroscopy

Model	7313	7333	7383
Area(mm ²)	10	30	80
Depth(mm)	3	3	3
Resolution* at 5.9 keV FWHM	160 eV	to 170 eV	185 eV
Price Range**	\$4800	to \$7000	

*Guaranteed worst case resolution @5.9 keV @1000 counts/sec.
Typical performance exceeds specifications. ** Includes
Si(Li) detector, cryostat with choice of 1.0, 0.5, 0.3 mil Be
window, Model 1708 pulsed optical feedback preamp.

Features

Liquid nitrogen cryostat, preamplifier, spectrum enhancer,
system alarm, LN monitor, and bias supply also available

References

Manufacturer's specifications

Address

Canberra Industries
45 Gracey Avenue
Meriden, CT 06450
(203) 238-2451

X-Ray Scintillation Detectors

Canberra Models 1701 and 1702

Principle of Operation

X-ray scintillation detector comprising a NaI(Tl) crystal and photomultiplier tube and a voltage divider network

Specifications

Model 1701

Window	0.005 in. Beryllium
Transmission	50% at 3.5 keV 88% at 6.0 keV
Resolution	55% or better at 6 keV
Collimator	Removable brass with 0.125" x 0.5 in. slit
Crystal	NaI(Tl) 1" in dia., 1 mm thick
Photomultiplier	Amperex XP 1010, selected

Model 1702 with preamplifier

Type	Charge sensitive, FET input
Power	24 V
Output	0 to 20 V, 40 nsec risetime 60 sec fall time, 50 ohms impedance

References

Manufacturer's specifications

Cost

\$695.00 (1701) \$1,100.00 (1702)

Address

Canberra Industries
45 Gracey Avenue
Meriden, CT 06450
(203) 238-2351

Pure Germanium Planar Detectors

Class	Laboratory
Principle of Operation	Pure Germanium Planar Detectors used for low energy photon spectrometry.
Specifications	Active Area (mm ²) 25 - 2000
	Diameter (mm) Up to 50
	Depth (mm) 5 to 20
	Resolution, FWHM
	eV for 122 keV 450 - 1000
Features	eV for 5.9 keV 150 - 900
	Cryostats and associated electronics also available.
References	Manufacturer's specifications
Cost	Price Range: \$ 7,500 - \$18,000
Address	Canberra Industries 45 Gracey Avenue Meriden, CT 06450 (203) 238-2351

Radioisotope Calibrator

Class	Laboratory calibration for gamma-emitting radioisotopes	
Principle of Operation	Pressurized argon well-type ionization chamber and digital electrometer system.	
Sensitivity and Range	CRC-5/10	CRC-30
Function	Calibration	Calibrator/Computer
Energy Range	25 keV to 3 MeV	
Sensitivity Range	To 6 Ci maximum	
Resolution	0.1 μ Ci	
Performance	<p>Each calibrates over 200 radionuclides. For CRC-30, computer memory can accommodate up to 19 samples and concentration recall displays decay adjusted concentrations for push-button isotopes with a ticket printed patient dose hard copy record. Accuracy: $\pm 3\%$ between calculated and observed values</p> <p>Operating temperature: 50° F to 85° F</p>	
Requirements	<u>CRC-5/10</u>	<u>CRC-30</u>
Power:	117 VAC $\pm 10\%$, 50/60 Hz; 220 VAC $\pm 10\%$, 50/60 Hz on request	
Size:	38.7cm x16.5cm x23.5cm 15 1/4"W x6 1/2"H x9 1/4"D	26.8cm x16.5cm x40.6cm 14 1/2"W x6 1/2"H x16"D
Weight:	18.1 kg (40 lb)	23.6 kg (52 lb)
Features	<ol style="list-style-type: none">1) Push button isotope selection2) Geometry independence3) ^{99}Mo (equilibrium with ^{99}Tc) assay capability4) Accommodates small animals for total body assay5) CRC-5/10 field upgradable to computer/printer Model (CRC-30)	
Reference	<ol style="list-style-type: none">1) Suzuki, A., Suzuki, M.N., Weis, A.M., "Analysis of a Radioisotope Calibrator," Journal of Nuclear Medicine Technology, Volume 4, pg. 193-198, (December 1976).2) Manufacturer's Specifications	
Cost	CRC-5/10 \$3900.00 - \$4450.00	CRC-30 \$7500.00
Address	Capintec, Inc. 136 Summit Avenue Montvale, New Jersey 07645 (201) 391-3930

Exposure/Exposure-Rate Meter
Capintec Model Diarad III

Class	Portable (battery operated) and line operated
Principle of Operation	Air ionization chamber detector and low impedance current/charge meter. The output digitally displayed in R, R/min, mR and mR/min.
Sensitivity and Range	Radiation Detected: X-ray Energy Range: 50 to 250 KvP Resolution: 0.1 mR (mR/min) Interference: From gamma background only Accuracy: $\pm 3\%$ Precision: $\pm 0.1\%$ of full scale display Linearity: 0.5% Stability: $\pm 1\%$ /year
Sampling	Sample discrete field in diagnostic radiology
Performance	Temperature Range: 59°F to 85°F Humidity Range: Up to 60% RH without condensation unit is also usable 50°F to 100°F and up to 95% RH
Requirements	Power: Line Voltage: 115 VAC 50/60 HZ Size: 18 cm x 23 cm x 7 cm plus chamber and cable Weight: 10 lbs.
Features	Can be used to make measurements with other Capintec Ionization Chambers in other areas, e.g. mammography, electron beam therapy, intercavitary, etc.
References	Product Bulletin and Instrument Manual
Cost	\$2,565
Address	CAPINTEC, INC. 136 Summit Avenue Montvale, N.J. 07645 (201) 391-3930

Exposure/Exposure Rate Meter

Capintec Model Therarad I

Class	Laboratory and line operated portable
Principle of Operation	Air ionization chamber detector and rate/integrate electrometer digital display R,R/min, mR and mR/min
Sensitivity and Range	Radiation Detected: X-ray Energy Range: Orthovoltage (0.05 KeV) 30 MeV to 35 MV with suitable build-up cap Resolution: 0.001 R(R/min) Range: 2000 R(R/min) Interference: From gamma background only Accuracy $\pm 0.5\%$ Precision $\pm 0.1\%$ of full scale Linearity $\pm 0.1\%$ Response Time 0.01 sec in integrate mode
Sampling	Sample discrete field in radiation therapy
Performance	Temperature Range: 59°F to 85°F Humidity Range: Up to 60% RH; without condensation unit is also usable 50°F to 100°F and up to 95% RH
Requirements	Power: 115V $\pm$ 10V; 60 Hz Size: 14" x 11" x 8" plus chamber and cable Weight: 15 lbs.
Features	Can be used to make measurements with other Capintec Ionization Chambers in other areas, e.g. mammography, electron beam therapy, interconnecting intracavitary measurements, etc.
References	Product Bulletin and Instrument Manual
Cost	\$3,010
Address	CAPINTEC, INC. 136 Summit Avenue Montvale, N.J. 07645 (201) 391-3930

Exposure/Exposure-Rate Meter
Capintec Model Tomorad I

Class	Laboratory and line operated portable
Principle of Operation	Air ionization chamber detector and rate/integrate electrometer digital display R,R/min, mR and mR/min
Sensitivity and Range	Radiation Detected: X-ray Energy Range: 40-200 KeV effective for CT Scanner Measurement Resolution: 0.001 R cm, R cm/min Range: 2000 R cm, R cm/min Interference: From gamma background only Response to narrow beam X-ray: $\pm 3\%$
Sampling	Sample discrete field in radiation therapy
Performance	Temperature Range: 50°F to 85°F Humidity Range: Up to 60% RH; without condensation, unit is also usable 50°F to 100°F and up to 95% RH
Requirements	Power: 115V $\pm 10V$; 60 Hz Size: 14" x 11" x 8" plus chamber and cable Weight: 15 lbs.
Features	Can be used to make measurements with other Capintec Ionization Chambers in other areas, e.g. mammography, electron beam therapy, intercavitary, etc.
References	Product Bulletin and Instrument Manual
Cost	\$3,650.00
Address	Capintec, Inc. 136 Summit Avenue Montvale, N.J. 07645 (201) 391-3930

Exposure Rate Meter

Capintec Model 192

Class	Dosimeter System
Principle of Operation	System consists of ionization chambers and a low impedance current/charge meter. Direct reading digital output in mR, R, mR/minute, R/minute, nano-coulombs and nano-coulombs/minute.
Sensitivity and Range	20 mR (mR/min) full scale, to 2000 R (R/min) full scale 2.000 nc (nc/min) full scale, to 2000 nc (nc/min) full scale
Performance	Temperature stability: $\pm 0.02\%$ per $^{\circ}\text{C}$
Requirements	Power: 117 VAC, 50/60 Hz, 10 W
Features	<ol style="list-style-type: none">1. Calibrate X-ray diagnostic machines.2. Calibrate Linac, X-ray/cobalt-60 therapy machines.3. X-ray head leakage checks.4. Scattered radiation surveys.5. Intracavitary measurements.6. Monitoring CT system
References	Manufacturer's Specifications
Cost	Model 192 System: \$2190.00 Special Purpose Probes: \$520.00 to \$805.00
Address	Capintec Inc. 136 Summit Avenue Montvale, NJ 07645 (201) 391-3930

Digital Exposure and Exposure - Rate Meter

Class	Portable	
Principle of Operation	System consists of ionization chambers and low input impedance current/charge meter. Direct reading digital output in mR, R, mR/minute, R/minute and/or nano-coulombs and nano-coulombs/minute.	
Sensitivity and Range	<u>174</u>	<u>192</u>
	Range: Rate = 20 mR/min to 200.0 R/min (Full Scale): Integrate = 20 mR to 200.0 R Resolution: 0.01 mR/min or 0.01 mR	2.000 mR/min to 2000 R/min 2.000 mR to 2000 R 0.001 mR/min or 0.001 mR
Sampling	Continuous	Continuous
Performance		
Accuracy:	±0.5% of Electrometer	±0.5% of Electrometer
Response time:	0.61 sec in integrate mode	0.01 sec in integrate mode
Temperature Stability:	±0.02% per °C	±0.02% per °C
Requirements	Rechargeable Ni Cad Batteries	
Power:	117 VAC, 50/60 Hz (230 VAC optional)	117 VAC, 50/60 Hz (230 VAC Optional)
Size:	21.6cm x10.2cm x27.3 cm (8 1/2"W x4"H x10 3/4"D)	5.6cm x26.0cm x19.1cm (14"W x10 1/4"H x7 1/2"D)
Weight:	4.5 kg (10 lb)	Including wood carrying case
Features	<ol style="list-style-type: none">1) Digital numerics display, 3 1/2 digits2) Automatic range selection and display of units3) Selection of probes for wide range of x rays, gamma rays and electron energies4) 174: 3 decade dynamic range for probe 192: 4 decade dynamic range for probe5) 174: Input selector for 2 probes 192: Input selection for 3 probes and more on electrometer mode

Cost 174 192
 \$1750.00 not including detector \$2190.00

Address Capintec, Inc.
 136 Summit Ave.
 Montvale, N.J. 07645
 (201) 391-3930

Capintec Ionization Chambers

Model	Range (digital full scale)		Energy		Response	Collection Efficiency Maximum Exposure- Rate; Continuous Wave.
	Rate (per min)	Integrate	KV	HVL		
PR-06C*	2-2000 R	2-2000 R	50-250	2mm Al to ⁶⁰ Co [†]	±3%	12 kR/min (99.8%)
PM-5*	2-2000 R	2-2000 R	50-250	2mm Al to ⁶⁰ Co [†]	±3%	11.4 kR/min (99.8%)
PM-30*	20 mR - 20R	20 mR - 20 R	50-250	2mm Al to 6mm Cu	±2%	84 R/min (99%)
PR-05**	2-2000 R	2-2000 R	¹³⁷ Cs, ²²⁶ Ra, ⁶⁰ Co		±2%	
PM-500	Suitable for 0.01 mR (mR/min.) to 2 R (R/min.) applications					
PS-033	Suitable for soft x-ray, high energy photon and electron applications					

^{*}Air Equivalent Plastic (Shonka) ^{**}For Intracavitary Application [†]With Equilibrium Cap

IONIZATION CHAMBER SPECIFICATIONS

	PR-06C	PR-05	PR-05P	PS-033	PR-16C	PM-30	PM-500	PM-05	PC-4D
Sensitivity (nC/R) (nominal)	0.200	0.042	0.021	0.160	0.48	9.2	160	1.6	$0.1 \text{ nC} \cdot \text{R}^{-1} \cdot \text{cm}^{-1}$
Resolution (max) R,R/min.*	0.001	0.001	0.01	0.001	0.001	mR,mR/min 0.01	mR,mR/min 0.001	mR,mR/min 0.1	0.001 R·cm or R·cm/min
Maximum Display R,R/min*	2000	2000	2000	2000	2000	20	2	200	2000 R·cm or R·cm/min
Stability (% per year)	± 0.5	± 0.5	± 0.5	± 1	± 1	± 1	± 2	± 1	± 1
Stem Effect (%)	$< \pm 0.5$	$< \pm 1.0$	$< \pm 1.0$	$< \pm 0.5$	$< \pm 1.0$	< 0.5	< 0.5	< 1.0	< 0.5 (for 0.3m×0.3m field)
Leakage Current (A)	10^{-14}	10^{-14}	10^{-14}	10^{-14}	10^{-14}	10^{-14}	10^{-14}	10^{-14}	10^{-14}
Recommended Polarization Voltage (V)	300	300	300	300	300	300	300	300	300
Chamber Volume (ml)	0.65	0.14	0.07	0.5	1.5	28	530	5.3	3.0
Wall Thickness (mm)	0.25	1.2	1.2	0.0036**	0.25	0.50	1.2	0.3	0.33
Material	AIR EQ PLASTIC	AIR EQ PLASTIC	AIR EQ PLASTIC	PET-Al**	AIR EQ PLASTIC	AIR PLASTIC	***	AIR PLASTIC	AIR PLASTIC
Vented	YES	YES	YES	YES	YES	YES	YES	YES	YES
Chamber Connector	BNC SIZE TRIAXIAL								
Averaging Area (cm ²)	N/A	N/A	N/A	N/A	N/A	N/A	100	N/A	N/A

* With Model 192 Digital Dosimeter ** Also available 0.13mm Air Eq Plastic and 25 μm PET-Al

*** Air Equivalent Plastic, Polypropylene, Graphite and Aluminum Alloy

X-Ray Monitors
Model Series 1015

Class	Portable, table-top	
Principle of Operation	Ionization chamber as detector, ratemeter, digital liquid crystal display	
Sensitivity and Range	<u>1015</u>	<u>1015C</u>
	Exposure rate: 1 mR/min to 650 R/min	1 mR/hr to 650 R/hr
	Exposure: 0.02 mR to 99.9 R	0.002 mR to 9.99 R
	Pulse exposure: 1 mR to 13R 1 ms to 99.9s	0.033 mR to 0.433 R 1 ms to 99.9s
	Radiation detected: x-ray, gamma ray from 20 keV to ~1.3 MeV	
Sampling	Continuous	
Performance	Energy dependence: $\pm 10\%$, $\pm 10\%$, 30 keV to 20 keV to 1.33 MeV, 1.33 MeV	
	Exposure rate dependence: $\pm 5\%$, $\pm 5\%$, 20 mR/hr to 0.4 mR/s to 80 R/s 2000 R/hr	
	average, and up to 500 R/s for 50 μ s pulser	
	Operating temperature range: 5°C to 45°C	
	Humidity: up to 95% RH, without condensation	
	Barometric pressure: 5°C to 45°C 600 to 1050 millibars	
	Resolution: 1% or 0.02 mR for exposure 1 mR/min for exposure rate and 0.1 ms for pulse width	
	1% or 0.002 mR for exposure 1 mR/hr for exposure rate and 0.1 ms for pulse width	
Requirements	Power: battery operated Size: 2.54cm x30.4 x20.9cm	
Features	1. Panel control to compensate for barometric pressure changes. Dial millibar settings and readings are temperature and pressure corrected automatically. 2. Analog to digital conversion in probe to eliminate cable noise	
References	Manufacturer's Specifications	
Cost	Consult factory	
Address	Dosimeter Corporation of America 6106 Interstate Circle Cincinnati, OH 45242 (513) 793-6051

Source Position Indicator

Eberline Model SPI-2

Class	Laboratory
Principle of Operation	Uses a halogen quenched GM tube mounted in an Eberline HP-270 hand probe to detect gamma radiation fields; designed primarily for use in areas to indicate when a radiation source is exposed (flashing red display) or when a radiation source is not exposed (flashing green display).
Sensitivity and Range	Alarm level adjustable for fields from 3 mR/hr to 30 mR/hr, threshold is factory-set at approximately 5 mR/hr, unless special request is made. Minimum energy detected: approximately 70 keV Maximum energy detected: greater than 3 MeV Maximum allowable flux: 1000 R/hr
Sampling	Continuous
Performance	Accuracy: Alarms when exposure rate exceeds threshold Temperature: +40° to +110°F (4.5°C to 43°C)
Requirements	Power: 110-125 VAC, 60 Hz at 0.2 amps Size: 22.9Wx21.6Hx6.4cmD (9"Wx8.5"Hx2 1/2"D) Weight: 2.5 Kg (5.5 lbs)
Features	Go, no-go display, battery backup, measures scattered radiation from teletherapy machines, will not saturate in radiation fields as high as 1000 R/hr

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References	Manufacturer's Specifications
Cost	\$765.00
Address	Eberline Instrument Corporation P.O. Box 2108 Santa Fe, New Mexico 87501 (505) 471-3232

Survey Meter
Elscint Model GSM-1

Class	Portable, handheld
Principle of Operation	GM detector, amplifier, meter readout, speaker
Sensitivity and Range	Range: 0-0.5; 0-5.0; 0-50 mR/hr Energy dependence: $\pm 20\%$ from 60 keV to 1.3 MeV
Sampling	Continuous
Performance	Accuracy: $\pm 2\%$ FS Time constant: 2 to 4 sec. Operating temperature: -20°C to 60°C ; -13°F to 140°F
Requirements	Power: Four 1.5 V C batteries. 90 hour continuous operation Size: 16.5cmH x 10.3cmW x 19.1cmD (6 1/2"H x 4 1/16"W x 7 1/2"D) Weight: 1.1 kg (2.4 kbs)
Features	In very high radiation fields meter indicates fullscale not zero Geiger tube may be operated with a cable at a distance up to 50 meters from the reading instrument.
References	Manufacturer's Specifications
Cost	Available on request
Address	Elscint, Inc. P.O. Box 832 138-160 Johnson Avenue Hackensack, NJ 07602 (201) 487-5885

Gamma Monitor
Fisher Controls Model GM-700

Class	Laboratory
Principle of Operation	Ion chamber, amplifier, meter readout, alarm options
Sensitivity and Range	Energy sensitivity: 90 keV to 1 MeV Range: 0.5 to 10 ⁴ mR/h, log
Sampling	Continuous
Performance	Accuracy: $\pm 10\%$ Temperature Range: -10°C to 55°C Relative Humidity: 5 to 100% Drift: Less than 10% of indicated value after 400 hrs. Less than 20% of indicated value after 5000 hrs.
Requirements	Power: 25 watts Size: Monitor, 30.5cm x30.5cm (12"x12"); Head Amp., 17.8cm x12.7cm (7"x5"); Ion Chamber, 55.9cm x19.1cmD (22"x7 1/2"D) Weight: Monitor, 12.7 Kg (28 lbs), Head Amp., 4.1 Kg (9 lbs); Ion Chamber, 12.7 kg (28 lbs)
Features	Alarm level adjustments, recorder output
References	Manufacturer's Specifications
Cost	Refer to manufacturer
Address	Fisher Controls Ltd. Measurement and Analytic Division Century Works, Lewisham London SE13 7LN, England Attn A.E. Pegg Tel. 01-692 1271 Ext. 260 International # +44 (1) 692-1271 Telex 22469

Four Channel Gamma-Ray Spectrometer
Fisher Controls Type 0928

Class	Portable, designed for field use
Principle of Operation	Used with a variety of probes; it has a discriminator and three energy channels all of which may be preset within the range 50 keV to 3 MeV. Counts derived from γ -ray energies exceeding the discriminator threshold or lying within the channel boundaries, may be displayed one at a time on the front panel meter and, simultaneously, on an external 4-channel recorder; alternatively they may be displayed one at a time as an accumulated count on a digital display.
Sensitivity	Range (for channels 1,2, and 3): 0-10, 0-30, 0-100, 0-300, 0-1000 cps Range (discriminator): 0-100, 0-300, 0-1000, 0-10,000 cps Sensitivity: depends on probe Energy limits: Discriminator 50 keV to 1 MeV Channel 1 Channel 1 1 MeV to 2 MeV Channel 2 Channel 2 1.25 MeV to 2.25 MeV Channel 3 Channel 3 2 MeV to 3 MeV
Features	1) Spectrum stabilized using 60 keV line from ^{241}Am reference source contained in probe 2) Front panel switch allow test pulser of 1Hz or 136Hz to be injected into digital counting circuits so that the correct operation of each counting chain may be monitored.
References	Manufacturer's Specifications
Cost	Refer to Manufacturer
Address	Fisher Controls Nuclear Controls Division, Century Works Lewisham, SE13 7LN, England 01-692-1271 Telex: 22469

Gamma Survey Meter
Fisher Controls Type 0593

Class	Portable
Principle of Operation	γ -ray detector, solid-state circuitry, scaler, liquid crystal display, and audible alarm
Sensitivity and Range	Range: 0-5 digit readout
Sampling	Continuous
Performance	Count cycling time: 2 seconds Operating temperature range: -5°C to 50°C
Requirements	Power: Six "C" cells Size: 7.4cm x12.4cm x24.0cm (3 in. x 5 in. x 9 1/2 in.) Weight: 1.6 kg (3.5 lb)
Features	1) Audible alarm has an adjustable threshold and pitch related to count rate 2) Hermetically sealed 3) Fully compensated to accommodate voltage changes between 9V and 6.6V 4) Battery test switch
References	Manufacturer's Specifications
Cost	Refer to Manufacturer
Address	Fisher Controls Century Works Lewisham, London SE13 7LN, England Tele: 01-692-1271

Gamma Survey Meter
Fisher Controls Model NE-148A

Class	Portable, handheld
Principle of Operation	Sodium iodide scintillator, photomultiplier tube, ratemeter, meter readout
Sensitivity and Range	Energy sensitivity: Range 0.15 MeV to 1.25 MeV Range: 0 - 30, 300, 3000 μ R/hr (130, 1200, 10000 cps) [radium equilibrium] Response time: 2, 0.2, 0.02 sec
Sampling	Continuous
Performance	Meter accuracy: $\pm 10\%$ full scale Temperature range: -25°C to $+44^{\circ}\text{C}$
Requirements	Power: Two 1.5V "D" cells, 400 hours Size: 21.5cm x 6cm x 14.6cm (8.5"x2.4"x5.7") Weight: 1.8 kg (4 lbs)
Features	Audio output jack, waterproof to a depth of 5 ft., meter light A.E.R.E. Type 1597A
References	Manufacturer's Specifications
Cost	Refer to manufacturer
Address	Fisher Controls Ltd. Measurements and Analytics Division Century Works, Lewisham London SE13 7LN, England Attn: A. E. Pegg Tel. 01-692 1271 Ext. 260 Telex 22469

Survey Meter

Gamma Industries Models 150B, 155B, 250B

Class	Portable	
Principle of Operation	Geiger-Muller tube as detector, amplifier, electrometer, meter readout	
Sensitivity and Range	<u>150B, 155B</u>	<u>250B</u>
Range:	0 - 2 mR/hr 0 - 20 mR/hr 0 - 200 mR/hr	0 - 10 mR/hr 0 - 100 mR/hr 0 - 1000 mR/hr
Sampling	Continuous	
Performance	150B - internal probe 155B - external probe

Performance (cont.)

Accuracy: $\pm 20\%$ FS
Operating temperature range: -17.8°C to 75°C (0°F to 167°F)
Time constants: ~ 5 sec
Energy range: 250 keV to 1.5 MeV

Requirements

150B, 155B

250B

Power:

Two "D" cells

Two "D" cells

Size:

8.25cm x20.32cm x10.16cm
(3 1/2"W x8"L x4"H)

8.26cm x20.32cm x10.16cm
(3 1/4"W x8"L x4"H)

Weight:

1.135 kg (2.5 lb)

1.25 kg (2.75 lb)

Features

- 1) Will not reverse in high intensity gamma radiation fields
- 2) Low power consumption: > 400 hours from ordinary batteries, up to 1000 hrs on alkalines.

References

Manufacturer's Specifications

Cost

Model 150B \$370.00
Model 155B \$415.00
Model 250B \$275.00

Address

Gamma Industries
924 Joplin Street
P.O. Box 2543
Baton Rouge, LA 70821
(504) 387-0846
TWX (510) 993-3494

Portable, Low Level Gamma and X-Ray Survey Meter
Harshaw Model TASC-21

Figs.1 and 2. Two methods of carrying the TASC-21 are shown. Figure 1 illustrates a shoulder strap and Fig. 2 illustrates a neck strap with the electronics resting on the surveyor's chest.

Class	Designed for rough handling in the field. Fully portable.
Principle of Operation	A Phoswich, a two phosphor scintillation detector employs materials with different decay times and gamma sensitivities to distinguish between pulses from low energy photon emitters and background events. A pulse shape analyzer and time to amplitude converter separates low energy photon interactions from background events as they occur in the phosphors and processes the resulting signals to actuate output display and alarm devcies.
Sensitivity and Range	The detector is optimized for high efficiency detection of 15 - 100 keV photons. Phoswich detector enables a background reduction of three to six times as compared to single NaI crystal types such as the FIDLER detector, by reduction of Compton events, photomultiplier tube noise rejection and active guarding.
Sampling	Continuous. Microprocessor based counting circuits provide continuous monitors via a logrithmic ratemeter with internally selectable time constants. A preset alarm threshold controls an audible "beeper" which responds to an average count rate. Static count measurements are made using operator selected counting intervals. An eight digit LCD provides numeric indication of counts, time and background.

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Requirements	Power; provided by battery pack Size; 8" x 11" x 9" high Weight: 36 lbs.
Features	Light-weight - No line power required - Designed for remote area surveying. Battery charger included.
References	Harshaw Technical Literature <u>A Battery Operated Portable Phoswich Detector for Field Monitoring of Low Levels of Transuranic Contaminants by C. J. Umbarger and M. A. Wolf of Los Alamos Scientific Laboratory.</u> Nuclear Instruments and Methods, 155 (1978) Pg. 453 - 457.
Cost	TASC-21 \$8,000.00
Address	Harshaw Chemical Company 6801 Cochran Road Solon, OH 44139 (216) 248-7400

Environmental Radiation Monitoring System - Air/Water

Harshaw Model TASC-4

Class	Continuous, real-time gamma radiation monitor for water, air or gaseous effluent pathways; non-portable.
Principle or Operation	NaI(Tl) scintillation detector drives stabilized electronic counting chain. Output meters calibrated in $\mu\text{R/hr}$ or pCi/l . Recorder outputs and alarms available.
Sensitivity and Range	Air radiation monitor range is $1 \mu\text{R/hour}$ to $10 \mu\text{R/hour}$. Water effluent radiation monitor sensitivity is 10 pCi/liter . Energy dependence: substantially independent of gamma ray energy in the range of 0.1 to 2.0 MeV .
Sampling	Continuous, real-time monitoring for both area and effluent systems.
Performance	Rugged POLYSCIN R NaI(Tl) Scintillation Crystals are used. They are packed in foam plastic and housed in a weather-proof/waterproof container. An ^{241}Am pulser is incorporated in the scintillator for stabilizing the system against temperature changes and photomultiplier tube gain shift.
Requirements	Power: 115/230V AC Size: $8 \frac{3}{4} \text{ H} \times 19 \text{ W} \times 15 \text{ D}$ per channel Weight: ~60 pounds per channel
Features	"Rugged and Reliable" environmentally protected scintillation detection. No gas counters. Counting instruments can be located up to 1 mile away from detector. Long term accuracy assured by ^{241}Am pulser. Continuous performance verification by reference pulser alarm.

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References	Harshaw Product literature available
Cost	\$7500.00 per channel - water monitor \$10,685.00 - Air Monitor
Address	Harshaw Chemical Co. Crystal and Electronic Products Dept. 6801 Cochran Road Solon, OH 44139 (216) 248-7400

Tissue Equivalent Digital Survey Meter
Health Physics Model 1010

Class	Portable, handheld		
Principle of Operation	Sealed tissue equivalent proportional chamber detector, electrometer meter readout or liquid crystal display.		
Sensitivity and Range	Range (rate): 0-0.1, 1, 10, 100, 1000 mrad/hr (Dose): 0-0.01, .1, 1 mrad		
Sampling	Continuous		
Performance	Wall thickness: 0.1 inches Time constants: 0-0.1 mrad/hr 10 sec. 0-1 mrad/hr 5 0-10 2.5 0-100 <1 0-1000 <1 Energy response: $\pm 5\%$ from 100 keV to 1.25 MeV (with response - 10% at 70 keV to soft tissue) - 20% at 40 keV		
Requirements	Power: Four "D" cells, 80-100 hours lifetime Size: 15cmx10cmx35cm (5.75"x4"x13.75") Weight: 2.4 kg (5.25 lb)		
Features	1. Solid state circuitry, IC-controlled high voltage supply 2. Chamber has center wire, filled with tissue - equivalent gas 3. Controls for high voltage, zero, and three calibration adjustment controls 4. Ion chamber is two inch spherical shape.

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References	Manufacturer's Specifications
Cost	\$1950.00
Address	Health Physics Instruments, Inc. 124 Santa Felicia Drive Goleta, CA 93017 (805) 685-2612

Portable Remote Area Monitor
Health Physics Instrument Model 1072

Class	Portable, handheld
Principle of Operation	Uses an air equivalent, unsealed ionization chamber as detector coupled with amplifier, ratemeter, and liquid crystal display readout with alarms
Sensitivity and Range	Dose rate range: 0.1 to 199.9 mr/h (1072-1) 1 to 1999 mR/hr (1072-2)
Sampling	Continuous
Performance	Accuracy: overall, $\pm 15\%$ or two digits, whichever is greater Detector volume: 1 liter Temperature dependence: zero shift, ± 5 in least significant digit 0-50°C Operating range, 10°C to 50°C Storage range, -20°C to 65°C Pressure dependence: Reading dependent on elevation and barometric pressure Warm-up time: 30 seconds Response time: 3 seconds Exposure rate response: 90% collection at 2 R/h off scale indication, 1- - - (a numeral in the left-hand digit with other digits blanked).
Requirements	Power: Batteries, eight "D" cells Size: 25cmHx16cmWx30cmL (10"x6 1/2"x12") Weight: 4 kg (8.8 lbs.)
Features	Audible and visual alarm LCD displays arrow for low battery condition
References	Manufacturer's Specifications
Cost	\$985.00
Address	Health Physics Instruments, Inc. 124 Santa Felicia Drive Goleta, CA 93017 (805) 685-2612

X-Ray Survey Meter
Johnson and Associates Model TVX-1

Class	Portable, handheld
Principle of Operation	Six GM tubes spaced equidistantly to provide a large scanning area (20"x5") for the detection of x rays from color TV receivers
Sensitivity and Range	Energy sensitivity: 15 keV to 10 MeV Range: 1.05 mR/hr to 1 R/hr; 3000; 10K; 30K, 100K cpm Minimum sensitivity: 0.05 mR/hr
Sampling	Continuous
Performance	Accuracy: Unavailable
Requirements	Power: Three 9 volt batteries Size: 65cmL x11cmW x4cmD (26"L x4.5"W x1.5"D) Weight: 1.6 kg (3.5 lbs)
Features	The TVX-1 graphite-coated lucite screen shields RF radiation while permitting low energy x rays to pass unattenuated.
References	Manufacturer's Specifications
Cost	Refer to manufacturer
Address	William B. Johnson and Associates Research Park Boonton Avenue Montville, NJ 07045 (201) 334-9222

Survey Meter
Johnson Model GSM-8

Class	Portable, handheld
Principle of Operation	Halogen quenched GM detector, amplifier, meter readout
Sensitivity and Range	Gamma ranges, linear 0-10, 0-100, 0-1000 mR/hr
Sampling	Continuous
Performance	Linearity: $\pm 10\%$ full scale Meter display - 10 second time constant Instrument Saturation: over 1000 R/h
Requirements	Power: 4.5 volts D.C. (3 D cells) Size: 10.2cm x 16.3cm x 17.8cm (4"x 6"x 7") Weight: 1.6 kg (3 1/2 lbs) (including batteries and detector)
Features	Internal detector, light weight, non-saturable Intermediate range
References	Manufacturer's Specifications
Cost	\$295.00
Address	Wm. B. Johnson and Associates, Inc. Research Park, Boonton Ave. Montville, N.J. 07045 (201) 334-9222

X-Ray Proportional Detectors

LND Models

X-ray Detectors

Class	Detector Maximum Dia.		Overall Length		Operating	Price	Gas
Model	(cm)	(in.)	(cm)	(in.)	Voltage		
456	2.54	1.00	10.16	4.00	1800	\$650.00	Xenon
4561	3.00	1.181	9.50	3.740	1850	\$570.00	Xenon
4562	2.54	1.00	7.62	3.00	1300	\$550.00	Xenon
4563	2.54	1.00	10.16	4.00	1400	\$775.00	Neon
4564	2.54	1.00	10.16	4.00	1400	\$800.00	Neon
4565	3.33	1.31	10.16	4.00	1200	\$555.00	Xenon
4566	2.54	1.0	13.97	5.5	1800	\$585.00	Xenon
4567	6.35x6.35	1.25x1.25	6.67	2.625	1850	\$538.00	Xenon
4568	4.20x4.20	1.653x1.653	14.47	5.697	1900	\$780.00	Xenon

Reference Manufacturer's Specifications

Address LND
3230 Lawson Blvd.
Oceanside, NY 11572
(516) 678-6141

Gasfilled Counters
Ionization Chambers
LND Models

Class	Detectors		
Model	X Ray proportional Counters		Price
	Type	Remark	
400	End Window	$\beta > 30$ keV γ or X from 0.5 to 1.0nm	\$395.00
4011	End Window	$\beta > 30$ keV γ or X < 1.1 nm	435.00
4012	End Window, Removable		455.00
4031	End Window	$\beta > 30$ keV γ or X < 1.0 nm	685.00
4032	End Window, Removable		595.00
420	Quadralateral Side Window	β, γ or X < 1.0 nm	535.00
421	Quadralateral Side Window	β, γ or X < 1.0 nm	540.00
422	Quadralateral Side Window	β, γ or X < 1.0 nm	545.00
423	Quadralateral Side Window	β, γ or X < 1.0 nm	540.00
424	Quadralateral Side Window	β, γ or X < 1.0 nm	630.00
425	Quadralateral Side Window	β, γ or X < 1.0 nm	820.00
426	Quadralateral Side Window	β, γ or X < 1.0 nm	1320.00
428	Large Area	2 mil Beryllium Window	2950.00
450	Side Window		550.00
451	Side Window		745.00
4515	Windowless		630.00
452	Side Window		590.00
4522	Windowless		595.00
453	Side Window		730.00
4543	Side Window		695.00
4548	Side Window		735.00

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4544	Side Window	695.00
4545	Side Window	735.00
45431	Side Window	605.00
4549	Side Window	790.00
4546	Side Window	605.00
4545	Side Window	735.00
45410	End Window	755.00

Ionization Chambers

Model	Sensitivity	Range	Remark	Price
503	1x10 ⁻¹⁰ Amp R ⁻¹ hr ⁻¹	10 mR/hr to 10 ⁴ R/hr		\$560.00
512			High Pressure Argon	1335.00

Ionization Well Chambers

Model	Background	Remark	Price
507	1 x 10 ⁻¹⁴ Amps		\$735.00
508	1 x 10 ⁻¹⁴ Amps		735.00
509	1 x 10 ⁻¹⁴ Amps		780.00

Reference	Manufacturer's Specifications
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Remarks	Many more detectors with differing specifications for many uses.
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Address LND
3230 Lawson Blvd.
Oceanside, NY 11572
(516) 678-6141
Telex 14-4563

Survey Meter
Ludlum Model 5

Class	Portable, handheld
Principle of Operation	Geiger-Muller detector, amplifier, electrometer, meter readout
Sensitivity	Energy Range: lowest → highest dependent on thickness of entrance window and volume of the detector Activity Range: 0-0.2 mR/hr to 0-2000 mR/hr in 5 linear ranges Saturation In excess of 200 R/hr
Detector	LND 712 with 0.032 in. cadmium shield for 0-20 mR/hr LND 714 for 0 to 200 and 0 to 2,000 mR/hr scales
Requirements	Power: Two "D" cells, 300 hours Size: Model 5 - 9 cm x 9 cm x 18 cm, exclusive of handle (3 1/2" H x 3 1/2" W x 7" L) Weight: Model 5 - 1.6 kg (3.5 lb)
Performance	Calibration accuracy: -5% Temperature: -10°F to +120°F Calibration: Radium-226
Features	Built-in speaker; temperature compensated
References	Manufacturer's specifications
Cost	Model 5 \$345.00
Address	Ludlum Measurements, Inc. 501 Oak Street Sweetwater, TX 79556 (915) 235-5494

Survey Meter
Ludlum Model 14-C

Class	Portable, handheld
Principle of Operation	Uses two GM tubes, amplifier, electrometer, meter readout Two GM tubes - one internal one external
Sensitivity and Range	Energy Sensitivity: see attached curve Full scale ranges: 0.2, 2, 20, 200, 2,000 mR/hr (internal GM) Minimum detectable energy: 50 keV for internal GM - external GM depends on type selected
Sampling	Continuous
Performance	Meter Accuracy: 2% Response time: Fast - 3 seconds Slow - 11 seconds
Requirements	Power: Two "D" cells, 300 hours life Size: 9 cm x 9 cm x 18 cm (3.5" x 3.5" x 7") Weight: 1.5 kg (3-1/2 lbs)
Features	Speaker provided built-in
References	Manufacturer's specifications
Cost	\$335.00 plus cost of external detector
Address	Ludlum Measurements Inc. 501 Oak Street Sweetwater, TX 79556 (915) 235-5494

Survey Meter
Ludlum Model 14-C

Ludlum Measurements, Inc. Geiger counter
Model 14 Serial No. 85 *

Ludlum Measurements, Inc. Geiger counter
Model 14 Serial No. 117 *

* "The Photon Energy Response of Several Commercial Ionization Chambers, Geiger Counters, and Thermoluminescent Detectors" Los Alamos Scientific Laboratory Report LA-4052, UC-37, Instruments TID-4500, Feb. 1969.

X-Ray Monitor
MHD Model 1015C

Class	Portable, designed for verifying proper operation of diagnostic x-ray machines.
Principle of Operation	Uses an air-equivalent ion chamber in conjunction with a current-to-frequency converter and digital-processing techniques to measure exposure, exposure rate, pulse exposure, and pulse width.
Sensitivity	Ranges· exposure rate; 1 mR/hr to 59,940 R/hr exposure (integrated) 2 μ R to 999R pulse exposure 1 mR to 13R pulse duration: 1 msec to 99.9sec
Sampling	Continuous
Performance	Operating temperature· 5°C to 45°C Operating humidity: up to 95% RH, without condensation Barometric pressure. 600 to 1050 millibarns Energy dependence· depends upon chamber used Exposure rate dependence depends upon chambers Noise level 1 digit, rms Resolution· 1% or 1 digit Uncertainty in pulse-amplitude discrimination Level. 5%, rms
Requirements	Power· battery with lifetime ~750 operating hours Size· } Weight· } refer to manufacturer

3.3.4.md-1

Features	A number of options are available The 20 ft multiconductor cable between electrometer and readout unit carries only digital signals, bias voltage, and power
References	Manufacturer's Specifications
Cost	\$2,450.00 options and chambers available for a price
Address	MDH Industries, Inc. 426 West Duarte Road Monrovia, CA 91016 (213) 357-7921

X-Ray Monitor
MDH Model 2025 AC

Class	Portable
Principle of Operation	Uses air-equivalent ion chambers with associated electronics for autoranging over 6 decades. Display is 3 1/2 digit liquid crystal.
Sensitivity	Exposure rate range: 0.001 R/h to 600,000 R/h Integrated exposure range: 0.01 mR to 2000R
Sampling	Continuous
Performance	Energy response: typically $\pm 5\%$ (refer to chamber specs) Accuracy: $\pm 4\%$ Repeatability: $\pm 1\%$ Bias voltage: 300V battery
Requirements	Power: 4 - "C" cells batteries, life over 1,000 hours Size: Carry case, (14"x14"x5" nominal); carries total system Weight system weight, (12 1/2 lb)
Features	Uses several different chambers for different purposes and energy ranges Included is a circuit check to be sure electronics are performing as intended
References	Manufacturer's Specifications
Cost	\$925.00 (exclusive of electrometer/ion chambers)
Address	MDH Industries, Inc. 426 West Duarte Road Monrovia, CA 91016 (213) 357-7921

Gamma Counters

N. Wood Models

Class Portable and/or Laboratory
Principle of Proportional and GM as specified
Operation

Proportional Gamma Counters

MODEL	DIAMETER	LENGTH	VOLTAGE	GENERAL	Cost
D-5-3L	5/8 or 1/2"	3"	2000 volts	Originally made for space re- search--adapted for use in port- able instruments. Useful for all cosmic ray counting.	\$ 70.00
D-10-5L	1"	5"	2050 volts		85.00
D-15-12L	1 1/2"	12"	2100 volts		100.00
D-15-34L	1 1/2"	34"	2100 volts	Filled to operate in the propor- tional region, these counters have an almost unlimited life.	113.00
D-20-38L	2"	38"	2200 volts	Larger models useful for cosmic ray telescopes, anticoincidence shields, carbon dating, etc.	155.00

Geiger Mueller Counters

D-5-3	5/8"	3"	1000 volts	Used in portable instruments and probes--design adapted to associ- ated counting equipment. Specify length up to 16".	\$ 42.00
D-5-16	5/8"	16"	1000 volts		50.00
D-10-5	1"	5"	1050 volts		56.00
D-10-12	1"	12"	1050 volts		63.00
D-15-5	1 1/2"	5"	1100 volts	General purpose gamma counters for research, geological survey, well logging, carbon dating, anticoincidence shields, and many industrial uses. 5" to 34" active length for counters 1 1/2" in diameter. 12" to 46" active length for counters 2" in diameter.	63.00
D-15-12	1 1/2"	12"	1200 volts		70.00
D-20-12	2"	12"	1200 volts		92.00
D-20-20	2"	20"	1200 volts		105.00
D-20-34	2"	34"	1200 volts	High sensitivity GM counters.	120.00

References Manufacturer's specifications

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Remarks Counters can be made to specifications by special order.

Address N. Wood Counter Laboratory, Inc.
 1525 E. 53rd Street
 Chicago, IL 60615
 (312) 324-1114

Survey Meter

Nuclear Associates Model 05-571,572

Class	Portable, handheld	
Principle of Operation	GM tube, amplifier, meter readout	
Sensitivity and Range	<u>Model 571</u>	<u>Model 572</u>
	Range: 0-10, 100, 1000 mR/hr	0-500, 5000, 50,000 cpm
Sampling	Continuous	
Performance	Accuracy: $\pm 20\%$ of full scale	$\pm 10\%$ of full scale
	Operating temperature: $\pm 20^{\circ}\text{C}$ to $+55^{\circ}\text{C}$	-20°C to $+55^{\circ}\text{C}$
	Energy dependence: $\pm 20\%$, 60 keV $\rightarrow$ 1.2 MeV	
	Time constants: 10, 2, & .3 sec	10 sec(X1), 2 sec(X10), .3 sec(X100)
Requirements	Power: Four "AA" alkaline batteries, ~500 hour life	Same as Model 571 ~500 hour life
	Size: 15.2 cm x 8.9 cm x 3.8 cm thick	15.2 cm x 8.9 cm x 5.1 cm
	6" x 3 1/2" x 1 1/2"	6" x 3 1/2" x 2"
	Weight: 0.43 kg (15 oz.)	0.62 kg (22 oz.)
Features	571 - . A nonoverload circuit assures indicator operation in radiation field greater than 500 R/hr	
	572 - . High sensitivity (lower detection limit of 0.005 Ci)	
	. Large area, screened detector for surface contact measurements

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References Manufacturer's specifications

	571	572
Cost	<u>\$270.00</u>	<u>\$415.00</u>
Address	Nuclear Associates 100 Voice Road Carle Place, N.Y. 11514 (516) 741-6360

Hand/Shoe/Clothing Monitor
Nuclear Associates Model 05-696

Class	Floor Mounted
Principle of Operation	Xenon-Filled Proportional Counter with meter display
Sensitivity and Range	Three channels with 1 - 10^4 counts per cycle logarithmic scales 1900 cpm per 10^{-4} $\mu\text{Ci}/\text{cm}^2$ of I-125
Sampling	10 sec count cycle
Performance	Temperature Range: -5°C to $+50^\circ\text{C}$
Requirements	Power: 110 Vac - 60 Hz Size: 64" x 20" x 27" Weight:
Features	Four large area 200 cm^2 window detectors Window thickness: 6 mg/cm^2 -- background subtract
References	Manufacturer's specifications
Cost	\$8500.00
Address	Nuclear Associates 100 Voice Road Carle Place, N.Y. 11514 (516) 741-6360

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Primalert 10 Therapy Radiation Level Monitor

Nuclear Associates Model 05-433

Class	Wall-Mounted
Principle of Operation	Energy Compensated GM Detector and ratemeter with audible and visual alarm
Sensitivity and Range	Alarm Level 2.5 mR/hr or 20 mR/hr
Sampling	Continuous
Performance	Energy Response: -10%, +40% from 50 keV to 2 MeV referenced to Cs-137 Accuracy: $\pm 20\%$ (exclusive of energy dependence)
Requirements	Power: 110 Vac - 60 Hz Size: 6" x 3.5" x 1.5" Weight:
Features	Visual Alarm and Audible Alarm Optional Remote Alarm, Model 05-434, Cost: \$110.00
References	Manufacturer's specifications
Cost	\$375.00
Address	Nuclear Associates 100 Voice Road Carle Place, N.Y. 11514 (516) 741-6360

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Primalert 35 Area Alarm Monitor

Model 05-437

Class	Wall-Mounted
Principle of Operation	Energy Compensated GM Detector
Sensitivity	1, 2, 4, 8, 16 and 32 mR/hr Alarm can be set at any of the six levels
Sampling	Continuous
Performance	Energy Response: $\pm 20\%$ 80 keV - 1 MeV Accuracy: $\pm 20\%$ (exclusive of energy dependence)
Requirements	Power: 110 Vac - 60 Hz Size: 5.9" x 3.4" x 1.3" Weight:
Features	Aural and Visual Alarms Optional Remote Alarm, Model 05-434 (Cost \$110.00)
References	Manufacturer's Specifications
Cost	\$435.00
Address	Nuclear Associates 100 Voice Road Carle Place, N.Y. 11514 (516) 741-6360

05-582 "Transport Index" Radiation Monitor (Prima 7)
and Survey Meter
for measuring "Transport Index" of packages containing
radioactive materials

Class	Portable, handheld (with belt-clip)
Principle of Operation	Halogen-quenched GM tube, LED readout (2-digit)
Sensitivity and Range	Range: 0-99 mR/hr, in 1 mR increments (with over-range indicator)
Sampling	Manual Control

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Performance	4-second measurement cycle Accuracy: $\pm 10\%$ or 1 mR/hr whichever is greater Control: Press-to-measure (4 seconds); hold-to-read Energy response: +40%, -12% from 50 keV to 1.2 MeV Response time: 4 seconds Temperature: 0°C to 50°C (32°F to 120°F) with mercury cells -12°C to 50°C (10°C to 120°F) with alkaline cells
Requirements	Power: 4 "AA" mercury cells or 4 "AA" alkaline cells Size: 15.2 cm x 8.9 cm x 3.8 cm (6" H x 3 1/2" W x 1 1/2" D) Weight: 0.34 kg (12 oz.)
Features	1. Accurate LED readout 2. Automatic counting cycle 3. "over-range" and "read" indicators 4. 3-foot length telescoping spacing rod 5. Calibration source ^{137}Cs , ~10 μCi , License-free
References	Manufacturer's specifications
Cost	\$285.00
Address	Nuclear Associates 100 Voice Road Carle Place, N.Y. 11514 (516) 741-6360

X, Gamma Survey Meter
Nuclear Associates 05-670

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Class	Portable
Principle of Operation	Single plastic wall ion chamber used both as a proportional gas amplification chamber and as an ionization chamber (by placing -200 V d.c. on the chamber), amplifier, electrometer, meter readout.
Sensitivity and Range	Energy Dependence: See curve; $<\pm 10\%$ from 10 keV to 1.2 MeV Dose Ranges: 15 linear scales, 0.3 mR to 3000 R (full scale) Dose Rate Range: 15 linear scales, 0.3 mR/hr to 3000 R/hr (full scale)
Sampling	Continuous
Performance	Instrument Accuracy (Less Probe); 6% for dose rate measurement 8% for dose measurement Time Constants: 1, 3, 10 seconds
Requirements	Power: 110 V a.c., 60 Hz Size: 14 cm H x 25 cm W x 14 cm D (5.5" x 10" x 5.5") Probe cable 2 m L (6') Weight: 3.6 kg (8 lb)
Features	Measures both dose and dose rate
References	Manufacturer's specifications
Cost	05-670 X/Gamma Ray Survey Meter \$3895.00 05-671 Heavy-Duty Probe Holder 150.00 05-672 Probe Extension Cable, 20 Ft. 195.00
Address	Nuclear Associates 100 Voice Road Carle Place, N.Y. 11514 (516) 741-6360

NaI(Tl) Crystals
Nuclear Enterprises

Class	Detectors
Principle of Operation	NaI(Tl) crystal optically coupled to a photomultiplier tube
Specifications	<p>Crystal Sizes: Range from 1/2"x1/2" to 5"x6"</p> <p>Prices: Range from \$76 to \$3121</p> <p>De-mountable Assemblies: Can accept crystal sizes from 1/2" x1/2" to 6" x4"</p> <p>Prices: Range from \$300 up</p> <p>Integral Assemblies: Can accept crystal sizes with diameter up to and including 3"</p> <p>Prices: Range from \$178 to \$918</p> <p>NEI will consider manufacturing special or custom detectors (any quantity)</p>
Other Features Also Available	<p>Electronic accessories</p> <p>Low-background quartz windows & stainless steel mounts, Beryllium windows</p> <p>Anti-Compton set-ups</p> <p>Large well set-ups</p> <p>Cesium-iodide (thallium) detectors also available</p>
References	Manufacturer's Specifications
Address	<p>Nuclear Enterprises, Inc. 931 Terminal Way San Carlos, CA 94070 (415) 592-8663. Telex: 34871</p> <p>or</p> <p>Nuclear Enterprises, Ltd. Bath Road, Beenham Reading ENGLAND</p>

Area Radiation Monitor System TA-900

Class	Rack or Nema cabinet - mount w/remote detector
Principle of Operation	Microprocessor based using three (3) detector automatic ranging and digital readout in mR/hr, two significant digits and powers of ten.
Sensitivity and Range	0.01 mR/hr to 0.99×10^7 mR/hr 80 keV to 1.5 MeV energy range Higher ranges available on request
Sampling	Every four (4) seconds.
Performance	Accuracy $\pm 20\%$ Energy Dependence $\pm 20\%$ Temperature - Detector: -40 to $+75^\circ$, 95% Rh Control Unit: -20 to $+60^\circ\text{C}$, 95% Rh
Requirements	Power: 120 VAC, 30 watts, plus external alarm power Size: Nema control unit: $14 \times 15\frac{1}{2} \times 7$ Detector: $9\frac{1}{2} \times 11\frac{1}{2} \times 4\frac{3}{4}$ Weight: 20 pounds
Features	Outputs for remote alarms Microprocessor capability for background subtraction, and other special features.
References	
Cost	\$4,200.00
Address	Nuclear Research Corporation 1105 Industrial Highway Southampton, PA 18966

Si(Li) Photon Detectors

Ortec Series 7000

Class	Detectors
Principle of Operation	Silicon (lithium-drifted) detectors for low-energy photon spectrometry

Beryllium Entrance Windows		Price
Standard	{ 1 mil Be Window. Normally delivered with 4, 6, & 10mm diameter detectors. 2 mil Be Window, also.	
Optional		
	{ 0.5 mil Be Window. Thin, unsupported window for 4 and 6mm diameter detectors. 0.3 mil Be Window. Ultra-thin, unsupported window for 4mm diameter detectors.	\$275
		\$385
Features	(1) Price includes: (a) Si(Li) crystal (b) Cryogenic Dynamic Charge Restoration Preamplifier (c) Cryostat with entrance window (d) Liquid nitrogen dewar (2) Figure shows effect of different beryllium windows and Si(Li) thicknesses.	
References	Manufacturer's Specifications	
Address	ORTEC, Inc. 100 Midland Road Oak Ridge, TN 37830 (615) 482-4411

Specifications:

Cryostat Configuration	ORTEC Model No.	Dewar Size (Liters)	Liquid Nitrogen Use Time (Liters/Day)	Guaranteed Resolution* (eV) FWHM for ~ Area/Diam.				Range of Price
				12.5mm ² / 4mm	30mm ² / 6mm	80mm ² / 10mm	200mm ² / 16mm	
Vertical Dipstick	7013	30	1.1	205 160	215 175	230 190	325 285	From \$5750
Horizontal Dipstick	7113	30	1.1	205 160	215 175	230 190	325 285	to \$7230
Light Weight	7513	5	0.9	205 160	215 175	230 190	325 285	

Detectors are available with a guaranteed resolution better than that shown in table.

*Total system resolution for ⁵⁵Fe 5.9 keV at 1000 counts/sec measured with an ORTEC 716A, a 459, and a dc-coupled 512-channel analyzer. The FWTM is ≤ 2 FWHM. Guaranteed for one year. Upper value is standard. Lower value is premium.

Ge(Li) Gamma Spectrometry Detectors, Systems

Ortec 8000 Series

Vertical Dip Stick

Horizontal Dip Stick

Comparison of Ge(Li) LEPS and Si(Li) Detectors for High-Energy X-Ray Analysis.

Class	Laboratory
Principle of Operation	Germanium (Lithium-Drifted) detectors used for gamma spectrometry, and full systems including dewar, preamplifier, cryostat.
Features	<ol style="list-style-type: none">(1) Detector efficiency is compared to that of 3" x 3" NaI(Tl).(2) Geometry is true cylindrical on all coaxial detectors.(3) Electronic accessories (preamplifier, etc.) available.(4) Spectrometer system has $\leq 1 \mu\text{m}$ 'window'.(5) End cap is 0.005" beryllium on all models except 0.010" on 25-mm- and 32mm diameter models.(6) Cryosorption pumping used in all spectrometers.
References	Manufacturer's Specifications
Address	ORTEC, Inc. 100 Midland Road Oak Ridge, TN 37830 (615) 482-4411

Specifications

8000 Series Ge(Li) Coaxial Detectors^a

Detector Series	Vertical Cryostat Model No.	Horizontal Cryostat Model No.	Relative Photopeak Efficiency (%)	Resolution FWHM ^b (keV) for ⁶⁰ Co 1.332 MeV Gammas	Peak-to-Compton Ratio	Price
VIP10 Series Delivery: from stock	8001-1020V	8101-1020V	≥10	≤2.0	≥33:1	
	8001-1022V	8101-1022V	≥10	<2.2	≥30:1	
WIN15 Series Delivery: from stock	8001-1521W	8101-1521W	≥15	≤2.1	≥36:1	
	8001-1523W	8101-1523W	≥15	≤2.3	≥33:1	
TEC20 Series Delivery: from stock to 60 days	8001-2021T	8101-2021T	≥20	≤2.1	≥41:1	
	8001-2023T	8101-2023T	≥20	≤2.3	≥38:1	
CUSTOM SYSTEMS Delivery: as per quotation	Single and multiple detector To 75 arrays available in a wide range of cryostat, end cap, and dewar configurations			To 1.7 Cooled FET Systems Available	200:1	

^aThe ORTEC Ge(Li) Detectors are true coaxial geometries to avoid the problems of "Dead Layer" and poor time resolution associated with closed-end type geometries.

^bFWTM is ≤2 x FWHM (Full Width at Tenth Maximum is ≤ twice the Full Width at Half Maximum).

8000 Series Ge(Li) Low Energy Photon Spectrometers.^a
Delivery: stock to 30 days

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Vertical Cryostat Model No.	Horizontal Cryostat ^b Model No.	Diameter (mm)	Drifted Depth (mm)	Active Area (mm ²)	Energy Resolution FWHM ^c (eV) At 5.9 keV (⁵⁵ Fe)	At 122 keV (⁵⁷ Co)
8013-06180	8113-06180	6	5	30	180	485
8013-06185	8113-06185	6	5	30	185	500
8013-06200	8113-06200	6	5	30	200	525
8013-10190	8113-10190	10	5	80	190	490
8013-10210	8113-10210	10	5	80	210	510
8013-10225	8113-10225	16	5	80	225	525
8013-16260	8113-16260	16	5	200	260	525
8013-16275	8113-16275	16	5	200	275	535
8013-16300	8113-16300	16	5	200	300	550
8013-25360 ^d	8113-25360 ^d	25	7	500	360	595
8013-25380 ^d	8113-25380 ^d	25	7	500	380	615
8013-25400 ^d	8113-25400 ^d	25	7	500	400	615
8013-32500 ^d	8113-32500 ^d	32 ^e	7	800	500	750

^aAll systems supplied with 5-mil-thick Be windows and 31-liter dewars except where noted.

^bHorizontal common-vacuum configuration available special order.

^cUsing optimum time constant of 2 to 6 μ s, FWTM $\leq 2 \times$ FWHM. Total system resolution for 5.9 keV ⁵⁵Fe source at 1000 counts/s measured in accordance with IEEE Standard 325 (1971), using ORTEC standard electronics. Guaranteed for one year.

^d10-mil-thick Be window.

^eSpecial order.

Specifications for Cryostats and Dewars

Model No.	Type	Cryostat Weight (kg)	LN ₂ Volume (liters)	LN ₂ Weight (kg)	Dewar Weight (kg)	Typical Holding Time (days)	Total Weight Full of LN ₂ (kg)
80	Vertical Dipstick	2.45	31	24	15	28	41.5
81	Horizontal Dipstick	3.32	31	24	15	28	42.9

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Gamma System

Packard Auto-Gamma Systems Model 5130

Class	Laboratory
Principle of Operation	Automatic gamma system. Fully adjustable pulse-height analyzer, four energy scale ranges. Preset error. Pushbutton digital display. Three-inch NaI(Tl) crystal (2-inch crystal system available).
Sensitivity and Range	Range: Gamma and X-ray investigation
Sampling	Reversible 300 (or 600) sample adapter free changer. Automatic or manual loading.
Performance	Temperature: Refrigerated detector over range of 5 to 15°C Efficiency: ^{125}I efficiency minimum 65% Background maximum 25 cpm
Requirements	Power: 117 or 234 $\pm$ 10% VAC - 50 or 60 Hz Size: 161cm (height 63.5") 105cm (width 41.5") 79cm (depth 31") Weight: 342.4kg (755 lbs)
Features	1) 300 sample unit expandable to 600 sample 2) Modular electronic design for maximum operating flexibility Choice of 1,2, or 3 separate channels of pulse-height analysis

Features (cont.) 3) Empty sample carriers are optically sensed and bypassed
 4) Pedestal-type elevator with anti jam protection
 5) Short group counting
 6) Empty carrier bypass
 7) Molded lead shielding designed for elimination of
 "crosstalk"
 8) Sample level adjust

Cost \$18,140.00

Address Packard Instrument Co., Inc.
 2200 Warrenville Road
 Downers Grove, IL 60515
 (312) 969-6000

Gamma Counter
Packard Model Prias^{T.M.} PGD

Class	Laboratory
Principle of Operation	Benchtop automatic gamma counter. Capacity for tubes 10 to 16mm in diameter. Front panel pushbuttons and digital display for user entry, review and modification of program data. Storage of up to seven different data reduction or counting programs.
Sensitivity and Range	Range: Gamma and X-ray investigation
Sampling	240 sample capacity (408 and 504 sample capacity also available) Multi-user capability Samples loaded in cassettes Sample changer tray is removable
Performance	Temperature: Ambient system Efficiency and Background: ¹²⁵ I efficiency; 65% minimum Maximum background: 50 cpm
Requirements	Power: 117 or 234 $\pm$ 10% VAC; 50 or 60 Hz Size: 72cm (height 28"); 63cm (width 25"); 66cm (depth 26") Weight: 181 kg (400 lbs)

Features

- 1) Pushbutton entry of programming data- up to seven programs can be stored.
- 2) Data output device integral to the system with results printed adjacent to the sample.
- 3) Screening function.
- 4) Preset counting conditions for ^{125}I , ^{57}Co or ^{76}Sr , ^{51}Cr or ^{131}I , ^{22}Na or ^{59}Fe or ^{58}Co .
- 5) RS232 compatible output.
- 6) Power fail protected memory.

Cost \$14,390

Address Packard Instrument Company, Inc.
2200 Warrenville Road
Downers Grove, IL 60515
(312) 969-600

Isotope Calibrator
Picker Model 632-507

Class	Stationary, Laboratory
Principle of Operation	Pressurized, well-type, argon filled, ion chamber, electrometer, dose computing circuit, digital display, calibration
Sensitivity and Range	Energy Range: Gammas from 25 keV to 3 MeV Activity Range: 1 μ Ci to 999 mCi
Performance	Accuracy: $\leq \pm 5\%$ for any isotope Temperature: 12°C to 45°C Measurement Time: <1 sec for samples above 200 μ Ci Repeatability: $\leq \pm 3\%$ for 99m Tc above 100 μ Ci Stability: $\leq \pm 1\%$ drift for $\pm 10\%$ line voltage variation
Requirements	Power: 115 VAC, $\pm 10\%$, 60 Hz (optional 230 VAC, 50 Hz) 28 watts Size: 30.4cmH x 27.9cmW x 43.8cmD (12" x 11" x 17.3") Weight: 21.8 kg (48 lb)
Features	Calibration factor feature allows selection of the following radioisotopes and a variety of syringe and serum vial geometries. ^{18}F , ^{22}Na , ^{24}Na , ^{32}P , ^{51}Cr , ^{57}Co , ^{60}Co , ^{59}Fe , ^{67}Ga , ^{68}Ga , ^{75}Se , ^{85}Sr , $^{87\text{m}}\text{Sr}$, $^{99\text{m}}\text{Mo}$, $^{99\text{m}}\text{Tc}$, $^{113\text{m}}\text{In}$, ^{125}I , ^{131}I , ^{133}Xe , ^{137}Cs , ^{197}Hg , ^{203}Hg , ^{198}Au , ^{226}Ra .
References	Manufacturer's Specifications
Cost	\$3000.00
Address	Picker Corporation Clinical Laboratory Department 12 Clintonville Road Northford, Connecticut 06472 (203) 484-2711 Picker Corporation Marketing Services 6119 Highland Road Cleveland, OH Attn: Mr. Frank Dean

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RAD-GAM, XRA
Spectrometry
Princeton Gamma-Tech
May 1980

Germanium Detectors
Princeton Gamma Tech

Class	Laboratory Spectrometer
Principle of Operation	Ge(Li) and Intrinsic (high purity) germanium detectors, amplifiers and other electronics, cryostats and accessories
Features	Many variations in full systems available. Special detector specifications available on request.
References	Manufacturer's Specifications
Address	Princeton Gamma-Tech Box 641 Princeton, NJ 08540 (609) 924-7310

Specifications For selected detector as follows:

Geometry	Range of Efficiency Relative to 3 x 3 NaI(Tl) at 1.33 MeV	Energy Resolution (keV)		Peak/Compton Ratio at 1.33 MeV	Price
		1.33 MeV	7.6 MeV		
Modified Coaxial	5 to 26	1.7 to 2.5	4.5 to 6.0	20 to 50	\$5000 to \$28,000
True Coaxial	5 to 16	1.8 to 2.6	5.5 to 6.0	16 to 40	\$6000 to \$19,000
Well Detector	5 to 20	2.1 to 2.7		25 to 38	

Standard Grade Coaxial Intrinsic (High Purity) Germanium Detectors,*
Straight Dipstick Cryostat, Pre-amplifier, 17-Liter Dewar

Area (mm ²)	Resolution at 5.9 keV	Resolution at 122 keV	Be Window Thickness	Price
25	155 eV	485 eV	0.0013"	\$ 6,300
25	145	480	0.0013"	7,300
50	175	490	0.0013"	6,800
100	190	495	0.0013"	7,900
200	240	540	0.0023"	8,800
300	300	570	0.006"	9,600
400	350	610	0.006"	10,300
500	400	650	0.006"	11,300
1000	600	800	0.006"	13,300

*Also available as borehole probe, well detector, or in detector
arrays (up to 18 detectors/array).

Semi-Planar Ge(Li) Detectors, SP-Series

Model	Active Area (cm ²)	Drift Depth (mm)	Energy Resolution (eV FWHM at 122 keV)	Preamp Type	Price
LGSP10R	10	10	1000	RT	\$3,400
LGSP10C	10	10	750	CF	3,800
LGSP15R	15	10	1050	RT	4,400
LGSP15C	15	10	800	CF	4,800

Other sizes also available

Extra charges for other cryostats:

Model ED, horizontal-looking dipstick	\$100.
Model HT, horizontal-looking unitary (15-liter)	\$250.
Model DT, down-looking unitary (15-liter)	\$250.

Germanium Well Detector
PGTGe(Li) Well or High Purity Ge Well Detector

Fig. 1. Absolute efficiency curve for a typical Ge(Li) Well Detector with the source placed in the well.

Class	Laboratory
Principle of Operation	A solid state semiconductor operated in reverse bias in order to be gamma spectrum detector. With associated electronics delivers a gamma spectrum.
Sensitivity and Range	γ energy range detected: 20 keV to 10 MeV Typical efficiency (see Fig. relative to 3"x3" NaI @ 1.33 MeV: ~10% to 20% Typical Peak to Compton Ratio (1.33).
Sampling	Continuous when operated
Performance	Usable well dimensions: 1 cm D x 3-4 cm L Detector volumes: 50-100 cm ³ Operating temperature: at spectrometer, liquid nitrogen temperature; high purity Germanium well can be stored or cycled to room temperature Typical Resolutions: 2.1 to 2.8 keV FWHM @ 1.33 MeV
Requirements	Power: Requires high voltage supply Size: Depends on dewar Weight: Depends on dewar } check with manufacturer
Features	Well detectors are supplied in vertical dipstick cryostats with 8" end caps to facilitate shielding

References	Manufacturer's Specifications
Cost	Refer to Manufacturer
Address	Princeton Gamma-Tech Box 641 Princeton, NJ 0854 (609) 924-7310

Gamma Detectors

Quartz Products

**FOUR SEGMENT
WELL-TYPE SCINTIFLEX**

305 YPFE 305

for pair spectrometry operable as an
anti-Compton annulus by series
connexion of segments

Monocrystal NaI(Tl) $\varnothing$ 305 mm
h 305 mm

4 Photomultipliers XP 1031 or RCA 8054

Well $\varnothing$ eff 83 mm

Class	Laboratory
Principle of Operation	NaI(Tl) scintillators and photomultipliers
Sensitivity and Range	Energy Range: 10 keV to 3 MeV, gammas
Performance	Accuracy: About 9% Temperature: -50°C to 50°C Calibration: ¹³⁷ Cs resolution, ⁶⁰ Co peak-to-valley ratio, ⁵⁵ Fe for X rays
Requirements	Scintibloc Crystals (Solids and Well Types) Diameter mm: 32 38 44 51 76 76 127 Height mm: 25 25 51 51 51 76 51 Energy Resolution: 7 - 9% for Cs-137 (661 keV) Cost: \$137.00 - \$1194.00 Scintibloc X Crystals Diameter mm: 25 32 44 76 to 250 Thickness mm: 1 to 5 1 to 5 1 to 5 1 to 5mm Energy Resolution: 45 - 60% for Fe-55 (220 keV) Cost: \$265.00 - \$1194.00 Scintibloc and Scintiflex Crystals (Solids and well-types) Diameter mm: 102 102 127 127 127 152 203 228 292 Height mm: 76 102 76 102 127 102 102 102 102 Energy Resolution: 7.5 - 9.5% for CS-137 (661 keV) Cost: \$1159.00 - \$8500.00 Note: Scintibloc refers to Direct Coupling Scintiflex refers to Demountable Phototube
Features	1. Anti-Compton Annuli (Pair Spectrometer Annulus Option) from 8" x 8" to 12" x 12" 2. Phoswich detector (NaI[Tl]/CsI[Tl]) for Plutonium studies 3. Crystals up to 16" diameter Other scintillating CsI (Tl), CsI (Na), Anthracene, Stilbene, ZnS (Ag) materials.
References	Manufacturer's Specifications
Address	Quartz Products Corporation 688 Somerset Street P.O. Box 628 Plainfield, NH 07061 (201) 757-4545

X-ray Proportional Counters

Reuter-Stokes

Model	RS-P3-1605-261	RS-P3-1605-285
Energy Range	3 to 35 keV	1.5 to 8 keV
% FWHM	10% at 14.4 keV	17% at 5.98 keV 12% at 14.4 keV
Window	Be, 1"D 0.01"Thick, 46 mg/cm ²	Be, 2"L x1.7"W 0.005"Thick, 23 mg/cm ²
Temperature Range	-25°C to +80°C	-25°C to +55°C
Fill Gas	Xe, CO ₂ 1 atm.	97% Xe, 3% CO ₂ 76 cm Hg
Power Required	1500 to 2200V dc.	1500 to 2200 dc.
Size	19.05cm x5.16cm 7.5"L 2.1"D	19.3cm x5.1cm 7.6"L, 2"W
Weight	0.28 kg (0.62 lb)	0.5 kg (1.25 lbs)
Applications	Mossbauer Effect studies	Industrial XRF analysis

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MONITORING

RAD-GAM, XRA
Spectrometry
Reuter-Stokes
Page 2

Cost	\$510.00 base	\$1302.00
Features	Optional gas, windows, and gas pressures	
Model	RS-P3-0803-287	RS-P3-1608 -271
Energy Range	3 to 35 keV	6 keV - 15 keV
%FWHM	11% at 14.4 keV	17% at 5.98 keV
Window	Be, 1"D, 0.005"T	Be, 0.010"T
Temperature Range	-25°C to +80°C	-25°C to +55°C
Fill Gas	Kr, CO ₂ 152 cm	Xe, CO ₂ 76 cm
Power Required	1700 → 2200V	2000-2500 V dc.
Size	14.3cm x2.54cm 5.5"L, 1"D	26.7cm x2.5cm x8.3cm 10.5"L, 1"W, 3 1/2"D
Weight	0.1 kg (.25 lb)	0.5 kg (1 lb)
Applications	Spectroscopy diffraction	Industrial XRF
Cost	\$365.00	
Features	Optional gas, windows, and gas pressures	
Remarks	Other detectors meeting different specifications are available. Please contact the manufacturer to learn about these.	
Address	Reuter-Stokes 18530 S. Miles Parkway Cleveland, Ohio 44128 (216) 475-3434

Gamma Ionization Chambers
Reuter-Stokes RS-C4-1606 Series

Class	Portable
Principle of Operation	Ionization chamber
Sensitivity and Range	Energy Sensitivity: 0.4 MeV to 7 MeV

Typical Operating Characteristics

Model No.	Sensitivity (amp/R/hr)	Flux Range (R/hr)
RS-C4-1606-209 (456 cm Hg-Xe)	7×10^{-10}	5×10^3 to 5×10^2
RS-C4-1606-203 (760 cm Hg-N ₂)	1.2×10^{-10}	10^{-2} to 10^4
RS-C4-1606-205 (76 cm Hg-N ₂)	1.2×10^{-11}	10^{-1} to 10^5
RS-C4-1606-207 (6.8 cm Hg-N ₂)	1.2×10^{-12}	10^0 to 10^6
RS-C4-2510-101 (xenon)	3.5×10^{-9}	500×10^{-6} to 25×10^1

Flux range: 5 mR/hr to 10^6 R/hr

Sampling	Continuous
Performance	Accuracy: $\pm 3\%$ Tracerable to NBS - sensitive volume: 160cm ³ Temperature: 200°C max
Requirements	Power: 2000 V dc max Size: 23cmL x 5cm D (9.1"L x 2"D) Weight: 0.34 kg (0.75 lb)
Features	Stainless steel walls and all seals are ceramic to metal Many other detectors available for x-ray and gamma

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References	Manufacturer's Specifications
Cost	\$1090.00 - \$1525.00
Address	Reuter-Stokes 18530 S. Miles Parkway Cleveland, Ohio 44128 (216) 475-3434

Environmental Gamma Monitor
Reuter-Stokes Model RSS-111

Class	Stationary or portable field instrument
Principle of Operation	High-pressure spherical argon-filled ionization chamber, solid-state electrometer. Separate housing for control circuitry, recorder, panel meter, batteries.
Sensitivity and Range	Sensitivity: 20mV/ μ R/hr Energy range: 50 keV to 5 MeV (gammas) Flux range: 1-500 μ R/hr (standard, to 5000 μ R/hr optional)
Sampling	Continuous
Performance	Accuracy: $\pm 5\%$ at 10 μ R/hr Temperature: -24°C to $+54^{\circ}\text{C}$ Recording: Strip-chart recorder, optional magnetic-tape or telemetry readout Response time: 5 sec. Humidity: 0 to 100% (waterproof)
Requirements	Power: Ni-Cd rechargeable batteries for electrometer; 300 volt power cell for ion chamber; ac operation also available Size: Two weatherproof housings (1 ft cubes = 30 cm cubes) Weight: Sensor housing 23 lb (10.5 kg), tripod 6 lb (2.7 kg), remote housing 38 lb (17.2 kg)

Features	<ol style="list-style-type: none">1. Spherical ion chamber yields flat response.2. Solid-state electrometer with MOSFET circuitry.3. Weatherproof construction.4. Optional ac power supply hookup.5. Optional dose integrator, magnetic tape output and tape reader.6. Automatic range changing.
References	Manufacturer's Specifications
Cost	\$8727.00 with strip chart recorder; digital displays and parallel digital output; optional readouts additional
Address	Reuter-Stokes 18530 South Miles Parkway Cleveland, OH 44128 (216) 475-3434

Area Monitors

Model FML-5A
Model FML-5B

Class	Rack mount or case
Principle of Operation	Geiger detectors, log count ratemeter, high level and system failure alarms
Sensitivity Range	FML-5A: 6 decades 0.01 mR/hr to 10 R/hr standard other ranges to 1000 R/hr FML-5B: 4 decades 1 mR/hr to 1 R/hr standard or other 4 consecutive ranges
Sampling	Continuous
Performance	Energy dependence: $\pm 20\%$ from 100 keV to 1.2 MeV
Requirements	Power: 110 VAC, 60 Hz Size: 10 1/4"W x 8 3/4"H x 13 1/2"D Weight: console plus detector 21 lbs.
Features	All plug-in modular (can add channels) Anti-saturation High and low level alarms Rack mount or case Rugged, heavy duty probes with energy compensation Local or remote monitoring up to 500 ft. cable Activation of movable check source in probe (option)
References	Manufacturer's Specifications

RAD-GAM,XRA
Stationary Monitor
Technical Associates
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Cost	FML-5A: \$885.00 includes probe, first channel \$630.00 per additional channel
	FML-5B: \$850.00 includes probe, first channel \$630.00 per additional channel

Address Technical Associates
7051 Eton Avenue
Canoga Park, CA 91303
(213) 883-7043

High Range Underwater Monitor
Technical Associates Model CP-MU

Class	Portable, handheld	
Principle of Operation	Remote air ion chamber with MOSFET amplifier, meter readout	
Sensitivity and Range	0 to 1 kr/hr 0 to 10 kr/hr 0 to 10^5 R/hr 0 to 10^6 R/hr	optional other ranges (lower)
Sampling	Continuous	
Performance	Accuracy: $\pm 15\%$ FS Time Constant: 2 seconds	
Requirements	Power: Single "D" cell Size: 22.9cmL x 14.0cmH x 10.2cmW (9"L x 5 1/2"H x 4"W) excluding handle (cable (probe) length: 60 ft. (longer optional) Weight: 1.8 kg (4 lbs)	
Features	Underwater monitoring of reactor spent fuel etc. or hot cell monitoring to very high levels. Stability and accuracy (adjustment pot for each range) Portability

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Survey
Technical Associates 2
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References	Manufacturer's Specifications
Cost	\$1380.00
Address	Technical Associates 7051 Eton Avenue Canoga Park, CA 91303 (213) 883-7043

NaI(Tl) Gamma Spectrometry Systems

Teledyne Isotopes

Typical Assembly

Class	Laboratory
Principle of Operation	NaI(Tl) crystals, photomultipliers, mounts, and assembled systems for photon spectrometry.
Specifications	<u>NaI(Tl) Crystals</u> (1) Wide variety of standard cylindrical crystals, from 1"Dx1/2"L to 9"x4". Custom sizes available. (2) Well shapes and thin crystals also available. (3) Table shows a few representative crystals (4) <u>Response Time</u> can be pre-determined to satisfy customer needs through the use of specific photomultiplier tubes

Designation	Crystal Diameter		Crystal Thickness		Crystal Price
	(cm)	(in.)	(cm)	(in.)	
S-42	2.54	1"	1.27	1/2"	\$ 75.00
S-44	2.54	1"	2.54	1"	75.00
S-64	3.81	1 1/2"	2.54	1"	90.00
S-88	5.1	2"	5.1	2"	250.00
S-1212	7.6	3"	7.6	3"	700.00
ST-41-A	2.52	1"	1 mm		100.00
ST-46-A	2.54	1"	6 mm		100.00
ST-81-A	5.1	2"	1 mm		160.00
ST-86-A	5.1	2"	6 mm		160.00

Assemblies

- (1) Many types of assemblies available
- (2) All "Integral Series" assemblies have photomultiplier attached to NaI(Tl), all hermetically sealed
- (3) Magnetic shield included
- (4) Tube base available as extra option
- (5) Aluminum housing (0.020") standard: copper or stainless steel also available
- (6) Phototubes used are RCA 6342AVI, 4524, 2060; EMI 9805, 9759; and Amperex XP 2010.
- (7) Well-type and other geometries also available

Designation	Crystal Diameter (in.)	Crystal Thickness (in.)	Price
S-64-I	1 1/2"	1"	\$235.00
S-78-I	1 3/4"	2"	300.00
S-84-I	2"	1"	260.00
S-88-I	2"	2"	325.00
S-1208-I	3"	2"	675.00
S-1212-I	3"	3"	790.00

References

Manufacturer's Specifications

Address

Teledyne Isotopes
50 Van Buren Avenue
Westwood, NJ 07675
(201) 664-7070

NaI(Tl) Spectrometer Systems

Tennelec, Inc.

Class	Laboratory Spectrometers
Principle of Operation	NaI(Tl) crystal, photomultiplier tube for gamma spectroscopy. (See Page 2 of this note)
Features	Electronic accessories also available. Individual NaI(Tl) crystals also available. All assemblies have magnetic shielding, hermetic seals.
References	Manufacturer's Specifications
Address	Tennelec, Inc. P.O. Box D Oak Ridge, TN 37830 (615) 483-8405

Specifications

Crystal Size	Model Type	Performance Data	Photomultiplier Tube	Price
Diameter Thickness				
<u>Thin Window NaI(Tl) X-Ray Detectors</u>				
(55Fe Peak/Valley Ratio, 5.9 keV)				
25mm x 1.5mm (1" x 0.2")	TD 25 SX 1/5	15:1	Philips XP1010	\$ 430.00
32mm x 1.5mm (1.25" x 0.2")	TD 32 SX 1/5	15:1	Philips XP1010	500.00
44mm x 1.5mm (1.75" x 0.2")	TD 44 SX 1/5	15:1	Philips 153 AVP 02	600.00
<u>Standard NaI(Tl) Assemblies</u>				
(137Cs Energy Resolution, 662 keV)				
25mm x 25mm (1" x 1")	TD 25 S 25	8.5%	RCA 6199	\$ 140.00
32mm x 25mm (1.25" x 1")	TD 32 S 25	8.5%	RCA 6199	150.00
38mm x 25mm (1.5" x 1")	TD 38 S 25	8.5%	RCA 6342A	165.00
44mm x 51mm (1.75" x 2")	TD 44 S 51	8.5%	RCA 6342A	270.00
51mm x 51mm (2" x 2")	TD 51 S 51	8.5%	RCA 6342A	320.00
76mm x 51mm (3" x 2")	TD 76 S 51	8.5%	RCA 8054	700.00
76mm x 76mm (3" x 3")	TD 76 S 76	8.5%	RCA 8054	850.00
102mm x 102mm (4" x 4")	TD 102 SE 102	9.0%	RCA 8055	1490.00
127mm x 127mm (5" x 5")	TD 127 SE 127	10.0%	RCA 8055	2290.00
<u>Well Type Assemblies</u>				
(137Cs Energy Resolution, 662 keV)				
44mm x 51mm (1.75" x 2")	TD 44-SP-51	9.0%	RCA 6342A	\$ 330.00
51mm x 51mm (2" x 2")	TD 51-SP-51	9.0%	RCA 6342A	390.00
76mm x 76mm (3" x 3")	TD 76-SP-76	9.0%	RCA 8054	990.00
102mm x 102mm (4" x 4")	TD 102-SPE-102	10.0%	RCA 8055	1645.00
127mm x 127mm (5" x 5")	TD 127-SPE-127	11.0%	RCA 8055	2645.00

Geiger-Mueller Detectors

Texas Nuclear Model 9361

Class	Detectors
Principle of Operation	Geiger Mueller Detector
Sensitivity and Range	Plateau length: ~200 volts Plateau slope maximum: ~6%/100 volts Pulse output: 2 to 20 volts
Sampling	Continuous
Performance	Anode material: 0.0051cm (0.002") tungsten Insulators: Alumina ceramic Cathode: Bismuth screen Envelope material: Stainless steel Operating temperature: Up to 150°C Filling gas: Specially purified organic quenched gas
Requirements	Size: 2.54 cm (1.0") outside diameter; 19.53cm (8 11/16") length: 15.2cm (6") active length Operating voltage: ~ 1100 volts Weight: 1.36 kg (3 lbs)
Remarks	Basic design available in other active lengths and large diameters
Cost	Refer to factory
Address	Texas Nuclear Division Ramsey Engineering Company P.O. Box 9267 Austin, TX 78766 (512) 836-0801 Ext. 311

GM Tubes
Partial Listing
Manufactured by TGM

Class	Portable
Principle of Operation	Geiger Mueller tubes for replacement purposes and new equipment design.
References	Manufacturer's Specifications
Address	TGM Detectors Inc. 34 Bear Hill Road Waltham, MA 02154 (617) 890-2090

Geiger-Mueller Detectors
 α β γ End Window, Cylindrical GM Tubes

Model	Cathode Wall		Window Thickness (mg/cm ²)	Operating voltage	Recovery time(μ sec)	Remarks	Revised Price, April 1980
	Inside diameter	Active length					
A201	20mm(.781")	102mm(4.0")	1.4-2.0	900	100 μ sec	detects x-ray	\$80.00
K201	20mm(.781")	102mm(4.0")	1.4-2.0	1300	100 μ sec	detects x-ray	90.00
N201	20mm(.781")	102mm(4.0")	1.4-2.0	900	100 μ sec		55.00
N202	3.18mm(.125")	7.94mm(.313")	1.0-1.2	750	50 μ sec		65.00
N204/ 7618	20mm(.781")	12.7mm(0.5")	1.4-2	900	150 μ sec		70.00
N210	29.4mm(1.157")	48mm(1.875")	1.4-2	900	200 μ sec		56.00
N210BNC	29.4mm(1.157")	48mm(1.875")	1.4-2	900	200 μ sec		66.00
N222	6.35mm(0.25")	33.5mm(1.32")	1.4-2	900	100 μ sec		70.00
H220	29.4mm(1.157")	25.4mm(1.0")	1.4-2	1300	100 μ sec	organic quenched	60.00
H220/ 7840	29.4mm(1.157")	12mm(0.50")	1.8-2.2	1400	80 μ sec	organic quenched	60.00

γ Cylindrical GM Tubes

Model	Cathode Wall		Thickness	Operating voltage	Recovery time(μ sec)	Remarks	Revised Price, April 1980
	Inside diameter	Active length					
N302	6.4mm(.250")	7.9mm(.310")	.51mm(.020")	900	20 μ sec		\$35.00
N308	3.18mm(.125")	12.7mm(.50")	.94mm(.037")	600	50 μ sec		25.00
N309	15.2mm(.60")	17.5mm(2.687")	.23mm(.009")	900	80 μ sec		30.00
N310	15.2mm(.60")	*148mm(5.81")	.23mm(.009")	900	80 μ sec		35.00
N314	18.4mm(.726")	*148mm(5.81")	.31mm(.012")	900	110 μ sec		35.00
N315	6.4mm(.250")	*96mm(3.8")	.51mm(.020")	700	70 μ sec	Pt plated wall	120.00 (NP315)
NP358	15.2mm(.600")	*108mm(4.25")	.23mm(.009")	900	100 μ sec	Pt plated wall high temperature	150.00
NP334	18.4mm(.726")	*108mm(4.25")	.31mm(.012")	900	130 μ sec	Pt plated wall high temperature	160.00
TGM-16	25mm (.980")	305mm(12")	.51mm(.020")	1200	60 μ sec	organic quenched	80.00

α β γ End Window, Pancake GM Tubes

Model	Cathode Wall		Window Thickness (mg/cm ²)	Operating voltage	Recovery time(μ sec)	Remarks	Revised Price, April 1980
	Inside diameter	Active length					
N1002/ 8767	44.5mm(1.75")	10.7mm(.42")	1.6-2	900	30 μ sec		\$ 90.00
N1002/ 2x	44.5mm(1.75")	10.7mm(.42")	1.6-2	900	30 μ sec	dual window	250.00
N1003	29.4mm(1.157")	9.5mm(.375")	1.6-2	900	40 μ sec		70.00
N1005	44.5mm(1.75")	10.7mm(.42")	1.6-2	900	40 μ sec		125.00
N1006	44.5mm(1.75")	10.7mm(.42")	1.6-2	900	30 μ sec		115.00
N1007	20.0mm(0.781")	9.0mm(.36")	1.6-2	900	30 μ sec		60.00

β γ Thin Wall Cylindrical GM Tube

	Cathode Wall		Thickness (mg/cm ²)	Operating voltage	Recovery time(μ sec)	Remarks	Revised Price, April 1980
	Inside diameter	Active length					
N106	15.2mm(.60")	89mm(3.5")	30-40	900	75 μ sec		\$ 32.00
N106HT	15.2mm(.60")	89mm(3.5")	30-40	900	75 μ sec	200°C	42.00
N107	15.2mm(.60")	227mm(8.94")	30-40	900	75 μ sec		50.00
N107HT	15.2mm(.60")	227mm(8.94")	30-40	900	75 μ sec	200°C	60.00
N108	15.2mm(.60")	144mm(5.68")	30-40	900	74 μ sec		35.00
N112/ 6833	18.4mm(.726")	54.8mm(2.156")	30-40	900	150 μ sec		33.00
N119HS	18.4mm(.726")	142.9mm(5.675")	30-40	900	150 μ sec		65.00
N114/ OCD-D-103	15.2mm(.60")	19mm(.75")	30-40	900	60 μ sec		30.00
N115	4.8mm(.190")	6mm(.25")	80-100	600	20 μ sec		30.00
N116	4.8mm(.190")	16mm(.625")	80-100	600	28 μ sec		30.00
N116-2	4.8mm(.160")	16mm(.625")	30-40	600	28 μ sec		35.00
N117	7.9mm(.310")	28mm(1.102")	30-50	600	45 μ sec		35.00
TGM-6M	15.2mm(.60")	89mm(3.5")	30-40	900	50 μ sec	organic quenched	37.00

Automatic Gamma Counter
Tracor Analytic Model 1185

Class	Laboratory
Principle of Operation	A well-type scintillation detector and lead shield and a NaI(Tl) crystal with nearly 4π detection geometry.
Sensitivity and Range	Sensitivity expressed in terms of E^2/b ratio is at least 200 with a 2" crystal Dynamic Range: 100:1
Sampling	300 sample capacity, counted individually
Performance	Typical efficiency is 80% with 25 CPM background when a 2" crystal is used Temperature:
Requirements	Power: 115V $\pm$ 10V, 60 Hz or 230V, 50 Hz Size: 162.6cmH x111.8cmW x85.1cmD (64"H x44"W x33.5"D) Weight: 907.2 kg (2000 lbs)
Features	Automatic background subtraction, two or three analysis channels with both preset and adjustable windows, automatic calibration, automatic correction for spillover in dual labeled experiments, ratemeter. RIA data reduction available.
References	Manufacturer's Specifications
Cost	
Address	Tracor Analytic, Inc. 1842 Brummel Street Elk Grove Village, IL 60007 (312) 364-9100

Area Gamma Alarm Monitor

Harwell Type 0467

Class	Laboratory, wall mounted
Principle of Operation	A halogen quenched GM tube mounted internally, or up to 10m external, with a log ratemeter and associated alarm equipment.
Sensitivity and Range	Dose rate range: 0.1mR/hr to 100 mR/hr, logarithmic. The meter is held at full scale when the GM tube saturates on high radiation overload.
Sampling	Continuous
Performance	Operating temperature: -10°C to 45°C Alarm trips level: adjustable over the full monitor range. Fail safe alarm: normal background level radiation used to check correct operation of monitor. If the monitor reading falls below this adjustable level, then the alarm is initiated as for a genuine alarm with the exception that it cannot be switched by the cancel button.
Requirements	Power: 24V $\pm$ 20% DC, positive with 240V A.C. 50 Hz Size: 30cmx18.5cmx18cm Weight: 5.5 kg

Features	Fail safe alarm with level adjustable Audible and visible high radiation alarm tripped by preset trip level. Three monitors may be interconnected to operate the alarm on a 2 out 3 principle when necessary to reduce the incidence of false alarm.
References	Manufacturer's Specifications
Cost	On application
Address	AERE, Harwell Oxfordshire, OX 11 0RA, England Tele: (0235, 2414) Attn: R. H. W. Sherwood, Bldg. 347.3 Electronics and Applied Physics Ext. 4766 R. G. Powell, Bldg. 329 Marketing and Sales Dept. Ext. 2942

Si(Li) X-Ray Detector/Cryostat

United Scientific Model 101

Class	Laboratory (101)																												
Principle of Operation	Cryogenically cooled Si(Li) detector and pulsed optical feed-back preamplifier. Vertical dipstick (17½) dewar																												
Sensitivity and Range	Element Range: Sodium Z = 11 to americium Z = 95 Sensitivity: 1 ppm = 100% full scale RESOLUTION AT 5.9 keV (FWHM in eV)																												
<table><tr><th>10mm² area</th><th>30mm² area</th><th>80mm² area</th><th>200mm²</th><th>300mm²</th><th>Price</th></tr><tr><td>144</td><td>148</td><td>162</td><td>225</td><td>250</td><td>\$9090</td></tr><tr><td>148</td><td>154</td><td>170</td><td></td><td></td><td>8090</td></tr><tr><td>156</td><td>162</td><td></td><td>260</td><td>290</td><td>7090</td></tr></table>						10mm ² area	30mm ² area	80mm ² area	200mm ²	300mm ²	Price	144	148	162	225	250	\$9090	148	154	170			8090	156	162		260	290	7090
10mm ² area	30mm ² area	80mm ² area	200mm ²	300mm ²	Price																								
144	148	162	225	250	\$9090																								
148	154	170			8090																								
156	162		260	290	7090																								
Sampling	Continuous																												
Performance	Accuracy: 0.1% Temperature Range: 10-35°C Readout Time: Instantaneous																												
Requirements	Power: ±110V ac, 2A Size: 91 cm H x 48 cm W (36" H x 17" W) Weight: About 45 kg (100 lb)																												
Features	1) Detector mounted in shock absorber 2) Other type dewars available 3) Electronic components available, also.

References	Manufacturer's Specifications
Cost	See table
Address	United Scientific Corporation, Analytical Instruments Division 1400 Stierlin Road P.O. Box 1389 Mountain View, CA 94042 (415) 969-9400

Area Monitor
Xetex Model 501A-1,2,3

Class	Laboratory
Principle of Operation	Halogen quenched GM detector, digital logic, digital alarm level set, LED display
Sensitivity and Range	Range: 0.1 to 99.9 mR/hr (Model 501A-1), .01 to 9.99 R/h (Model 501A-3) 1 to 999 mR/hr (Model 501A-2) Alarm Range: 1 to 99 mR/hr in 1 mR steps (501A-1) 10 to 990 mR/hr in 10 mR steps (501A-2), .1 to 9.9 R/h in 100 mR/h steps (501A-3) Gamma and X-ray
Sampling	Continuous (8 sec update period-1) (6 sec update period-2) (2.5 sec update period-3)
Performance	Accuracy: $\pm 15\%$ referred to Cs-137 Energy dependence: $\pm 15\%$ of true doserate from 70 keV to 1.3 MeV Operating temperature: -20° to 50°C
Requirements	Power: 115/230 V, 50/60 cycle, 15W Size: 10.8 cm H x 16.5 cm W x 20.3 cm D (4 1/4" x 6 1/2" x 8") Weight: 2.54 kg (5.6 lbs) w/battery
Features	Autorangeing, digital display, standby battery power, alarm tone and flashing display; all of the above models available with a remote probe.
References	Manufacturer's specifications

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Cost \$730.00 (local detector), \$790.00 (remote detector)

Address XETEX, Inc.
 660 National Ave.
 Mountain View, CA 94043
 (415) 964-3261

Accelerator Monitor

XETEX Model 303A

Class	Portable, subminiature
Principle of Operation	Solid State detector and associated electronics with meter readout
Sensitivity and Range	Radiation detected: X and γ rays Range: 0-10, 0-100, and 0-1000 mR/hr Zero drift: Less than 5% per hour on most sensitive scale (0-10mR/hr) Response time: four seconds or less
Sampling	Continuous
Performance	Operating temperature: 0-50°C Energy dependence: $\pm 15\%$ Pulse Capability: responds linearly to pulse peaks in excess of 1000 R/hr Accuracy: $\pm 10\%$ of full scale referred to ^{137}Cs
Requirements	Power: Battery 9V transistor radio type (Mallory MN 1604); Life > 200 hr Size: 7.14 x 15.5 x 3.89 cm (2.81 x 6.09 x 1.53 in.) Weight: 440g (15.5 oz.) including battery
Features	a) Conducting film on inside of box provides electro-static shielding to prevent RF pickup b) Calibration is facilitated by a single externally accessible multi-turn potentiometer for each range c) Power off and battery check positions provided on main switch d) Low power integrated circuits are used throughout

References	Manufacturer's Specifications
Cost	\$362.00
Address	Xetex, Inc. 660 National Ave. Mountain View, CA 94043 (415) 964-3261

High Level Probe
XETEX Models 302B

Class	Portable, handheld
Principle of Operation	Uses halogen quenched GM tube in waterproof housing,digital logic and LED readout
Sensitivity and Range	X, γ rays Range: 0.01-1000 R/h
Sampling	Continuous
Performance	Accuracy: $\pm 15\%$ referred to ^{137}Cs Energy Dependence: $\pm 15\%$ from 70 keV to 1.3 MeV Response Time: 3 seconds or less Operating Temperature: -20° to 50°C
Requirements	Power: Four size AA Nicad batteries, 8 hr life Size: Extended Probe ~ 3.66m (12 feet), collapsed probe - 1.14m (45 inches) Weight: 2.27 kg (5 lbs)
Features	Autoranging Large digital display with overrange indicator and battery check Telescoping probe with positive locks Rugged, lightweight
References	Manufacturer's Specifications
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1. Introduction

In this section we shall discuss gamma ray spectrometers, which are instruments for detecting and analyzing gamma rays in terms of intensity as a function of energy. Instrumentation of this type has played a leading role in many important research areas. Although much of this instrumentation is widely used for other purposes, that which concerns us here is instrumentation specifically for low-level measurements of gamma-emitting radionuclides in environmental-type samples. The two classes of instrumentation which will dominate this discussion are scintillation counter spectrometers (e.g., sodium-iodide), and semiconductor spectrometers (e.g., germanium, silicon). Both laboratory analyzers and portable field instruments will be discussed.

Although there is an important physical distinction between gamma rays and x rays, they are both types of electromagnetic radiation, and there is considerable overlap in their wavelengths. Gamma rays are the result of nuclear transitions while x rays are photons from the change in energy of electrons. Whenever either type of photon group is detected and its energy and intensity measured in order to identify a radioactive isotope in an environmental medium, it falls within the scope of this section. The reader may in many cases be able to substitute the word x ray for gamma ray in this text; therefore a separate section on x-ray spectrometry has not been necessary.

We shall exclude measurements of gamma ray beams, or other radiation fields when the aim of the measurements is other than to identify discrete energy groups for radionuclide analysis. External-radiation-field measurements are treated in "X and Gamma Radiation Monitoring Instrumentation", Section III.D.

We shall also not discuss procedures for the separation of various gamma-emitting radionuclides from each other by analytical chemical techniques. Such procedures are discussed for many radionuclides in Division IV of this volume. The National Academy of Sciences has published a series of documents (Ref. 1) discussing and reviewing the pertinent radiochemistry of certain radionuclides. The reader is referred to these publications in order

to solve any problem that might arise calling for chemical separations. Because gamma radiation is relatively penetrating, there is rarely a need to chemically prepare the sample in order to count the gamma radioactivity (as is often needed with alpha emitters and some low energy beta emitters). Generally speaking, such procedures are used mainly to allow measurements of one isotope in the presence of a much larger background of others in order to achieve a greater sensitivity and sometimes specificity.

The uses of gamma spectroscopy have increased over the past decade in response to the needs of the nuclear power industry and of regulatory agencies that monitor the nuclear facilities and their environmental residues. This expanding use derives from the fact that many environmentally important radionuclides from the fission process are gamma emitters. There are of course radionuclides of interest that are primarily beta emitters (^3H , ^{14}C , ^{85}Kr) or alpha emitters (transuranics). The measurements of beta and alpha radioactivity are discussed in other sections of Division III. Gamma spectroscopy is useful in a number of ways for performing environmental radiation surveys. A gamma spectrometer can be used as a laboratory device where samples are counted, also as a field device for performing in-situ analysis at a specific site for determination of background composition and man-made contributions from specific radionuclides. Additionally, the gamma spectrometer can be used at the source of radioactivity (such as a nuclear power reactor) in order to identify specific radionuclides in gaseous or liquid effluents. Since the physical requirements on the spectrometers are different, we will briefly discuss these uses later in this section. We will first review the basic considerations in the measurement of specific gamma rays by energy.

2. Physical Considerations

The term gamma ray is used to describe the electromagnetic emissions due to nuclear transitions. The term x ray describes the similar emissions from electron transitions. Both gamma rays and x rays are photons, the massless carriers of the electromagnetic field. Gamma and x-ray photons have higher energy than those of visible

light. The energy ranges of the two types overlap: gamma rays range from a few keV to several MeV and x rays from fractions of one keV to about 100 keV.

We shall not go into detail here about all the many features of x and gamma emission. However, two features are essential: first, the photons under discussion typically have discrete characteristic energies; and second, the interactions as these photons traverse matter are not at all continuous but are discrete events. In the detector material, a photon interacts once (or twice, less likely) with almost all or almost no energy transfer in each event. Hence it is not correct to characterize them by their "range", but rather by their "mean absorption length".

One common way of characterizing the interaction probability of photons with matter is the absorption coefficient, commonly denoted by μ . Given an incident intensity I_0 , the intensity $I(x)$ after a distance dx changes by an amount dI , proportional to:

$$1. \quad dI(x) = -I(x) \mu dx$$

$$2. \quad \frac{dI(x)}{I_0} = -\mu dx$$

$$3. \quad \ln \frac{I(x)}{I_0} = -\mu x$$

$$4. \quad I(x) = I_0 e^{-\mu x}$$

The units of μ are inverse length (cm^{-1}) if x is expressed in length (cm); more commonly, x is expressed in g/cm^2 and μ in cm^2/g .

a. Interactions of Photons with Matter

In the energy region from a few keV to a few MeV, three important types of interactions occur as photons traverse matter. In the order of their importance with increasing energy, these are: the photoelectric effect, Compton scattering, and electron-positron pair production. We cannot discuss these here in depth; the reader is referred to several excellent reference works (Refs. 2-6). However, a basic understanding of how these interactions manifest themselves, and of their energy dependencies, is crucial to any discussion of gamma spectrometry.

The photoelectric effect is the absorption of a photon by an atom and the simultaneous ejection of an electron, with an energy equal to the photon energy minus whatever binding energy is involved. This effect is significant at very low energies such as for visible light ($\sim$ a few eV); this is the basis for photocells. Here, we are concerned with higher energies, in the region above a few keV. At these energies, the cross-section for photoelectric effect falls off rapidly with energy, as shown in Fig. 1 (from Ref. 7). The dependence on the atomic number Z of the material goes as approximately the fifth power (Z^5), so that an important premium is gained by the use of high- Z materials for detectors. The energy dependence is ($Z^5/E^{7/2}$). Once a photoelectric interaction occurs, all of the photon energy is absorbed and the photon itself disappears. The ejected electron, stopping with its own short range, causes ionization that can be detected and measured.

The Compton effect is the scattering of a photon by an atomic electron. Depending upon the angle of scattering,

Fig. 1. Linear absorption coefficients (in cm^{-1}) versus gamma-ray energy for silicon, germanium, and sodium iodide. (Reproduced from Capellani and Restelli, Ref. 7, with permission of J. Wiley and Sons, © 1968).

the scattering process transfers a variable fraction of the incident photon energy to the target electron, with the remaining energy carried away by another outgoing photon of lower energy. Figure 2 (from Ref. 8) shows the spectrum of electron energies from Compton scattering of 0.51, 1.20 and 2.76 MeV incident photons. The outgoing electron's subsequent deposition of energy by ionization can be detected. The Compton cross-section is proportional to Z , and varies as a function of energy as seen in Fig. 1. The angular distribution of the Compton electrons becomes more and more forward-peaked as the energy increases.

Pair production occurs when a photon, interacting in the electromagnetic field of a nucleus, transforms its energy into the materialization of a positron-electron pair. The energy transferred to the recoiling nucleus is very small, with essentially all of the photon energy going into the e^+/e^- pair. The energy threshold for this process is 1022 keV, the sum of e^+ and e^- rest-mass energies. The

Fig. 2. Energy distribution of Compton electrons as a function of electron energy for various incident gamma-ray energies E . (Reproduced from Davisson and Evans, Ref. 8, with permission of the American Physical Society, © 1952).

Fig. 3. Idealized pair-production process followed by escape of both photons from e^+ annihilation.

cross-section rises quickly above threshold, becoming dominant over Compton scattering above about 10 MeV (Fig. 1). Its Z dependence is approximately $(Z^2 + Z)$. In the energy region of interest for most radionuclide environmental analyses, pair production is not a mechanism useful for sensitively detecting and measuring photon energies, although techniques which rely upon the pair-production process do exist (e.g., pair magnetic spectrometers). Both the electron and the positron liberated in pair production lose energy (and stop) by ionization, and the radiation from the positron's ultimate annihilation can be a signature useful for detection.

A physical effect which can limit system performance is the escape of annihilation quanta after pair production. This effect can be seen schematically in Fig. 3. Suppose that after pair production by a photon of energy E , both the e^+ and the e^- stop, depositing all of their kinetic energy into ionization. The positron then annihilates into two 511-keV gammas. If one or both of these escape undetected, the net energy deposited in the detector will be $(E - 511 \text{ keV})$ or $(E - 1022 \text{ keV})$ resulting in peaks at these energies. Such 'single escape' and 'double escape' peaks are sometimes very prominent, especially for small-sized detectors measuring high-energy gammas.

b. Characteristics of Photon Emission

There is a wide range of phenomena in which photons (gamma rays or x rays) are emitted by atoms or nuclei. Here, we are concerned with those monoenergetic photons useful for characterizing a radioactive nucleus.

The most familiar phenomenon is gamma emission. This occurs when an excited nucleus emits a photon as it makes a transition to a lower energy level. Usually, a parent nucleus has previously undergone a transmutation from alpha or beta decay, leaving the daughter in an excited state.

As a result, gamma emission is characterized by other features common to all radioactivity. Specific radionuclides have a definite half-life associated with them which, when coupled to energy discrimination, helps to make an identification. While most gamma rays occur singly, there are many important cases in which two or more gammas emerge in sequence, as the nucleus undergoes successive transitions. Usually, the times between these multiple transitions are much shorter ($\leq$ picoseconds) than instrumental resolving times, and the multiple emissions can be thought of instrumentally as being in 'time coincidence'. Such coincidences, where appropriate, can be measured and used as unique signatures to identify particular decay schemes. Coincidences between a gamma ray and another particle such as a beta ray may also be useful in a similar way, where appropriate.

For many alpha-emitting high-Z elements, the alpha decay is followed by both X radiation and gamma radiation, as transitions occur in the electronic and nuclear state of the daughter atom. The X radiation, if sufficiently energetic, can provide a signature for the radioactive alpha decay. Usually, the K x rays are detected. For the elements heavier than ones with $Z \geq 80$, such as uranium, the L x rays are in the 10 to 20 keV range, which is near the lower end of the energy range for the type of instruments to be discussed here.

3. Basic Measurement Considerations

In making accurate and sensitive measurements of photon energies in the keV to MeV energy range, a number of important considerations must be taken into account. Ideally, of course, one would like a system that: detects every photon with 100% efficiency; determines its energy with high resolution; has no spurious background indistinguishable from real events; covers the energy range of interest; accommodates all interesting samples; and costs very lit-

tle to purchase, operate, and maintain. Field instruments should be rugged and simple.

Actual instruments, of course, make compromises with all of these items. Inevitably, as well, the better systems are the more expensive. Yet precisely because the typical instrument user is trading one parameter for another in his choice of instrumentation (e.g., efficiency vs. resolution), it is important to determine the nature of the trade-offs and their interconnections.

The key parameters are efficiency, resolution, background, energy range, sample capability, and costs. Here these will be discussed in a general way, to provide background for subsequent discussions.

Usually, environmental radionuclide measurements must be made on minute concentrations. This places severe operational requirements on the instrumentation. Other restraints are imposed by the type and form of sample. The media which require sampling are numerous. Some are directly ingestible (water, food); some are materials used in monitoring direct human exposure (air filters); many are media making up the pathways through the environment to man (soil, animal tissues, plants); some are biomedical specimens (bioassay, whole-body determinations). It is therefore difficult to generalize about the levels of concentrations which require measurement, and we shall not attempt to do so here. However, many of these measurements are at levels requiring sensitivities that approach the limit available with the best systems. Hence there is nearly always a premium on sensitivity, and this is coupled with resolution requirements when a minor activity must be identified in the presence of larger activities.

We discuss here important characteristics of detection and measurement techniques: efficiency, sensitivity, and background. We subsequently examine various techniques in detail, then briefly describe some aspects of data handling and analysis.

a. Efficiency

Figure 4 (from Ref. 4) shows a typical energy spectrum as detected by a high-resolution Ge(Li) detector. The radionuclide shown is aluminum-28, which, although not of consequence in

Fig. 4. Spectrum at aluminum-28 as measured with a Ge(Li) detector. (Reproduced from Adams and Dams, Ref. 4, with permission of Pergamon Press, © 1970).

many environmental examples, is good for illustration purposes. This nuclide decays by emitting a beta-particle with maximum-energy 2.8 MeV, followed by a 1.7789-MeV gamma ray. Sometimes the beta particle emits bremsstrahlung photons as it decelerates; sometimes the gamma produces e^+/e^- pairs, either in the source, in the detector, or elsewhere.

The peak at the highest energy is due to those events in which the full energy of the 1.7789-MeV gamma is converted within the detector to ionization (by the photoelectric effect or by multiple interactions). The broad continuum below the peak contains those events in which an incident photon scatters by Compton effect from an electron in the detector and then escapes from the detector. The signal is representative of the energy imparted to the electron. A head-on Compton collision yields the highest energy electrons, detected at the "Compton edge", clearly seen in Fig. 4 near channel 1600.

Other features of the spectrum are peaks labelled "S 1778.9", and "D 1778.9", due to single-escape and double-escape, respectively, of 511-keV gammas after the annihilation of a positron following e^+/e^- pair production in the detector (see Fig. 3).

The "511" peak represents total absorption of 511-keV gammas, presumably from annihilation of positrons mostly within the ^{28}Al source itself. Another feature is the prominent low-energy continuum labelled "Bremsstrahlung 2.8 MeV", from the beta particles decelerating mostly in the source.

The ratio of the area under the photoelectric peak to that under the Compton spectrum is primarily determined by physical parameters (the photon energy, the geometrical size and shape of the detector, and the Z of the material). As this ratio is fundamental to the ultimate system sensitivity, it is naturally desirable that it be maximized.

The detector size and shape are important parameters in the overall efficiency: clearly some photons will not interact in a small detector, so large detectors are definitely advantageous if only from considerations of volume or solid-angle. There is more to the size question than mere first-interaction volume, however; for larger detectors, there is an increasingly larger probability that Compton-scattered photons from a primary interaction will interact again in the crystal. Such multiple interactions effectively enhance the full-energy-peak-to-Compton

Fig. 5. Peak-to-total ratios for two crystal sizes and the peak-to-total ratio that would be found if no secondary absorption occurs. (Reproduced from Neiler and Bell, Ref. 9, with permission of North Holland Publishing Co., © 1965).

ratio. Figure 5 (from Ref. 9) shows this effect for two sizes of sodium-iodide crystals.

b. Energy Resolution

The second key parameter is energy resolution, since the width of a peak determines its ability to stand out above a continuum background and its ability to be distinguished from other nearby peaks. Resolution is usually stated in terms of the full width of a peak at half maximum (FWHM) and sometimes also by full width at one-tenth maximum (FWTM), this latter as an indicator of the extent of tails in the peak distribution.

Because energy resolution is so extremely important, much of the discussion on the individual measurement techniques is devoted to it. Here, only a few general comments will be made.

Resolution by itself is not a definitive parameter by which to compare performance of various systems. For example, resolution enters directly into the ability of a system to detect a

small peak sitting on top of a large background continuum. Yet in many systems improving the resolution is not as important as suppressing continuum backgrounds. To combine these effects, various figures-of-merit have been proposed. One such is the "full-energy-peak-to-Compton-edge ratio" defined as the ratio of full-energy-peak height to the height of the Compton edge, as measured in the final system energy distribution. This ratio is, in large part, inversely proportional to the peak energy resolution, but also depends upon many factors such as the overall detection efficiency and the ability of the system to suppress Compton events.

Such suppression is usually accomplished by surrounding the detector with an "anti-coincidence shield" to detect events in which, after a photon Compton scatters in the detector, the secondary lower-energy Compton photon subsequently escapes from the detector. If such Compton events are vetoed electronically, using such a shield, the full-energy-peak-to-Compton ratio can be enhanced by as much as two orders of magnitude. These veto systems are discussed in more detail below.

c. Background

The third important parameter is the degree to which counts not arising from the radionuclides of interest constitute "noise" or background. There are several types of background: (a) cosmic radiation; (b) contaminating radioactivity in the counting area, the shielding, or the detector equipment itself; and (c) electronically-induced background. Much effort has been expended devising techniques to minimize the radioactivity backgrounds by shielding and by choice of materials. This is particularly important when great sensitivity is desired.

In the instrumentation discussions below, we point out the performance of the various systems with respect to efficiency, resolution, and background.

4. Detection and Measurement Techniques

There are two types of gamma-ray spectrometers in common use at present; one employs the scintillation counter,

the other a semiconductor detector. However, in special cases other types of detectors are useful for spectrometry. The areas of applicability along with the basic properties of each detector type will be discussed below. Areas where new developments show promise will also be indicated. We shall not discuss spectrometer types which find only occasional use, such as pair spectrometers and diffraction spectrometers, nor shall we discuss liquid and plastic scintillators. We will begin our discussion with the gas proportional counter.

a. Gas Proportional Counters

The use of gas proportional counters for gamma and x-ray spectrometry dates from about the same period (late 1940's) as the development of scintillation spectrometry. Indeed, for low-energy photons this was the most widely used technique until the development of semiconductor detectors. Today, gas proportional counters are used for low-level gamma spectrometry only when special problems arise.

The passage of photons through the gas produces pairs of electrons and positive ions. About 25 to 30 eV is required to make one such pair, which is about 10 times greater than the 2.9 eV required for an electron-hole pair in solid germanium, but a factor of about 10 smaller than required to produce one photoelectron from a NaI(Tl) scintillator-phototube instrument. Hence, in the region where statistical fluctuations dominate the energy resolution, the resolution of the gas counter is intermediate between that of the Ge and NaI(Tl) systems discussed later.

The total primary charge liberated by a low-energy photon is very small, so that the signal-to-noise ratio obtainable with a practical amplifier is poor. Gas counters operated in the ionization-chamber mode have therefore seldom been used for gamma ray spectrometry. It is usually necessary to operate the gas counters in the proportional mode, in which the primary signal is amplified by successive ionizing collisions of the electrons on the way to the collector anode. Such amplification can raise the output pulse size above the noise level of present amplifiers. Most commonly, a cylindrical geometry is used, with the positive anode wire on the axis of a cylindrical gas container.

Because of limitations on size and pressure, gas counters do not have the stopping power of denser detectors (solids, liquids) for high-energy gammas. Yet at sufficiently low energies, they are in many ways superior to the scintillation detectors: high energy resolution is obtained, and lower energies can be detected and measured reliably. Another advantage is the inherent energy discrimination against high-energy radiation. Proportional counters are most important for photon energies below 2 keV; they have been used down to 250 eV.

There are a number of factors which must be considered in the design and operation of a gas proportional counter as a photon spectrometer. The choice of window material is important: it must be very thin and of low atomic number (Z) to minimize interactions in the window; and it should be chosen to avoid fluorescence (excitation of secondary x radiation by the primary incident photons). For these reasons beryllium, carbon, or a plastic is often chosen.

The gas fill is usually high-purity xenon, krypton, or argon: the choice is based upon the energy region to be measured, and primarily takes into account the gas absorption characteristics as a function of energy. Figure 6 (from Ref. 10), for example, shows the absorption of photons for each of these three gases in a typical commercial counter. The x-ray 'escape' peak from excitation in the gas may also be important: the K x rays for the three gases occur at 29.5 keV (xenon), 12.5 keV (krypton), and 3.0 keV (argon). A quench gas, typically a few percent CO₂, is used to prevent continuous discharge, to improve pulse characteristics, and to lengthen counter lifetime.

The choice of wall material is also important: interference from fluorescence in the walls should be avoided, and this often dictates using a low-Z liner such as aluminum if a higher-Z material such as stainless steel is used for the main wall structure.

The performance of the best gas proportional counters at low energies is very good. For example, resolutions in the range of 15% (FWHM) can be routinely achieved for the 5.9-keV line from iron-55 in the better counters, and sometimes even finer resolution is obtained. For example, resolution values of 11.6% and 12.2% for 5.9 keV have

Fig. 6. Gas absorption (%) at one atmosphere pressure in Reuter-Stokes RSG-61, a typical commercial gas proportional counter of 2" diameter, 4.7" sensitive length. The effect of window absorption is not included. Gases are (a) 97% xenon/3% CO₂; (b) 97% krypton/3% CO₂; (c) 97% argon/3% CO₂. (Reproduced with permission of Reuter-Stokes Electronic Components, Inc., Ref. 10).

been obtained experimentally (Ref. 11) for proportional counters with gas fillings Ne + 0.5% Ar and Ar + 0.5% C₂H₂. To achieve the best resolution, a very stable high-voltage supply must be used. The resolution obtainable at various energies with instruments of this type is shown in Fig. 7 (Ref. 4).

Among the advantages of gas counters are that a large source area can be utilized, up to about 100 cm² for some specially designed counters; backgrounds can be very low, with an especially important low sensitivity to high-energy (≥ 100 keV) photons. The instruments are easy to operate and can be made rugged; a variety of configura-

Fig. 7. Resolution of x-ray proportional counter as a function of energy, from measurements of several investigators. (Reproduced from Adams and Dams, Ref. 4, with permission of Pergamon Press, © 1970).

tions are commercially available, with choices of side or end windows and various gases and pressures; and the costs are usually low. It might be noted also that often the same proportional counter can be used for obtaining spectra of beta-particle energies.

Two reviews of the use of gas proportional counters for photon spectrometry are those of Crouthamel (Ref. 4) and Curran and Wilson (Ref. 12).

b. Scintillation Counters:
NaI and CsI

The scintillation counter consists of a clear phosphor scintillator, a photomultiplier tube, and electronic circuitry. An incident photon produces ionization that excites the phosphor and causes emission of light. This light can eject photoelectrons from the photomultiplier cathode, which (after secondary-emission amplification by the cascaded dynodes) appears as a current pulse at the anode. The electronics analyzes and displays or records the results. Since the amount of light in the flash is proportional to the energy deposited in the phosphor as ionization, the size of the output pulse is a measure of that energy. If the electronics includes a multi-channel pulse-height analyzer, it will record the spectrum of the ionization energies.

(1) Sodium-Iodide Crystal Spectrometers. Scintillation counters employing thallium-activated sodium iodide, commonly denoted NaI(Tl), have been a mainstay of gamma spectrometry since the development of NaI(Tl) as a detector, by Hofstadter in 1948 (Ref. 13). A vast literature exists on this subject, and reviewing it is beyond the scope of this work. Instead, this discussion shall concentrate on outlining the basic properties, uses, sensitivities, and limitations of sodium iodide in gamma spectrometry. For more detailed discussions, the reader is referred to the proceedings of the Scintillation Counter Symposia (Ref. 14) as well as several reviews (Refs. 9, 15-18).

Crystalline NaI is quite dense (3.67 g/cm^3), and has a high effective Z because of the iodine ($Z = 53$). The thallium activator, an impurity of about 0.1 to 0.3% introduced into the crystal, converts some of the absorbed

ionization energy into visible light. The decay time of the light (time for light intensity from an instantaneous event to fall to $1/e$ of its peak value) is about 250 nsec. The pulse size is proportional to the ionization deposited. Although the light pulse from NaI(Tl) lasts several hundred nanoseconds, the time resolution achievable is much shorter; using electronic analysis methods such as constant fraction discrimination or time-of-pulse-peak determination by differentiation and zero-crossing circuits, time resolution as low as a fraction of a nsec can be achieved.

Physically, it is desirable for every photon to interact by photoelectric effect, imparting all of its energy except the electron binding energy to the ejected electron. The binding energy subsequently goes into either characteristic x rays or Auger electrons as the electron vacancy is filled. In NaI(Tl), the dominant photoprocesses occur in the iodine innermost shell and the photoelectrons are nearly always accompanied by 28-keV K_{α} x rays; when this radiation escapes from the crystal, two peaks result: a full energy peak and lower energy escape peak. This is most important for low-energy incident photons, both because they interact near the crystal surface, and because the loss of 28 keV is a relatively large effect for low energies. The angle of incidence also makes a substantial difference. This is shown in Fig. 8 (from Ref. 4), in which the ratio of the iodine-escape peak to the full-energy peak is shown as a function of energy. This figure demonstrates that this effect can be very important at the lower energies.

There are a number of parameters to be considered in the choice of a NaI(Tl) system. The most important are probably the crystal size and geometry, and the type of electronic instrumentation. The importance of crystal size in enhancing the efficiency for detecting multiple-interaction events has already been discussed above. The geometrical shape of the crystal is also important; right cylindrical crystals are most commonly used, since both optical coupling to the phototube and edge-effect losses are minimized in cylindrical geometries. Another common geometrical arrangement is the well-type, in which the sample is placed in a well cavity in the center of a NaI(Tl) crystal. This provides increased solid-angle, but at a sacrifice

Fig. 8. Ratio of escape peak to photopeak in "infinitely thick" sodium iodide crystal. Crystal subtends conical shell of radiation. (Reproduced from Adams and Dams, Ref. 4, with permission of Pergamon Press, © 1970).

in effective detector volume for the same overall size.

The energy resolution of a NaI(Tl) system is primarily dominated by statistical uncertainties: in a typical NaI(Tl) crystal about 200 to 300 eV of ionization energy must be deposited on the average for every photoelectron created at the photocathode of the multiplier tube. Since the light-emission, the photoelectron process, and the tube multiplication are all subject to statistical fluctuations, there is an inevitable 'line-width' for monoenergetic energy deposition with the crystal. This line-width, which is a function of energy, is broader for scintillation detectors than for any of the other commonly used detectors. Indeed, it is the principal drawback to the use of crystals such as NaI(Tl).

If the statistical fluctuations were due only to energy deposition, the variation would be such that the aver-

age number of electrons N in the ultimate electronic pulse would be proportional to the energy deposition E , and the fractional line width $\Delta E/E$ would be proportional to $\sqrt{N}/N = 1/\sqrt{N}$. This is nearly the correct model at low energies, but breaks down above about 200 keV, because of the added contributions from Compton scattering and edge effects. This is shown in Fig. 9 (Ref. 4), where for a 2.5" (diameter) x 2" (high) cylindrical crystal $(\Delta E/E)^2$ is seen to be nearly linearly proportional to $1/E$.

The choice of electronic instrumentation is crucial to overall system performance. The quantum efficiency of the multiplier tube's photocathode is one key parameter in the system resolution, along with the uniformity of efficiency over the photocathode area. Low tube noise and good multiplication

Fig. 9. Calculated values of resolution for a 2.5 in. dia. x 2 in. high sodium iodide-thallium activated crystal. $(\Delta E/E)^2$ is plotted against reciprocal of energy $(1/E)$. Curve (a) includes only intrinsic broadening and photo-multiplier statistics. Curve (b) includes also the contribution from the variance in photon transfer. (Reproduced from Adams and Dams, Ref. 4, with permission of Pergamon Press, © 1970).

statistics are other parameters which must be considered, especially for detecting the lowest-energy photons. Sometimes two photomultipliers are used in coincidence, in which case the contribution of direct tube noise to the background coincidence counting rate is almost entirely eliminated. Much has been written about the detailed considerations (optical, mechanical, and electronic) involved in phototube choice and use. One reference for the interested reader is the book by Crouthamel, Adams, and Dams (Ref. 4).

The ultimate performance of any NaI(Tl) system is governed by all of the parameters touched upon above as they affect both the total efficiency and the photoelectric-to-Compton ratio; time resolution; energy resolution; crystal size and shape; and the optical collection/phototube parameters.

(2) Cesium-Iodide Crystal Spectrometers. While sodium-iodide is by far the most widely used inorganic scintillator, thallium-activated cesium iodide, CsI(Tl), also has useful applications. CsI(Tl) differs from NaI(Tl) in three fundamental ways: (i) it produces only 40 to 50% as much light per eV of deposited energy; (ii) its mean decay time for gammas is about 1600 nsec compared to about 250 nsec for NaI(Tl); and (iii) cesium has a much higher Z(55) than sodium (11), so that the photoelectric interaction is somewhat enhanced.

CsI(Tl) has not achieved widespread use for routine gamma spectrometry. This is probably because the cost of CsI(Tl) is much greater than that of NaI(Tl), yet its overall resolution and other performance characteristics are somewhat similar.

Perhaps the use of CsI(Tl) most pertinent here is the phosphor-sandwich or "phoswich". This arrangement, shown schematically in Fig. 10 (Ref. 15), places a NaI(Tl) crystal in front of a CsI(Tl) crystal, which in turn is in front of the photomultiplier tube. The entire system is optically coupled together. Low-energy gamma-rays stopping within the NaI(Tl) crystal appear as a fast pulse, while higher-energy rays that penetrate to the CsI(Tl) yield a phototube pulse containing both short and long components, as shown in the figure. This configuration allows the low-energy photons, which have very short penetration, to be separated electronically by pulse-shape analysis.

Fig. 10. The "Phoswich". (a) Fast-slow phosphor sandwich; (b) Dual pulse from particle traversing XX in (a). t_{of} and t_{os} are the $1/e$ times for the fast and slow components of the resulting pulse. (Reproduced from Birks, Ref. 15, with permission of Pergamon Press, © 1964).

c. Semiconductor Detectors

The importance of semiconductor detectors in gamma spectrometry cannot be overemphasized. Since the use of these devices became widespread in the mid-1960's, many important fields of research and measurement have been revolutionized. This is mainly because of the superb energy resolution achievable with these detectors, but also because of their stability and reliability. It is now possible to make routine measurements of many radionuclides at low concentrations which are beyond the capabilities of other systems.

The basic useful property of this material is that electron-hole pairs can be created by the ionization process. A potential applied to the detector's electrodes causes the charge carriers to move toward the electrodes. For our purposes the crucial properties are two: First, the average ionization energy ϵ required to produce a hole-electron pair must be small, to minimize the fractional fluctuations from statistical uncertainties. For germanium $\epsilon \approx 2.9$ eV, while for silicon $\epsilon \approx 3.7$ eV. Second, the mobility must be sufficiently high that essentially all of the carriers are collected at the electrodes. Finally, the carrier lifetime must be great enough so that recombination is unimportant. Under this condition, after suitable amplification, an electronic pulse may be produced that is proportional to the energy deposited in the active area of the detector.

We shall not discuss in detail the theory behind the choice of materials and the operating parameters for these detectors. Several excellent discussions exist (Refs. 4,5,19-25). Suffice it to say here that only two materials presently approach the requirements for a high-resolution detector of reasonable size; the most desirable materials are germanium or silicon of ultra-high purity.

Semiconductor radiation detectors consist essentially of a piece of solid material whose nature and size is matched to the absorption characteristics of the radiation to be measured. Producing an electrical signal that is a direct measure of the energy of the absorbed photon, may not be easy to achieve, and much of Ref. 25 is concerned with appropriate detector characteristics, some of which have already been mentioned above.

If the electrical contacts on the material permit a free exchange of holes and electrons between the semiconductors and the electrodes, large currents will flow through the detector when voltage is applied across it. To overcome this, "blocking" contacts must be used - that is, the positive contact must not inject holes, nor the negative contact inject electrons into the detector material. This can be achieved either by careful choice and application of the metals used for the contacts, or by forming heavily doped n and p regions by high temperature diffusion or by ion implantation into the detector surfaces. The n^+ surface is used as the positive detector electrode, and the p^+ surface as the negative electrode. (The n^+ and p^+ symbols refer to heavily doped n and p material.) The resulting junction is then operated in the reversed biased mode, driving free holes in the p region and free electrons in the n region away from the junction. The depletion layer that is left is the actual detector. In this depletion layer, the supply of free carriers available to move in the applied field direction is small, and the bias causes only a very low leakage current. However, electron-hole pairs produced by the passage of radiation are effectively collected because of the applied electric field.

The requirement of low leakage current historically prompted effort to compensate for impurities that produce unwanted free carriers. In recent years, it has often been possible to avoid such compensation by the use of

recently available ultra-high purity semiconductor material. However, the usual method for compensating is drifting of n-type lithium; This has been well discussed elsewhere (Refs. 4,5, 20). Large volume germanium detectors are now available, with $\geq 100 \text{ cm}^3$ crystals in common use.

(1) Germanium. Because photoelectric photon conversion (with its Z^5 dependence) is a key detection mechanism, germanium ($Z = 32$) is favored over silicon ($Z = 14$) by a factor of about 40. At energies above about 30 keV, this is very important. The actual peak results from both the Compton and photoelectric effects. Most germanium detectors in use today are lithium drifted, usually designated Ge(Li), but intrinsic germanium detectors, which do not employ such drifting, are increasingly coming into use.

The average number N of electron-hole pairs created by deposition of energy E is E/ϵ , and the standard deviation σ_T due to the statistical fluctuations can be expressed as $\sigma_T = \sqrt{N} = \sqrt{E/\epsilon}$. Experimentally the fluctuations turn out to be much smaller. If the experimentally determined standard deviation is denoted by σ_{exp} , then the ratio $\sigma_{\text{exp}}/\sigma_T$ serves to define the "Fano factor" F :

$$F = (\sigma_{\text{exp}}/\sigma_T)^2 \\ \sigma_{\text{exp}} = \sqrt{(E/\epsilon)F}.$$

The full width at half maximum (FWHM) is 2.355 times the standard deviation, as is exactly true for a Gaussian distribution: $\text{FWHM} = 2.355 \sigma_{\text{exp}}$.

The discrepancy between σ_{exp} and $\sigma_T (= \sqrt{N})$ can be explained as follows: not all of the energy deposited in the semiconductor goes into creating electron-hole pairs; a fraction goes into exciting other modes such as lattice vibrations. If all of the energy loss were to go into creating electron-hole pairs, then N would equal E/ϵ , there would be no statistical fluctuations in the number N of pairs formed, F would equal zero, and there would be no contribution to σ_{exp} from this source. On the other hand, if only a small fraction of the energy loss were to go into creating pairs, the fluctuations in this small fraction would be truly random and F would equal one. In semiconductors, the true situation lies between these two extremes. Experimental values of F differ widely, depending probably upon

the techniques used in crystal fabrication. Fano factors as low as $F = 0.08$ have been measured (Ref. 26) but values in the region of about 0.12 are typical of present detectors. The intrinsic resolution of such detectors can be read off of Fig. 11.

The electronic instrumentation used in conjunction with these detectors is extremely important (Ref. 28). Developments in this area were recently reviewed by Goulding and Landis (Ref. 29), and further progress continues to produce improved techniques. Early in their development, semiconductor detectors suffered from degraded resolution because of electronic amplifier inadequacies. Today, the best available electronic equipment contributes little to the overall system resolution, which (at least at higher energies) is dominated mainly by the statistical fluctuations discussed above.

The primary aim of the electronic instrumentation is to provide a final spectrum as free as possible of both non-linear distortion and noise. The first component in this system is the preamplifier. The need in semiconductor applications for extremely low-noise preamplifiers has stimulated intensive theoretical and experimental research, and during the past few years it has become possible to approach quite closely the fundamental noise limits of these devices. The aim has been that the contribution of preamp noise to the final line-width be significantly smaller than the detector's own line-width. A charge-sensitive (integrating) preamp is used almost universally, primarily because its gain is independent of the detector capacitance, and despite the fact that the noise performance may be slightly worse than that of a voltage amplifier. The preamp and the circuits that follow it

Fig. 11. Resolution width as a function of energy, parametric in Fano factor, for germanium detectors. The results of measurements at various energies are shown. (Reproduced from Mann, Bilger, and Sherman, Ref. 27, with permission of the Institute of Electrical and Electronic Engineers, © 1966).

normally convert the signal to a standard shaped pulse whose height is proportional to the charge at the input.

The time resolution achievable with germanium detectors is in the range of a few to about 10 nsec, depending upon the detector size, geometrical shape, and electronic parameters. This is not as fast as that achievable with NaI(Tl) crystals, but is fast enough to be very valuable in coincidence applications, to be discussed below. A good discussion of fast-timing considerations for both Ge and NaI(Tl) is the paper by Williams and Nutt (Ref. 30).

Several reviews discuss the design considerations which go into not only the preamplifier, but also the other key components (pulse-shaping networks, base-line restoration circuits, analog-to-digital converters, etc.). The reader is referred to the general reviews of Adams and Dams (Ref. 4) and Benoit (Ref. 31), in addition to the reviews mentioned above by Goulding and Landis.

A number of varied detector geometries have been developed for semiconductor gamma detectors. Planar or parallel plate p-i-n configurations were the earliest detector geometries to achieve wide use. These consist of an intrinsic (compensated) interior with a p-type substrate on one side and an n-type on the other surface through which the photons enter. Unfortunately, the practical limit of lithium drift (15-20mm) limits the total volume which can be compensated in this way. To overcome this problem, the coaxial p-i-n detector was developed. High-quality p-type germanium is used. Lithium drifting compensates the major fraction of the volume, and the central core remains p-type; the lithium-diffused surface layer is n-type. Figure 12 schematically shows two kinds of coaxial detector, the true 'open-ended' coax and the 'closed-end' coax. A major advantage of the coaxial geometry is that large volumes can be compensated. Furthermore, low leakage currents result because the exposed junction area at the end is constant even for very long cylinders and resulting large sensitive volumes. The primary disadvantage is that for very long cylinders there is larger capacitance.

The larger volumes obtainable with coaxial geometries are a clear advantage at high energies ($\geq$ a few hundred keV), where gamma ray penetration beneath the thin surface layer is sub-

Fig. 12. True coaxial and wrap-around coaxial geometries shown schematically. (Reproduced with permission of EG&G ORTEC, Ref. 32).

stantial and the entire volume is useful for detection. At low energies (≤ 100 keV), the exposed detector area is a key parameter. The choice of geometry thus depends on the energy range which is being measured. An example of the trade-off is illustrated in Fig. 13 (from Ref. 33). At low energies, the

Fig. 13. Ratio of counting efficiency as a function of energy for a 17-cm³ planar and a 20-cm³ coaxial germanium detector. At high energies the volume governs the efficiency, while at low energies the efficiency is governed by the area. (Reproduced from Armantrout, Bradley and Phelps, Ref. 33, with permission of the Institute of Electrical and Electronic Engineers, © 1972).

ratio is governed by the areas, and at high energies by the volumes. A closed-end coaxial configuration made with n-type germanium using an ion-implanted outer contact with about 0.3 micron thickness exhibits enhanced response to low energy photons. Conventional closed-end coaxial detectors made from p-type germanium have entrance window thicknesses ranging from 300 to 1000 microns.

For some measurement requirements the closed-end-coaxial geometry can have serious disadvantages compared to the open-ended configuration, since it puts a dead region right at the "heart" of the measurement, near the source. The dead region can absorb many of the low-energy electrons generated near the "Compton edge", preventing them from escape and subsequent detection in a NaI(Tl) anticoincidence detector.

Cooper has expressed a key geometrical consideration (Ref. 34):

"In selecting detectors for low-level counting applications, care should be taken to maximize the ratio of the detector's cross-sectional area facing the source to its volume. Since the natural background is generally isotropic, its detection efficiency depends mainly on the volume while the detection efficiency for source gamma rays depends strongly on the detector cross section. Thus, although long detectors may have greater high-energy efficiencies than short detectors, their natural background interferences will be higher at all energies."

For many applications the short fat coaxial geometry (40-50mm diameter, 25-30mm length) is best, since it has fairly low capacitance (10-15 pf) but reasonably high detection efficiency. Indeed, there are applications in which a detector either too short or too long would be equally undesirable.

Germanium semiconductor detectors must be operated at or near to liquid-nitrogen temperatures (77°K). This must be done since the Ge band-gap is not large enough to allow room temperature operation. The low temperature operation reduces the leakage current from thermal excitation of holes and electrons. In addition, Ge(Li) detectors must also be cooled to prevent movement of Li, which is very mobile at room temperatures. Cooling is usually accomplished by a regular re-filling of

a dewar with liquid nitrogen. For a concise description of these germanium gamma ray detectors, the reader is referred to Ref. 35.

Because even the largest germanium detector crystals are of limited size, they cannot achieve the performance of NaI(Tl) crystals in terms of overall detection efficiency. This can be seen in Fig. 14 (Ref. 7), which compares several Ge(Li) detectors with a medium-sized (3" x 3") NaI(Tl) detector. Detectors are now produced commercially with a peak efficiency at 1.33 MeV which is 30% that of a 3" x 3" NaI detector (Ref. 36). Also germanium detectors cannot as easily accommodate large samples, or achieve the multiple interaction performance of NaI(Tl) which can be seen in Fig. 5. Figure 15 (Ref. 37) illustrates the full-energy-peak-to-total enhancement from use of Ge detectors. To overcome some of these drawbacks, hybrid systems have been developed. For example, NaI(Tl)

Fig. 14. Experimental full energy peak efficiency for detectors of different volumes. Curves 1 and 2 refer to planar detectors, while curves 3, 4, 5, and 6 refer to coaxial detectors. Curve 7 is for a 3" x 3" NaI(Tl) detector. Distance of source from detector face is indicated. (Reproduced from Capellani and Restelli, Ref. 7, with permission of J. Wiley and Sons, © 1968).

Fig. 15. ^{60}Co spectrum showing resolutions and peak-to-Compton ratios for a Ge(Li) coaxial detector and a NaI(Tl) detector. (Reproduced with permission of EG&G ORTEC, Ref. 37).

can be used to surround a Ge inner detector; NaI(Tl) detects in time coincidence many of the Compton-scattered photons escaping from the Ge crystal, enabling their rejection (to help low energy photopeaks to stand out above a Compton continuum) or occasionally their inclusion (to improve the system sensitivity). These hybrid systems will be discussed further below.

(2) Silicon. Silicon is not widely used for gamma-spectrometry except at low energies. The reason that Ge is preferred over Si in the 0.1 to 1.0 MeV region is germanium's much greater photopeak-to-Compton ratio. This was shown above in Fig. 1. However, this figure also showed that at energies below about 30-50 keV, the advantage of Ge becomes less important, and in fact at the lowest energies Si is now most widely used. Many of the considerations set forth above are generic to semiconductor detectors, applying to silicon as well as germanium.

One important advantage of Si is that for some applications it can be operated at room temperatures. This eliminates the need for any windows, and makes spectrometry of very low energy photons (a few keV) much simpler. For the best resolution, however, low temperatures are still necessary.

As has been pointed out above, a disadvantage is the lower density (2.33 g/cm^3) of silicon compared to german-

ium (5.33 g/cm^3). Another is that its intrinsic resolution is not quite as good, although when operated with only a small amount of cooling (near 0°C) the differences are slight. Figure 16 (Ref. 38) shows a typical very-low-energy spectrum taken with a Si(Li) detector, measuring the radioactivity on an air filter. Note that the 5.9-keV photon from ^{55}Fe is resolved quite easily. Today, commercially available Si(Li) detectors offer resolutions (FWHM) for ^{55}Fe in the vicinity of 160 eV.

Silicon detectors of small size are often used when it is necessary to count low-energy photons in the presence of large backgrounds of high-energy photons, since the detection efficiency for the higher energies will be quite small.

d. Comparison: NaI(Tl) vs. Ge

NaI(Tl) crystal spectrometry was the most important gamma measurement technique from the time of its discovery in 1948 until Ge systems began (in the 1960s) to play an increasingly larger role in many important types of measurements. Hence, it is appropriate to highlight the relative advantages and disadvantages of these two techniques. Such a comparison is particularly important because the systems under discussion usually

Fig. 16. Si(Li) photon spectrum of the airborne radionuclides collected on an air filter at Richland, Washington. (Reproduced from Cooper, Wogman, Palmer, and Perkins, Ref. 38, with permission of Pergamon Press, © 1968).

comprise the largest single initial expense in equipping a laboratory for environmental radionuclide measurements.

The basic trade-off is between the high efficiency of NaI(Tl) and the high resolution of Ge. The superb semiconductor resolution is obviously crucial in some applications, such as in resolution of closely-spaced lines. This can be seen, for example, in Fig. 17 (Ref. 38), in which the NaI(Tl) system is totally incapable of resolving a number of closely-spaced lines which appear clearly separated in the Ge(Li) measured spectrum.

The contrasting parameter is overall efficiency, determined both by intrinsic efficiency and by detector size. Even medium-sized NaI(Tl) crystals have an efficiency advantage over the largest Ge(Li) detectors. This was seen above in Fig. 14 in which the overall efficiency of a 3" x 3" NaI(Tl) crystal was compared to that of several Ge(Li) detectors. Even the 54-cm³ Ge(Li) detector has overall efficiency less by a factor of 3 or 4 than the 3" x 3" NaI(Tl) crystal.

Another key consideration is the amount of effort required to obtain measured activity values from the raw energy spectrum. In this regard, it should be observed that for complex spectra NaI(Tl) systems almost certainly require substantial computer processing, while Ge systems often can yield important information with only moderate data analysis. The data-analysis part of gamma spectrometry is discussed separately below.

While these basic facts about resolution, efficiency, and data handling are paramount, a number of variations have been developed to optimize the usefulness of the detectors in particular applications. We shall discuss some of these here.

e. Special Detector Configurations

The most common complement to an ordinary system is availability of a low-background counting room. Two important sources of background can be

Fig. 17. A comparison of the gamma-ray spectra of the radionuclides present one year after collection on an air filter at Richland, Washington as measured with a 5-in. dia. by 3-in. thick NaI(Tl) scintillator and a 20 cc Ge(Li) diode. (Reproduced from Cooper, Wogman, Palmer, and Perkins, Ref. 38, with permission of Pergamon Press, © 1968).

minimized: cosmic radiation and natural terrestrial backgrounds. To minimize these, completely enclosed lead "caves" are used, and in some cases counting rooms have been built below ground. The gains can be substantial: background reductions by factors up to a few hundred are common. For example, studies of backgrounds 110 meters beneath Grand Coulee Dam (Ref. 39) are shown in Fig. 18. Use of special low-activity lead also offers advantages particularly at the lower energies. Studies of natural backgrounds in cement (Ref. 40) and concrete (Ref. 41) have led to the use of low-background materials in counting enclosures (Ref. 42). Background reductions by additional factors of 2 to 5 over backgrounds in normal concrete rooms can be

gained by use of such materials, even if lead shielding is already in use around the detector system.

Another common improvement which can be added to a normal detector system is the use of anticoincidence counters. In the simplest arrangement, thin plastic scintillation counters can be used to veto (cancel electronically) pulses from high-energy cosmic-ray charged particles traversing the detector system. A more costly but much more effective arrangement is surrounding the detector with larger volumes of anticoincidence scintillators. In addition to eliminating cosmic ray events, these detect and veto events in which some of the energy escapes after a primary photon interaction in the inner detector. The large volumes are needed to achieve high detection efficiency. The suppression of the Compton continuum can be particularly effective, along with the detection of escaping 511-keV photons from positron annihilation after pair-production.

In one common arrangement, large NaI(Tl) detectors have been surrounded by additional NaI(Tl) scintillators: for example, Fig. 19 (Ref. 43) shows the effect of such a system on suppressing the Compton continuum below the 662-keV gamma peak from ^{137}Cs . The suppression of the cosmic-ray muon background by 3 to 4 orders of magnitude is shown in Fig. 20 (Ref. 43). It should be emphasized that when anticoincidence methods are used, the addition of lead shielding provides little extra benefit. This can be seen, for example, in Fig. 21 (Ref. 39).

Surrounding a Ge detector with anticoincidence counters provides even more impressive Compton suppression, especially because the small size of even the largest Ge detectors results in very large Compton backgrounds. For example, Fig. 22 (Ref. 44) shows that a reduction of nearly an order of magnitude can be achieved by surrounding a 20-cm³ Ge detector with plastic phosphor of 26" diameter and 24" thickness. The peak-to-Compton height ratio is 2455:1, as indicated. The detection sensitivity of this system for various conditions is shown in Fig. 23 (Ref. 44). Figure 24 (Ref. 44) shows a particular example in which the Compton continuum below the two ^{60}Co photopeaks normally would completely obscure the lower-energy photopeaks from several other gamma emitters. The anticoincidence system, while reducing the sensitivity for the ^{60}Co lines by a fac-

Fig. 18. Effects of massive shielding on backgrounds of simple NaI(Tl) detector system. Comparison is between Richland, Washington and various configurations at top of and inside Grand Coulee Dam, Washington. (Reproduced from Kaye, Brauer, Connally, and Pieck with permission of Nuclear Instruments and Methods, © 1972).

Fig. 19. Comparison of the ^{137}Cs peak to Compton distribution for two sodium iodide detector systems; the bottom curve includes surrounding the larger NaI(Tl) detector with an anti-coincidence shield. The peak is at 662 keV. (Reproduced from Wogman, Perkins, and Kaye, Ref. 43, with permission of Nuclear Instruments and Methods, © 1969).

tor of about 6, reduces the ^{60}Co Compton edges by about a factor of 50. By storing the spectra both with and without anticoincidence, no primary information is lost.

The ability of systems to measure large samples is another aspect. Especially for the Ge systems, losses in efficiency with large samples can significantly offset other positive factors. As an example, Fig. 25 (Ref. 45) shows the efficiency of a 100-cm³ Ge(Li) cylindrical detector for a point source and for a sample of 6" diameter and 1" thickness (sample volume of about 450 cm³). With the larger sample, the efficiency is lower by a factor of 5 to 10.

Fig. 20. Comparison of the muon background of an all-sodium-iodide detector system to various mass and anticoincidence shielding conditions. The 110-meter overburden is beneath Grand Coulee Dam. (Reproduced from Wogman, Perkins, and Kaye, Ref. 43, with permission of Nuclear Instruments and Methods, © 1969).

Recently the Marinelli beaker has been introduced to give better geometry for a large sample. Since the Marinelli beaker completely covers an area all but under the detector (see Fig. 26), greater sensitivity results. Environmental samples usually involve large volume sources so it is important to determine the detectable activity in terms of gammas/min/ml of the sample material, as given in Table 1 (from Ref. 46). Clearly the Marinelli geometry in this respect gives the higher sample sensitivity.

Well-type NaI(Tl) systems are now in common use. Insertion of a sample within an enclosed well increases the solid angle to nearly a complete sphere. However, for the same size total crystal some losses in efficiency can occur before passing out of the sensitive volume of the detector. Germanium well detectors have also been built and used to advantage (Ref. 7). With the availability of high purity germanium, better well-type detectors

Fig. 21. Background measurements at Richland, Washington, with an anticoincidence-shielded sodium iodide detector, showing effect of anti-coincidence and of 10.2 cm of additional lead. (Reproduced from Kaye, Brauer, Connally, and Piech, Ref. 39, with permission of Nuclear Instruments and Methods, © 1972).

Fig. 22. Gamma-ray spectra of ^{137}Cs recorded in the normal and anti-coincidence modes, with a germanium detector. Full-width-half-maximum is noted as 2.38 keV, and peak-to-Compton height ratio as 245. (Reproduced from Cooper, Rancitelli, and Perkins, Ref. 44, with permission of Journal of Radioanalytical Chemistry, © 1970).

Fig. 23. Gamma-ray detection sensitivity for a normal germanium diode (1000-minute counting time) compared to sensitivities of anti-coincidence shielded diode for 1000, 4000, and 10,000 minute counting times. (Reproduced from Cooper, Rancitelli, and Perkins, Ref. 44, with permission of Journal of Radioanalytical Chemistry, © 1970).

can be made than is possible with Ge(Li) (Ref. 48). In an evaluation of well-type Ge(Li) detectors for low level radiochemical analyses (Ref. 49), the well detectors were found to have better detection efficiencies and as low background as either large coaxial detectors or opposed coincident detector systems.

Coincidence counting is another variation on gamma spectrometry, using unique signatures to decrease backgrounds and identify certain radionuclides in the presence of others. The most common method is to use two NaI(Tl) crystals on opposite sides of a sample. Dual-Ge systems and hybrid Ge-NaI(Tl) coincidence arrangements have also found some application. Each has its advantages, the trade-offs being (as always) efficiency vs. resolution. This trade-off is especially critical for coincidence work because the overall efficiency is the product of the two individual detector efficiencies.

Fig. 24. Coincidence and anticoincidence γ -ray spectra of muscle tissue from a tuna collected in the region of the Bikini atoll. Data taken with a 3800-minute counting time using a Ge(Li) diode. (Reproduced from Cooper, Rancitelli, and Perkins, Ref. 44, with permission of the Journal of Radioanalytical Chemistry, © 1970).

Fig. 25. Counting efficiency of a large Ge(Li) detector for two source sizes. (Private communication from R. W. Perkins, Ref. 45).

Marinelli Beaker Placed upon Detector

Fig. 26. Marinelli sample-detector geometry used for measurement of weak activities. (Reproduced from Walford, Cooper, and Keyster, Ref. 46, with permission of the Institute of Electrical and Electronic Engineers, © 1976).

Table 1. Minimum Detectable Activity in $\gamma/\text{min}/\text{ml}$ of Sample Material for the Marinelli Beaker and for the Disk Sample (from Ref. 46)
One VIP¹⁰ Detector
(a 10% efficient Ge(Li) detector)

Energy (keV)	(For Disk Sample) $\gamma/\text{min}/\text{ml}$	(For Marinelli Beaker) $\gamma/\text{min}/\text{ml}$	Marinelli Vol. Reduced Evaporation to 5ml Disk ¹
185	0.3	8.9×10^{-3}	2.7×10^{-3}
609	0.64	1.7×10^{-2}	5.6×10^{-3}
1238	1.1	2.7×10^{-2}	9.6×10^{-3}
1764	1.2	3.1×10^{-2}	1.1×10^{-2}
2204	1.5	3.8×10^{-2}	1.36×10^{-2}

One WIN¹⁵ Detector
(a 15% efficient Ge(Li) detector)

Energy (keV)	(For Disk Sample) $\gamma/\text{min}/\text{ml}$	(For Marinelli Beaker) $\gamma/\text{min}/\text{ml}$	Marinelli Vol. Reduced by Evaporation to 5ml Disk ¹
185	0.3	7.2×10^{-3}	2.7×10^{-3}
609	0.6	1.4×10^{-2}	5.4×10^{-3}
1238	0.9	2.0×10^{-2}	7.6×10^{-3}
1764	0.8	1.9×10^{-2}	7.6×10^{-3}
2204	1.2	2.7×10^{-2}	1.1×10^{-2}

¹Expressed as the activity in $\gamma/\text{min}/\text{ml}$ of original sample volume.
Count time = 40,000 sec
Statistical precision = 0.2 (standard deviation of peak area divided by peak area)

The efficiencies of some dual-NaI(Tl) systems are shown in Fig. 27 (Ref. 43). The curves labelled "2E" denote the total efficiency for the paired system, counting single gammas. The "2E²" curves are efficiencies in coincidence mode. The suppression of background is shown in Fig. 28 (Ref. 50).

Another arrangement is to place a Ge detector inside a NaI(Tl) well. Here the NaI(Tl) can be used in both anticoincidence mode and coincidence mode. The effect of such an arrangement of a Ge spectrum from an air filter is shown in Fig. 29 (Ref. 51). A large number of radionuclides can be observed in the coincidence spectrum, in which a coincident NaI(Tl) pulse gates the Ge pulse height into a separate region of the pulse-height-analyzer memory.

Fig. 27. Total counting efficiencies and total coincidence counting efficiencies of three multidimensional gamma-ray detector systems. Curves labeled "2E" show summed efficiency of two identical detectors. Curves labelled "2E²" show coincidence efficiency. (Reproduced from Wogman, Perkins, and Kaye (Ref. 43) with permission of Nuclear Instruments and Methods, © 1969).

Other types of coincidence arrangements have also been developed. For example, beta-gamma coincidence counting can be used to obtain substantial improvements in sensitivity for certain radionuclides. Using the coincidence configuration shown in Fig. 30, Brauer, et al. (Ref. 52) report measurements of iodine-131 at detection limits in the range of 0.01 pCi, with 1000-minute counting times. A typical spectrum for a chemically-separated ¹³¹I sample is shown in Fig. 31 (Ref. 52); this spectrum was taken using a 0.75" x 0.020" plastic beta scintillator, inside a 5" x 5" NaI(Tl) well detector. The detection limit is a factor of 100 lower than that achievable with the 5" x 5" detector alone. Similar arrangements for gamma-x-ray coincidence counting

Fig. 28. Comparison of the background of a single 6" thick x 11" diameter crystal with that of the two crystal multidimensional detector system. The detector system is housed inside a 4" thick lead and 4" thick borated-paraffin shield. (Reproduced from Wogman, Robertson, and Perkins, Ref. 50, with permission of Nuclear Instruments and Methods, © 1967).

Fig. 29. Coincidence and anticoincidence gamma-ray spectra of the radionuclides collected on an air filter from 6×10^6 ft³ of air. Detector is Ge(Li) semiconductor, counting interval of 1000 minutes. (Reproduced from Cooper and Perkins, Ref. 51, with permission of Nuclear Instruments and Methods, © 1972).

Fig. 30. Low level β - γ detector, showing beta phosphor inside well-type NaI(Tl) system. (Reproduced from Brauer, Kaye, and Connally, Ref. 52, with permission of the American Chemical Society, © 1970).

Fig. 31. γ -Spectra of separated iodine sample taken in a 5-inch NaI(Tl) well system, with beta detection with a small plastic phosphor inside the well (see Fig. 30). Gamma and beta-gamma coincidence spectra shown separately. (Reproduced from Brauer, Kaye, and Connally, Ref. 52, with permission of the American Chemical Society, © 1970).

have also been reported by Brauer et al., using a gas-flow proportional counter for x-ray detection inside a NaI(Tl) well.

Efficiency determinations for the simpler systems are often performed by calibration measurements on a few isotopes with gammas of widely-spaced energies; interpolations are then performed for absolute analyses at intermediate energies. In using any of the anti-coincidence or coincidence systems, however, precise quantitative analysis requires that efficiency calibrations be performed for each isotope to be measured.

f. Other Detectors: Present and Future

Research in detector technology is now proceeding mainly in the following areas: first, to develop other semiconductor materials besides

germanium and silicon; second, to improve and enlarge ultra-pure germanium crystals; third, to improve the resolution of scintillation counter systems; fourth, to develop improved proportional-counter techniques; and fifth, to develop gas proportional scintillation counters. All of these efforts have a common goal: the availability at lower cost of significantly larger detector sizes, with resolution at least comparable to that obtainable with existing detectors. A secondary goal of the semiconductor research is to find materials that do not require storage at low temperatures (a drawback of Ge(Li)) and that ultimately might even be operable at room temperatures. We shall attempt only to present an overview of current progress.

• The search for new materials has been going on since the beginning of semiconductor technology. Among the materials now being actively investigated are cadmium telluride (CdTe), gallium arsenide (GaAs), and mercuric iodide (HgI₂). A recent conference on CdTe (Ref. 53) contains much important information: an article by Miller (Ref. 54) gives an overview for these three materials.

Additional information is to be found in the proceedings of the International Workshop on CdTe and HgI₂ held in Jerusalem, Israel (Ref. 55). HgI₂ has an ϵ value (hole-electron pair formation energy) of 4.2 eV. Both CdTe and GaAs have ϵ values of about 4.3 to 4.6 eV/pair. For use at x-ray energies, these high-Z semiconductors offer efficient photon detectors at or near room temperature with energy resolution significantly better than NaI(Tl) scintillators (Ref. 56). However, due to size limitations their usefulness is not very great. The problems seem to be in the development of methods to make materials of sufficiently high purity, with good crystal perfection, and with accurate stoichiometry.

The desire to combine, in one detector, the advantages of room temperature operation and efficiency of the scintillators with the energy resolution of germanium detectors has motivated the research into high-Z room temperature semiconductor gamma radiation spectrometer detectors. CdTe and HgI₂ detectors represent the areas of greatest progress. A recent review (Ref. 57) discusses their history, general properties, nuclear spectroscopic performance, and device application. "In spring 1978, CdTe and HgI₂ detec-

tors are limited to areas of 100mm^2 and to a thickness of 2 and 0.7mm , respectively. This gives a gamma ray interaction efficiency for CdTe and HgI_2 , respectively, of about 99.9% and 99% for energies below 50 keV, 78% and 76% for 100 keV, and 6% and 4% for 662 keV. The optimal energy resolution is 18% and 8% at 6 keV and 1.3% and 0.7% at 662 keV for CdTe and HgI_2 , respectively" (Ref. 57).

Other materials were discussed in Ref. 56, such as AlSb.

● Work on high-purity germanium has been actively continuing. A group at Lawrence Berkeley Laboratory (Ref. 58, 59) has defined the nature of the difficulty in obtaining high-purity germanium as being due to the poor segregation of the impurities boron and aluminum (because oxygen and silicon form complexes with them which are electrically inactive). With proper attention to controlling Si and oxygen by appropriate choice of container and coating materials and ambient atmosphere, they have succeeded in reducing Al and B to levels below 10^{10}cm^{-3} . Other studies (Refs. 60, 61) have shown the importance of the absence of hydrogen in keeping net impurity concentration of "fast defects" low.

Although high-purity germanium detectors are commercially available, they are still in development (Ref. 62) and more improvement can be expected. For example, it has been found (Ref. 63) that having the p^+ contact on the coaxial periphery for high-purity germanium yields significant improvement in radiation damage resistance. Furthermore, it has been found (Ref. 64) that large germanium detectors and an array of Ge detectors can operate up to 160°K with good resolution. These results open doors to new applications and improvements. (See Ref. 35 for a discussion of some of these applications).

Intrinsic germanium detectors with volumes as large as 180 cm^3 have now been fabricated, and resolution is comparable to that obtainable with conventional lithium-drifted germanium detectors of similar size (Ref. 36). Research is now aiming toward larger volumes, and even lower impurity concentrations. Among the advantages of high-purity germanium is ease of handling; it can be warmed up to room temperature without harm (except for possible surface contamination).

● Research to improve the resolution of scintillation counter systems is proceeding mainly in two directions. First, the development of improved photomultipliers using materials with higher photocathode quantum efficiencies and higher gain in the multiplier dynodes has steadily been improving their performance. This work is continuing. Second, the search for new scintillation phosphors might lead to improved resolution if it is possible to decrease the average amount of ionization required to produce a photoelectron at the tube cathode. One particular avenue is the search for phosphor photocathode combinations operating efficiently in the near infra-red region ($\sim 800\text{nm}$).

● Research on improved gas proportional counters probably cannot lead to very significant improvements over current performance: the intrinsic limitation for low-energy x rays is the value of about 25 eV/ion pair required in typical gases. For high-energy gammas, the detection efficiency cannot be very great because gases (even at high pressure) are not sufficiently dense. This has led to research in the use of liquids as the ionization medium. One research program in this area is that using liquid xenon (Ref. 65), where feasibility of such detectors has been demonstrated. The goal is the potential availability of dense proportional counters of large size, high detection efficiency, and good resolution (for xenon, $\epsilon \approx 2\text{ eV/ion pair}$).

● Work has demonstrated the feasibility of gas proportional scintillation counters (Ref. 66, 67, 68). The concept is a gas counter operating in the ionization mode rather than the proportional mode. A photomultiplier detects the light emitted by the ionized electrons as they accelerate in the applied electric field. The counter relies upon the existence of an accelerating electric field lower than that inducing ion multiplication but high enough to produce secondary light. The phototube then acts as the 'amplifying' element. Resolutions of about 470 eV (full width at half maximum) for the 5.9-keV ^{55}Fe x ray have been obtained (Ref. 68). These counters, while still in the study/development stage, show considerable promise for even better resolution at the lowest energies. For example, resolutions of 100 eV have been obtained (Ref. 68) for carbon x rays (277 eV), 230 eV for aluminum x rays (1.49 keV).

5. Data Handling, Reduction, and Calibration

a. Pulse-Height Analyzers

The instrumental output of all the detector systems discussed here is an electronic pulse whose integrated pulse area is nominally proportional to the ionization energy deposited in the detector. The preamplifier and amplifier system usually converts pulse area to pulse height of a standard-shaped pulse. The next component is often a device to select or sort pulses from the amplifier.

(1) Threshold Discriminators. The simplest device is a threshold "discriminator" that permits counting of all pulses above a certain height (hence, all photons above a certain threshold energy). This type of measurement is often termed a "gross gamma" measurement and is not to be considered spectrometry except in a very restricted sense. This technique is useful for some applications, but what is being counted is highly dependent upon the spectrum composition. Finite energy resolution smears the exact meaning of the pulse-height cut-off, especially for low resolution systems like NaI(Tl). Even more important is that the cut-off eliminates some fraction of the Compton continuum; the fraction varies considerably as a function of gamma primary energy. Thus "gross gamma" measurements are not easily interpreted except as semi-quantitative indications of total activity in the sample.

(2) Single-Channel Analyzers. Another pulse-height analysis device is a single-channel analyzer, which counts pulses whose sizes are between two preselected values. This pre-selected region is usually termed the pulse-height "window". This type of analyzer is used primarily to count one specific photopeak to the exclusion of others. Again, the interpretation of data taken in this way may be ambiguous: the continuum underneath the selected pulse-height region must be understood before quantitative data can be obtained. The continuum can arise from gamma rays of higher energy which Compton-scatter in the detector, from external backgrounds, and from noise. In order to help correct for the latter two of these effects, it is common to

take background runs with no sample present. Indications of the amount of Compton continuum present can be ascertained by counting all events with pulse heights above the upper threshold setting of the 'window'. This can help to assure that there are not very large numbers of counts at high energies if none are expected.

The single-channel analyzer is most commonly used when the analyst is confident that there are no significant interferences: for example, after chemical separation has isolated one chemical species prior to counting. Often a multichannel analyzer is used to characterize the spectrum prior to the use of single channel analyzers.

(3) Multi-Channel Analyzers. A multi-channel analyzer converts the pulse height to a digital form and then stores that information in one of its many 'channels'. Each channel corresponds to a certain small pulse-height region or 'window' and hence to a certain small energy region. Today, analyzers with 4096 channels or more are routinely used in Ge and Si semiconductor spectrometry, while 400 channel analyzers are common for NaI(Tl), CsI(Tl), and gas-proportional counter spectra. A great deal of literature is devoted to the properties of such instruments, but little of it is recent and we shall not discuss it in detail here. The 1965 review by Goulding should be consulted (Ref. 69) for the basic principles; more recent developments have incorporated integrated circuitry, higher speeds, and more sophisticated features, such as the ability to sum channels.

A basic multi-channel analyzer has three parts. The first is some type of pulse-height analyzer that converts incoming pulse height information to a digital form; the second is a storage unit and controller; and the third is a system for displaying and transmitting the data. Besides these basic functions a wide variety of special features is available on commercial instruments, including methods for storing in only a part of the memory; for transferring memory directly to a computer; for gating on externally-supplied signals; for using various types of displays; for subtracting background spectra automatically; and for computer control of all of the above. These functions are important for the efficient handling of the large masses of data which can be accumulated by the

detectors themselves. In some systems, small computers are used to perform many of these functions.

In order to aid in the choice of these instruments, we shall make a few brief comments about particularly relevant modes of operation and cite some common problems.

Dead-time, which is the time during which the analyzer is processing a given pulse and hence will not accept any other pulses at its input, is a parameter that must be borne in mind at high counting rates. The analyzer dead time is always far more important than the dead time from considerations of pulse overlap elsewhere in the system. Advanced systems automatically display dead times, gate out the system properly during the dead periods, and have live-time clocks. However, the user should be aware that this is one of the important possible sources of systematic error.

Calibration techniques for the analyzers themselves should be available and used. Again, the user should recognize possible problems. Perhaps the best calibration method is to calibrate the entire system with a known radionuclide source. This will be discussed below.

Computer-compatible input and output will continue to grow in importance, as systems are expanded to include computer interfacing on both the input and output ends. These (often expensive) features are not always as versatile as might be desired, for example if interfacing to a different computer is necessary some time after the original purchase.

One commonly used mode of operation in radionuclide analyses is the 'gated' mode. An anticoincidence gate is an integral feature of many NaI(Tl) and Ge systems. This feature can lead to systematic error. For example, the anticoincidence unit can fail to perform its task because of dead-time, overlapping pulses, or other considerations.

Long-term stability is another parameter of importance, since many environmental samples must be counted for long time periods. Advances have been made in this regard and the purchaser of a new analyzer system can now usually acquire a system with stability exceeding his requirements. The stability of the detector system remains as a key consideration.

The user must pay attention to all of these areas in order to take fullest advantage of whatever system is being used.

b. Data Reduction and Analysis

The data-reduction aspect of the measurement system can lead to much difficulty and error if not carried out properly. Therefore, a number of sophisticated methodologies have been developed, some of them almost fully computerized.

Two basic limitations of all detector-analyzer systems are responsible for most of the difficulties. First, the finite resolution leads to overlap of peaks in complex spectra. This is particularly true with the scintillation counters. Second, each spectrum invariably includes some background or "noise" counts (for example from the Compton spectrum of a higher-energy gamma ray or from external backgrounds) which decreases the resolution of the peaks.

The simplest procedure for dealing with a complex spectrum is to analyze successively the various photopeaks beginning with that at the highest energy. Successive "stripping" subtractions are made. Unfortunately, error analysis is extremely difficult after even a few subtraction steps, so that with complicated spectra the uncertainties become large and very difficult to estimate for the lower-energy peaks. For this reason, simple 'stripping' is used only for the least complicated spectra.

A somewhat more powerful method is to generate a set of simultaneous equations for a number of radionuclides in terms of their observed photopeak count totals. Again, uncertainties can become very large for the smaller photopeaks or those at lower energies above large backgrounds.

Both of the above are unsophisticated methods, typically using only the full-energy-peak information and estimating backgrounds beneath the peaks by measurements in adjacent channels. More elaborate analysis techniques attempt to fit an entire spectrum including Compton-scattering contributions.

One of the most widely used methods of computer analysis is the "method of least squares". Essentially, this

method uses as input a set of standard spectra taken with the instrument in question and containing known activities of all radionuclides suspected of being present in the unknown sample. A least squares fit is then performed on the data; the variances are minimized and the most likely concentrations of each sought-for radionuclide are determined. The principle as applied to gamma spectra has been thoroughly discussed (Refs. 70-72) and a sample computer program is available (Ref. 73). Among the disadvantages of this method are the requirements that all sought-after radionuclides must have reference input spectra available, and that those thought to be present must be estimated as a starting point for the analyses.

"Multiple regression" is an analysis technique which combines the concepts of 'stripping' and 'least squares'. This procedure is described by Shumway (Ref. 74), using methods developed by Efroymsen (Ref. 75). Essentially, the method begins with a least-squares fit to the most prominent highest-energy photopeaks, and their accompanying full spectra. The less prominent and lower energy radionuclides are then introduced sequentially, with recomputation of "least squares" each time to improve the goodness of fit. Each successive recomputation can be programmed to eliminate a radionuclide previously included. Again, a library of standard spectra must be used.

The techniques just described are most useful for complicated, overlapping spectra such as NaI(Tl) spectra containing several radionuclides. For Ge spectra the peaks are usually narrow enough that backgrounds beneath them can be determined reasonably well from measurements in nearby channels just to the sides of each peak. This is usually what Ge analysis programs do, with less consideration being given to the full spectra. An additional refinement in Ge analyses is the use of peak-shape information to determine more exactly the entire peak area. Some analysis programs, tied to particular measurement systems, are almost completely computerized and automated; they can find peaks, determine the peak sizes, and even correct for detector efficiency and energy calibration effects. These depend, of course, on access to libraries of standard spectra. Examples are analyses described by Routti and Prussin (Ref. 76) and Gunnink and Niday (Ref. 77). Other sophisticated methods have been developed for analysis of Ge spectra in neutron activation

analysis (Ref. 78). The reader is referred also to several other papers for more details (Ref. 79-84).

c. Calibration Standards

The most desirable method for calibrating any gamma spectrometry system would be to use a standard source containing known activities from each of several radionuclides; the source should reproduce exactly the geometrical and physical characteristics of the unknown sample measured with the same system. The purpose of using a known standard source is to reproduce as much as possible the same operating conditions used in the actual measurements.

A set of published gamma ray energies from 45 to 1300 keV that are useful for germanium spectrometers have been evaluated to be consistent with both a new reference value for gamma ray energy scale and the 1973 least-squares adjustment of the fundamental constants. This new energy scale is based on recent measurements at the National Bureau of Standards of the absolute gamma ray wavelength for the 411 keV line from ^{198}Au (actually 411.804 41 ± 0.00108 keV) (Ref. 85). At approximately 1 MeV the proposed energies typically have measurement uncertainties of 4 eV and energy scale error of 3 eV compared to 14 eV and 17 eV in a previous publication (Ref. 86).

For this calibration function, the U.S. National Bureau of Standards furnishes some known standards (Ref. 87). Several of the NBS standards contain a known activity of a single radionuclide, and in addition there is a series of standards each containing several gamma ray emitters spanning the energy range from 88 keV to 1.8 keV. The reader is referred to a suggested method for calibration of germanium detectors by the American National Standards Institute (Ref. 88).

In calibration measurements, care must be taken to avoid or minimize error from a wide variety of problems. Perhaps the most difficult to correct for are geometrical effects from differences in source location and source size. This is especially true for the lower-energy photons, for which self-absorption can occur.

6. Laboratory Measurements: Sensitivity

Although some types of gamma spectroscopic measurements are made in the field, most such measurements are made in the laboratory on samples brought from the field. The field measurements are treated below in a separate discussion. Here we shall discuss the factors which influence the ultimate sensitivity of laboratory systems. For a discussion on quality control for environmental measurements, using gamma-ray spectrometry, the reader is referred to an EPA document (Ref. 89).

There are two classes of measurement aims which are so different that one might imagine completely different measurement systems for them:

- The sample to be measured may be known to contain only one radionuclide whose concentration must be determined.
- The sample may be suspected of containing an unknown number of radionuclides which must be identified. Presumably the activities of the unknown can vary widely in magnitude.

An example of the first category might be the measurement of iodine-131 extracted from a milk sample using a complicated series of chemical steps. An example of the second category might be measurement on an air filter taken near the stack of a nuclear reactor.

Clearly, the choice of instrument may be different in these two cases. However, a typical radioanalytical laboratory is often confronted with both extremes as well as the entire range of measurement situations that lie in between.

a. Sensitivity for Measuring a Single Known Gamma

The most sensitive instrument system for measuring a single known gamma, at least when measuring gamma energies above about 100 keV, uses a large NaI(Tl) crystal for its detector. The ultimate in sensitivity is then limited, essentially, by the background counting rate above which a signal must stand out. As we have seen, many of the better low-background facilities are able to achieve background counting rates (in the 100 to

1000 keV range) on the order of 0.1 to 1 counts per minute per keV of energy acceptance. (Sensitivities as low as 0.01 cpm/keV are sometimes achieved.) As an example, the spectrum from NaI(Tl) for a typical monoenergetic line (say, ^{137}Cs at 662 keV) may be about 50 keV wide (FWHM), and the background counting rate over that energy acceptance may be 5 to 50 cpm (sometimes as low as 0.5 cpm). Clearly, activities yielding about this count rate are at the minimum detectable level [1 pCi = 2.2 disintegrations per minute]. Even with 1000-minute counts, detection limits below 1 pCi are achievable only in the very best low-background facilities employing detectors for Compton scattered events to aid in the detection efficiency. This estimated minimum detectable activity is increased by any inefficiencies, e.g., inefficiencies from the fact that not all photons convert their full energy into ion pairs in the detector, from solid-angle, and from other losses.

When more than one photon energy is present, all but the highest-energy photon may suffer additional sensitivity losses because the "background" beneath their full-energy peak(s) may be enhanced by the Compton continuum from the highest-energy line.

b. Sensitivity for Measuring a Large Number of Unknown Radionuclides

This measurement problem is often succinctly termed 'peak hunting'. We have already seen (for example, in Fig. 15) that NaI(Tl) systems are totally incapable of resolving small closely-spaced peaks in the presence of much larger ones. Hence for peak-hunting the better-resolution instruments are preferred, and we shall restrict our discussion to them, in particular, to Ge detectors.

The considerations are few in number but interdependent in a complicated way: better resolution makes a peak stand out more above background; higher detection efficiency yields more counts; lower background and small Compton-scattering efficiency both improve sensitivity; high peak-to-Compton and peak-to-total values are important. One trade-off is that although larger detector volumes yield higher efficiencies they may make Compton tails worse, degrading sensitivity in the low-energy region.

Also, large volumes usually imply some sacrifices in resolution.

Cooper (Ref. 34, 90) has combined many of these considerations into one overall 'figure-of-merit' (FOM), which can be calculated for any Ge system as a function of energy being measured. The FOM was developed to aid in comparing the performance of systems measuring known lines, but has application for 'peak-hunting' as well. Some of the results are surprising. For example, at lower energies (~ 122 keV) the FOM of even a very small 0.4-cm^3 detector is only 1.5 times poorer than that of detectors with 75-cm^3 volumes. We quote from Cooper (Ref. 34):

"Although the 1333-keV peak efficiencies differ by almost a factor of 800 at 25 cm, the 122 keV peak efficiencies differ by only about 8 times at the cap. In addition, the 0.4-cm^3 detector has a greatly reduced Compton interference and much better resolution. All of these factors contribute to its competitive sensitivity at 122 keV despite the factor of 16 difference in their cross-sections."

For low-energy applications, the small planar germanium detectors have definite advantages over the larger coaxial detectors when used in 'peak-hunting' mode.

The ultimate detection sensitivity depends on too many factors to enable a brief discussion here. The situation is in many ways similar to that for NaI(Tl) as used to measure a single radionuclide. For Ge systems the background counting rates from cosmic radiation tend to be smaller, but the gains are counter-balanced by lower volumes and smaller efficiencies. However, a reasonable rule-of-thumb is perhaps that Ge systems in peak-hunting mode (in the presence of many other peaks) are perhaps one order-of-magnitude more sensitive than NaI(Tl) systems with comparable accoutrements (comparable low-background facilities; comparable electronic sophistication; etc.).

7. Field Measurements

Although most gamma spectrometry is performed in the laboratory on samples brought to the apparatus, a significant effort has been devoted in re-

cent years to the development of field measurement systems. There are three types: portable, mobile and stationary. The aims of the measurements can be to estimate either dose distributions or radionuclide distributions, sometimes even elemental distributions.

The reason for field measurements is usually to enable sensitive study of large areas. Because there are some measurements which cannot be made in any other way, these techniques have proven to be of value despite their drawbacks (chiefly lack of precision, logistical difficulties, and cost).

We quote from Beck et al. (Ref. 91), who have stated well the advantages of field measurements:

"In situ measurements of soil activity are more sensitive and provide more representative data than data obtained by sample collection and subsequent laboratory analysis. An unshielded detector placed about one meter above the ground detects gamma rays from an area within about a 10 meter radius, representing a large volume of soil compared to the typical soil sample, and comparable counting statistics can be obtained in only a small fraction of the time required for the laboratory analysis. For example, a field spectral analysis for the natural emitters, ^{40}K , ^{238}U , and ^{232}Th , can be carried out in approximately 15 minutes with a 4 in. by 4 in. NaI(Tl) detector. A comparable analysis in the laboratory, excluding transportation and sample preparation time, would require several hours. Furthermore, a single soil sample from a site may not be representative of the mean soil activity, so a number of samples or composite samples are required. A single field analysis averages out small local inhomogeneities in the sample."

The typical stationary system (Refs. 92, 93) is carried to the field, set up with the detector some distance ($\leq$ few meters) above the ground, and allowed to count whatever the detector finds. An energy spectrum results, but the problem involves the unravelling of the true radionuclide concentrations in the media being studied. There are three different sources of events: the air, the ground, and the cosmic radiation. Because the penetration of gamma rays is greater at higher energies, the detector samples larger soil areas,

deeper soil depths, and greater air volumes at high energies than low. Furthermore, because of the geometrical situation, any of the gamma rays being detected has a fair chance of coming from a Compton-scattering event rather than being a primary gamma ray from a mono-energetic source. These Compton gamma rays cannot be eliminated by alterations of the detector. This set of complications degrades the intrinsic accuracy, but perhaps the major difficulty is that the analysis depends on knowledge of the radioactivity distribution with soil depth, and to a lesser extent on soil composition (Ref. 94).

Various analysis methods have been developed for this technique. That of Beck et al. (Ref. 91) is typical: given the soil density and chemical composition, and having calibrated the detector response as a function of energy using NBS standards, an integral is performed taking into account the absorption, angular-response corrections, and Compton-scattering corrections. The result may be expressed as activity per unit area ($\mu\text{Ci}/\text{m}^2$); activity per unit volume, assuming a uniform depth distribution ($\mu\text{Ci}/\text{m}^3$); or a special-

ized activity distribution if detailed soil composition is known separately. A similar approach is described by Phelps et al. (Ref. 95).

Field measurements are also sometimes done on the gaseous and liquid effluents from nuclear facilities. This can be useful in correlating levels with normal operations and procedures (Refs. 96-105). See Table 1 in Ref. 100, for an example of the sensitivity for detecting certain radioactive gases. References 106 and 107 describe the advantages and limitations of soil sampling versus in-situ gamma counting.

With the development of high-purity germanium and detector arrays, it has been possible to expand the use of photon spectrometry to measure low energy x rays such as emitted by the transuramics (Refs. 108, 109).

Typical in-situ measurements with Ge and NaI(Tl) systems are shown in Figs. 32 and 33; both spectra were taken at the same site (Ref. 110). Clearly, the detail observed in the Ge spectrum is vastly superior to that in

Fig. 32. In situ spectrum, northeastern U. S. location, taken in 1971 with 10 cm x 10 cm NaI(Tl) crystal, 20-minute counting time. (Reproduced from Beck, Lowder, and McLaughlin, Ref. 110, with permission of the International Atomic Energy Agency, © 1971).

Fig. 33. In situ spectrum, taken with that in Fig. 32 with 60-cm³ Ge(Li) detector, 40-minute counting time. Photon energies in keV. (Reproduced from Beck, Lowder, and McLaughlin, Ref. 110, with permission of the International Atomic Energy Agency, © 1971).

the NaI(Tl) spectrum. Considerably more equipment is required for the Ge system, and this restricts the convenience and mobility; a large capacity multi-channel analyzer, a liquid nitrogen system, special amplifier electronics, extra power, and a more bulky and fragile detector set-up all make Ge systems logistically difficult. Whenever a NaI(Tl) system can provide sufficient information (for example, to study backgrounds in remote areas), its use is preferable. However, with the availability of high purity germanium detectors simpler cooling systems have been developed, so that more portable systems are being used (Ref. 105).

For truly 'portable' measurements (as contrasted to 'mobile' measurements using a small van), a Ge system is essentially ruled out. Some improvement on portability has been achieved for high purity germanium detector spectrometry systems (Ref. 111).

There have been a number of portable NaI(Tl) systems developed for hand-carriage in the field, some for general

gamma spectrometry and some for special applications. An example of the latter is the 'FIDLER' detector (Ref. 112), developed to survey for the 60-keV gamma rays from americium-241 as a means of assessing plutonium deposition on soil. This device consists of a thin, disk-shaped NaI(Tl) crystal viewed by a phototube, the whole assembly carried over the terrain about 50 cm above the soil to search for hot-spots of activity. The section on "Plutonium" elsewhere in this volume contains a more detailed discussion. Several other NaI(Tl) instruments for measurements of natural environmental radiation are described in the compilation by Adams and Lowder (Ref. 113).

8. Summary and Conclusions

This section has presented an overview of the main parameters which affect the operating characteristics of gamma-ray spectrometers. The emphasis has been on systems useful for meas-

uring low-level gamma radioactivities in various environmental media. The presumption has been made that the user desires to optimize his ability to measure minute concentrations.

The two main considerations in choosing among the various system types are efficiency and energy resolution. These two features are typified by the two most widely-used kinds of systems: NaI(Tl) systems with high efficiency, and Ge systems with excellent resolution. Other kinds of detectors are useful for particular classes of measurements, in particular Si(Li) detectors and gas proportional counters for measurements at low energies.

Because a gamma-ray spectrometry system can be one of the most expensive pieces of equipment in any radioactivity analysis laboratory, the user must be aware of the trade-offs among the several parameters. We have tried to illustrate most of these by reference to typical spectra obtained in various kinds of situations.

Perhaps the most revealing contrast is that between a NaI(Tl) system, on the one hand, used in the mode where one (or a few) gamma lines need to be measured at very low levels; and a Ge or Si system, on the other hand, used in the 'peak-hunting' mode on a sample containing a large number of unknown radionuclides. Neither of these systems could adequately do the other system's job!

It appears that the most important possible improvement foreseeable in the near future would be further development of large, ultra-pure germanium detectors. Larger detector sizes would give the semiconductor detectors sufficient overall efficiency to compete with NaI(Tl) for even the lowest low-level measurements; the superior resolution would enable even the largest detectors to resolve gamma lines far beyond the capability of any other existing detector type.

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E. INSTRUMENT NOTES: GAMMA SPECTROMETRY

Gamma spectrometry instrument notes will be found behind the X and gamma instrumentation section (section III. D.).

III.F. Neutron Monitoring Instrumentation

1. Introduction
2. Radiation Protection Considerations
3. Measurement Considerations
 - a. Thermal Neutrons
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1. INTRODUCTION

There are a large number and variety of applications where a significant amount of radiation is due to the presence of neutrons. Among these are:

- a. charged particle accelerators, particularly proton and ion accelerators and high-energy accelerators;
- b. nuclear reactors;
- c. sealed neutron sources.

These sources vary widely in the intensities, energy spectra, and temporal durations of the neutron fluences involved. From the highest-energy proton accelerators, primary beam interactions can produce neutrons with energies up to the full energy of the primary protons themselves, while in reactor situations there can be a heavy weighting toward thermal neutrons. Given this wide energy range, it is remarkable that one can give any unified treatment to the general problems of measurement of neutron fluences. However, a unification is indeed possible, for a large class of applications, primarily because of the characteristics of neutron interactions in shielding materials.

Neutrons traversing through material can interact in one of five general ways:

- a. elastic scattering from nuclei;
- b. inelastic scattering through nuclear-level excitation;
- c. quasi-two-body interactions such as (n,α) or (n,np) interactions;
- d. nuclear excitation or catastrophic break-up;
- e. the high-energy phenomena of many-particle production.

The relative importance of each of the five general types depends on both neutron energy and target material. Thus, elastic scattering from heavy nuclei is a very inefficient energy loss mechanism, as is easily seen from kinematical considerations. However, elastic scattering from hydrogen nuclei (protons) is one of the most useful

processes in degrading neutron energies in shielding (hence, the common use of concrete). Also, at energies much higher than those of typical nuclear levels (a few tens of MeV, say) a major energy loss mechanism is nuclear break-up or (above a few hundred MeV) multiparticle interactions, such as pion production. These facts have long been well known, and have influenced shielding design around major sources of neutrons and other particles. The unification of treatment mentioned above is primarily due to the effectiveness of energy-degradation processes in appropriately designed shielding, such that even at the highest energy accelerators the major components of neutron fluence are almost never at energies above a few tens of MeV.

Neutron interactions, then, generally tend to degrade energies continually, down to the region where the neutrons can be absorbed and eliminated by nuclear reactions. The ultimate energy degradation is neutron 'thermalization'; thermal neutrons are those with an energy spectrum in thermal equilibrium with the ambient material. Thermal neutrons can scatter without energy loss in, for example, air or concrete for relatively indefinite periods; while thermalization of a several-MeV neutron can take from a few to a few hundred microseconds, thermal-neutron lifetimes themselves can be as long as a few tens of milliseconds, until they are ultimately eliminated by nuclear absorption or reactions such as the (n,α) mentioned above.

Historically, it has been common to divide the neutron energy spectrum into the following approximate categories:

Thermal neutrons	0 to 0.5 eV
Intermediate energy neutrons	0.5 eV to 200 keV
Fast neutrons	200 keV to 20 MeV
Relativistic neutrons	>20 MeV

This division is convenient both because of the different types of processes which are typical of these energy regions, and because instrumentation is often sensitive in only one of the categories. We shall discuss instrumentation partially according to these divisions, noting that in many applications one has a priori knowledge

that the neutron energy extends up to only a particular energy region.

For detailed information about techniques of neutron dosimetry, the reader is referred to several excellent papers and conference proceedings. (Ref. 1,2,3,4,5,6)

2. RADIATION PROTECTION CONSIDERATIONS

In the general case of occupational exposure to external radiation, the relevant Maximum Permissible Exposure guideline (discussed in more detail in the introductory part of this volume) is a basic limit to dose-equivalent of 5 rem/year, with no more than 3 rem to be accumulated in any 13-week quarter and no lifetime accumulation up to age N years in excess of 5 (N-18) rem.

The complication in measuring neutron rem (absorbed-dose equivalent) is the variation of quality factor (QF) with energy. Figure 1 shows the ICRP recommended QF for neutrons as a function of energy. Figure 2 gives the flux density of primary neutrons, in equilibrium with their secondaries, which is recommended by ICRP as being equivalent to 2.5 mrem/hour (Ref. 7). The inverse curve (Figure 3) is the energy response required of a neutron detector such that different energies are weighted properly for a direct determination of dose-equivalent (rem).

3. MEASUREMENT CONSIDERATIONS

There are three main tasks in neutron-monitoring:

1. Measurement of the intensity spectrum as a function of energy (neutrons $\text{cm}^{-2} \text{MeV}^{-1} \text{sec}^{-1}$).
2. Measurement of the total absorbed-dose equivalent (rem or rem/sec).
3. Determination of the level of other possible particle species, and the response of neutron instrumentation to them.

There does not exist a single instrument or instrument type which can perform completely all of these functions (or indeed, any one of them separately). However, there does exist instrumentation more or less

adequate to the needs of the health physicist responsible for radiation protection around any of the sources mentioned above. The wide range of energies and fluences previously mentioned must be considered, along with several other concepts such as tissue-equivalence and particle equilibrium, which have been treated in the introductory section of this volume.

A. Thermal Neutron Measurements (below 0.5 eV)

Truly thermal neutrons, which have an energy spectrum in thermal equilibrium with the ambient environment, are centered at an energy of about 0.025 eV. However, in actual practice, very few situations exist in which the neutron fluence is solely thermal: one possible case is that in which neutrons are almost totally thermalized for some application such as a thermal column. Many detection systems for higher-energy neutrons depend upon the process of (partial or complete) thermalization, with subsequent detection of the thermal neutrons using one of the techniques to be discussed in this section.

There is usually both a truly thermal (Maxwellian) component and a component with a continuous energy spectrum above thermal, typically with a ("epi-Maxwellian") distribution inversely proportional to energy. Also, most of the reactions used to determine thermal neutron flux densities have cross-sections which are almost exactly proportional to $1/\text{velocity}$ in the range from thermal up to the cadmium cut-off energy. (Since a cadmium absorber prevents neutrons below about 0.5 eV from reaching a detector, it allows a convenient separation at about that energy.) Because flux density (neutrons $\text{cm}^{-2} \text{sec}^{-1}$) is proportional to velocity (if they are going twice as fast, they cross any area in half the time), the reaction rate per atom for a pure $1/v$ detector is independent of velocity and proportional only to the total neutron density. Table 1 (from Ref. 5) shows most of the common thermal neutron reactions; the coefficient $g(20^\circ\text{C})$ in the right hand column indicates the departure from $1/v$ dependence for each reaction.

FIGURE 1: QUALITY FACTOR vs. ENERGY for monoenergetic neutrons, as recommended in ICRU Report No. 20.

FIGURE 2: MONOENERGETIC NEUTRON FLUX vs. ENERGY which yields MADE of 1 mrem/hour.

FIGURE 3: NEUTRON DOSE (rem) vs. ENERGY for 1 neutron/cm² (for monoenergetic neutrons incident normally on a 30 cm-thick tissue-equivalent phantom: ICRU Report No. 20).

Table 1
Useful Thermal Neutron Detectors

Nuclide or Element	Reaction	Half-Life of Nuclide Produced	$\sigma \times 10^{24}$ for Production cm^2	$g(20^\circ\text{C})$
^3He	$(n,p)^3\text{H}$	12.3 y	5327	1.0
^6Li	$(n,t)^4\text{He}$	stable	945	1.0
^{10}B	$(n,\alpha)^7\text{Li}$	stable	3837	1.0
$^{23}\text{Na}(\text{IS})$	$(n,\gamma)^{24}\text{Na}$	15.0 h	0.534	1.0
$^{45}\text{Sc}(\text{IS})$	$(n,\gamma)^{46}\text{Sc}$	85d	22.3	
^{51}V	$(n,\gamma)^{52}\text{V}$	3.8 min	4.9	1.0
$^{55}\text{Mn}(\text{IS})$	$(n,\gamma)^{56}\text{Mn}$	2.58h	13.3	1.0
$^{59}\text{Co}(\text{IS})$	$(n,\gamma)^{60}\text{Co}$	5.24y	36.6	1.0
^{63}Cu	$(n,\gamma)^{64}\text{Cu}$	12.8h	4.5	1.0
^{115}In	$(n,\gamma)^{116}\text{In}^m$	54 min	157	1.019
^{157}Gd	$(n,\gamma)^{158}\text{Gd}$	stable	242000	0.854
$^{197}\text{Au}(\text{IS})$	$(n,\gamma)^{198}\text{Au}$	2.70d	98.8	1.005
^{235}U	fission	many	577	0.976

Notes: (IS) denotes natural monoisotopic element.

For a more detailed discussion of thermal neutron spectra, departures from Maxwellian distributions, and neutron temperature the reader is referred to ICRU Report No. 13 (Ref.5).

The most commonly used thermal neutron detector is the BF_3 filled gas proportional counter, which is commercially available in a wide variety of sizes and sensitivities. The reaction employed is $^{10}\text{B}(n,\alpha)^7\text{Li}$, in which the alphas are detected. The Q (kinetic energy released) is 2.8 MeV, but most of the capture goes through an intermediate excited state of ^7Li , which then decays to the ground state by emission of a 470 keV gamma. In most counters the BF_3 gas is enriched with boron-10 to about 96%. (Natural abundance is 20% ^{10}B and 80% ^{11}B). The cross-section of the above reaction is about 3800 barns with a $1/v$ dependence in the thermal-energy range. Thus BF_3 counters are relatively insensitive to particles with energies significantly above that of thermal neutrons (0.025 eV). The sensitivity depends on counter size, gas pressure, and enrichment. The response to gamma fields is very low. In fields of equal gamma and neutron fluxes, the gamma

response is typically two orders of magnitude smaller than the neutron response (Ref.1). An exception is when the BF_3 detector is inside a moderator, and pulsed fields are being detected (such as near a low duty-cycle accelerator). In that case, photons can pile up and count.

The ^3He gas proportional counter is another commonly used detector, similar to the BF_3 counter. It relies upon the reaction $^3\text{He}(n,p)^3\text{H}$, with a Q value of 765 keV. Both the proton and the triton (^3H) are available for ionization, but one of the problems is that near the walls efficiency can suffer, due to the rather low stopping power of the ^3He gas. This deficiency can be overcome either by increasing the volume or gas pressure, or (more commonly) by adding an impurity (usually argon or krypton) to improve the stopping power. The disadvantage of the impurity is that gamma sensitivity is increased, while pulse rise-time is decreased. Because of the small Q value, gamma sensitivity is greater than in the BF_3 -filled counter, but neutron sensitivity is also greater (Ref.1), until the recoil particles become too long (at neutron energies above about 10 MeV).

For neutron detection in high proton fields, ^{10}B -lined proportional counters are used. The reaction is the same as for BF_3 counters, except that the ^{10}B is coated on the counter walls, in thicknesses that range from about 0.2 to 1.0 mg/cm². Sensitivity is proportional to coating thickness, but self-absorption of the alphas prevents significant increases in sensitivity at greater thicknesses. These counters are extremely insensitive to photons, and are capable of neutron-sensitive operation in photon fields as high as 10^4 or even 10^5 R/hr. However, their sensitivity to neutrons is perhaps an order of magnitude smaller than that of the BF_3 counter (Ref. 1). A typical fill gas is CO_2 or Argon. The operating voltage is usually lower than that of the BF_3 or ^3He tubes, and typically larger output pulses with shorter rise times are produced.

Still another gas proportional neutron detector is the fission counter. A material such as ^{233}U , ^{235}U , or ^{239}Pu is coated on the walls. The kinetic energy of the fission fragments (approximately 200 MeV) produces large pulses and fast response time. The sensitivity of these counters is inherently lower by perhaps an order of magnitude than the ^{10}B lined counters, and two orders of magnitude below that of the BF_3 tubes (Ref. 1). However, operation at temperatures as high as 700°C and in extremely high gamma fluxes (up to 10^6 R/hr) makes them attractive for applications such as inside reactor shielding. It should be noted here that special licensing is often required to obtain the fissionable material within the counter.

Ionization chambers are also used for thermal neutron detection. Both ^{10}B -lined and BF_3 -filled ion chambers are used. Many of the same characteristics apply which have already been discussed in the case of the gas-proportional counters. The ion chambers are generally less sensitive than the gas proportional counters (Ref. 1). A fuller treatment of ionization chambers can be found below.

Scintillation counters (in which photomultiplier tubes detect the light produced by ionization in scintillator) are also used for detection of thermal neutrons (Ref. 1). Among the most sensitive are a lithium-iodide crystal with europium activation [$^6\text{LiI}(\text{Eu})$]; ^6Li -loaded, cerium-activated silicate glass; and ^6Li -loaded, zinc-sulfide crystals. In each case one counts the alpha and triton from the reaction $^6\text{Li}(n,\alpha)^3\text{H}$, with a Q of 4.8 MeV. If the detector is large in size there can be a significant perturbation of the field being measured. However, the gamma rejection is excellent, with gamma sensitivities typically one to two orders-of-magnitude below those for neutrons. The crystal itself (and the others as well) have negligible anisotropy of sensitivity, except for self-shielding and the flux perturbations already mentioned. A description of some of the properties of cerium-activated glass has been given by Ginther (Ref. 8); and a discussion of ^6Li -loaded glasses is given by Bormann et al (Ref. 9).

Another technique is the activation of a NaI (Tl) crystal, in which one counts the decay after activation of ^{24}Na (15 hour half-life, β with $E_{\text{max}} = 2.12$ MeV). An advantage of this technique is the good detection efficiency for the decay betas. The sensitivity is very low, perhaps 10^4 below that of the others previously mentioned. Although common at one time, this technique is not often used today. Small crystals (say, 1 mm thick) must be used to minimize the problem of perturbation of the incident flux. For a good description of the NaI method, the reader is referred to Grimeland (Ref. 10).

Zinc sulfide glass with ^{10}B loading is another material used as a scintillator; however, it suffers from two problems (Ref. 1) when compared to its counterpart, ^6Li -loaded ZnS: first, poorer gamma discrimination, because of the lower Q value of the (n, α) reaction; and second, typical sensitivities several orders of magnitude lower than for the lithium-loaded crystals. Thus, only in situations where neutron fluences are high and photon levels are known to be small can the ^{10}B -loaded ZnS be useful.

Foil Activation is another common technique for detection of thermal neutrons. Table 1 above showed the properties of some of the useful reactions. Perhaps the most widely used foil material when high sensitivity is required is indium, due to its very large thermal neutron cross-section. Use of cobalt is common for the integration of neutron fluence over very long periods, because of the long (5.2 year) half-life of ^{60}Co . Gold is used when high precision is required, since its capture cross-section is well-behaved for energies up to about 3 eV.

Thermoluminescence is still another detection technique for thermal neutrons (Ref. 11). The most commonly used thermoluminescence detector (TLD) for neutrons is ^6LiF , but this detector also has substantial gamma sensitivity (Ref. 11). Therefore, it is common practice when gamma backgrounds are present to use two or three TLD's with varying fractional concentrations of ^6Li and ^7Li , in order to determine the gamma and neutron contributions separately. TLD's are integrating devices which must be read out in a separate operation after exposure. The general discussion of TLD's is found in the section entitled "Personal Dosimetry." The sensitivity of ^6LiF is typically above 1000 neutrons/cm², and the sensitivity to fast neutrons is small. Another common TLD device is CaF_2 , but it has very small intrinsic neutron sensitivity and large gamma response; addition of ^{10}B or ^6Li is used to enhance the neutron response, but with the exception of pure neutron fields it is not recommended.

To summarize the discussion on thermal neutrons, it can be seen that there are many methods used for detection; however, complications can arise when higher-energy neutrons are present, since most thermal neutron detectors have at least some sensitivity to higher energies. When one suspects the presence of higher-energy fluences, a common technique to use is the "cadmium ratio" method. This involves shielding the detector with a layer of cadmium (an excellent absorber of neutrons below about 0.5 eV) and comparing detector response to the bare situation. To determine

the flux in the region near 1.5 eV, one can surround an indium foil with cadmium. The cadmium will absorb neutrons below 0.5 eV, and the indium foil (with its huge capture resonance of 26,000 barns near 1.5 eV) can be β -counted by any low background G-M counter to detect the beta from ^{116}In decay after the reaction $^{115}\text{In}(n,\gamma)^{116}\text{In}$. Accuracies within better than a factor of 2 in flux measurements can be obtained. A further description of this procedure can be found in the article by Hankins (Ref. 12).

B. Intermediate-Energy Neutrons (~0.5 eV to 200 keV)

Intermediate energy neutrons are most often detected by moderating them down to the thermal region and then employing one of the thermal methods just described.

Direct detection of recoils from neutron scattering does not work well in this energy region, because it is difficult to observe the recoils against even a small gamma ray background. Instead, the most common technique is to detect the energetic products of an exothermic reaction such as (n,α) . However, the cross sections for many of these reactions decrease rapidly at energies above a few eV, and thus neutrons in the keV range are usually detected by rather specialized means.

One such technique is the so-called "black detector" which absorbs all neutrons within a certain energy region. The most commonly used "black detectors" are a ^{10}B -doped liquid scintillator or a detector consisting of a boron plate which uses the associated photon from the (n,γ) reaction for detection (Ref. 13). The problem is that these instruments absorb all neutrons below a certain maximum energy. To derive absorbed dose-equivalent from flux density involves a detailed knowledge of the energy spectrum (see Fig. 3).

The moderator approach has been used with reasonable success. Moderators can be appropriately designed to yield either rem-equivalent response or flux density response weighted to yield values of absorbed dose. An

instrument with the response of Fig. 4 has been reported by Leake (Ref. 14); however, the figure indicates a substantial fall-off in sensitivity at high and low energies, necessitating a correction for neutrons outside of the sensitive range. This device uses a ^6LiI (Eu) crystal scintillator surrounded by polyethylene moderator. Another instrument (Ref. 15) uses a boron-loaded ZnS scintillator enclosed in a polyethylene sphere. Figure 5 (from Ref. 15) shows the energy response of this instrument for various moderator thicknesses. Note that the response can be made quite flat. Both of these instruments have excellent gamma discrimination (Ref. 1).

Instruments which cover the intermediate range in a rem-equivalent manner but are also sensitive over much broader energies will be discussed below in the section on dose-equivalent instruments.

C. Fast Neutrons (~ 200 keV to 20 MeV)

In situations where fast neutrons require measurement, there is usually an accompanying flux of lower energy neutrons, and there may be gamma backgrounds as well. For this reason, a frequently used technique for isolating the fast-neutron component is foil activation, because one can choose processes with a variety of thresholds and sensitivities.

FIGURE 4: The response of a dosimeter designed to measure dose equivalent of neutrons in the intermediate energy range.

FIGURE 5: Variation of efficiency and dose dependence of the optimum arrangement with neutron energy.

Activation of foils is also used for relativistic neutrons. Figure 6 shows the cross section as a function of energy for some of the commonly used reactions (Ref. 16). Table 2 (from Ref. 17) gives some properties of the reactions, and the half-lives of the activation products. The last two columns in Table 2 show the sensitivity for 1 rad, and the background, for the detector used by Cross (Ref. 17). The sensitivity with In, Fe, and Zn is perhaps an order of magnitude better than for the fission detectors, because a decay gamma peak stands out above background to a greater extent than when counting a wide spectrum.

Gas proportional counters which rely upon the measurement of proton recoils have found widespread use for

FIGURE 6: Response functions of high-energy neutron detectors.

fast neutron detection, because of their excellent gamma discrimination. Several such counters have been described in the literature. As an example, consider the counter built at Oak Ridge National Laboratory and described by Hurst and Wagner (Ref. 18). The counter, which reads out in rads, is completely lined with polyethylene (C_2H_5) and filled with ethylene (C_2H_4) to a pressure of 750 torr. Figure 7 (from Ref. 18) shows the response to 1.3 MeV Co^{60}

Table 2
Properties of Some Threshold Reactions

Reaction	E_{th} (MeV)	$T_{1/2}$	$\bar{\sigma}$ (b)	E_{γ} (MeV)	Sensitivity for 1 rad (cpm/g)	Background (cpm)
$In^{115}(n,n')In^{115m}$	1.2	4.5 h	0.3	0.335	121	14
$Ni^{58}(n,p)Co^{58}$	2.6	72 d	0.6	0.810	0.63	8
$S^{32}(n,p)P^{32}$	2.9	14.2 d	0.35	β	17	1
$Zn^{64}(n,p)Cu^{64}$	3.3	12.8 h	0.25	0.511	27	12
$Fe^{56}(n,p)Mn^{56}$	6.7	2.6 h	0.08	0.845	56	8
$Mg^{24}(n,p)Na^{24}$	7.2	15 h	0.15	2.76	13	1
$Al^{27}(n,\alpha)Na^{24}$	7.6	15 h	0.1	2.76	9	1
$In^{115}(n,2n)In^{114}$	10.6	50 d	1.4	0.192	0.46	17
$Cu^{63}(n,2n)Cu^{64}$	11.4	12.8 h	1.0	0.511	23	12
$Ni^{58}(n,np)Co^{57}$	12	267 d	0.7	0.122	0.58	20

E_{th} : Effective threshold energy (in MeV) at which cross-section reaches 25% of $\bar{\sigma}$.

$T_{1/2}$: Half life of product.

$\bar{\sigma}$: Average cross-section (in barns) over the most sensitive energy region.

E_{γ} : Energy of Decay γ (MeV)

Sensitivity: Counts/min per gram of detector for 1 rad of neutrons.

Bkgd.: Counter background (cpm).

FIGURE 7: Integral pulse-height curves for mixtures of gamma-rays and fast neutrons.

gammas (at rates from 1 to 100 R/hour) and to the neutron spectrum from a Po-Be source, at rates of 0.001 rad/hour. If one is willing to set a bias level to eliminate signals from neutrons below about 0.5 MeV, the gamma discrimination is excellent.

Another counter, filled with methane-argon, has been described by Dennis and Loosemore (Ref. 19). Its energy response is shown in Fig. 8. Although this counter has sensitivity which is roughly dose-equivalent (rem) in the fast neutron region, it suffers from inaccuracies due to a loss of counts below the bias level. Theoretical techniques have been developed to estimate corrections for these losses (Ref. 20), but some independent measurements are probably required when the neutron spectrum is unknown.

FIGURE 8: Average response of three type FN 2/3 counters as a function of neutron energy. The solid line is the response required according to the ICRP recommendations (ICRP Pub. 4) for a counting rate of 0.9c/sec/MPL. The dashed line is the theoretical response of the counter for end-on irradiation at a bias level equivalent to 0.09 MeV.

Other instruments of this type have been described by Anderson (Ref. 21) and Murthy (Ref. 22). Murthy's instrument can measure neutron fluxes up to 15,000 neutrons $\text{cm}^{-2}\text{sec}^{-1}$. These instruments can tolerate (Ref. 1) thermal/fast-neutron dose rate ratios of about 100/1 and gamma/fast-neutron dose rates of about 200/1.

Another fast-neutron detection technique is scintillation counting with plastic or liquid scintillators attached to a photomultiplier. Neutrons can be counted in a scintillator through their interaction and production of secondaries. The enhancement in high energy detection efficiency is accomplished by setting a rather high threshold. Figure 9 (from Ref. 23) shows the neutron detection efficiency of a typical liquid scintillator for various threshold values.

FIGURE 9: Neutron detection efficiency of a liquid scintillator, 4.7 cm x 4.7 cm, with thresholds of 2, 3, 4, and 5 MeV electron energy.

A problem with scintillation counting is that if any relativistic charged particles are present, they will count with nearly 100% efficiency. Therefore, to isolate the neutron component one must either determine that charged particles are absent or measure them separately.

Unfortunately, the photon response of some plastics, especially the organic plastics which contain significant fractions by weight of hydrogen, is also very large, making them unsuitable where gamma backgrounds are important.

Efforts have been made to overcome the problem of gamma sensitivity. The most useful method for distinguishing neutrons from gammas is pulse shape discrimination. As an example, Verbinski et al. (Ref. 24) have successfully differentiated the two species in the 1 to 10 MeV range using pulse shape discrimination on the response from NE-213 liquid scintillator manufactured by Nuclear Enterprises, Ltd. (Ref. 25). However, this technique suffers because it is delicate, and requires sophisticated electronic equipment.

Another fast neutron detector is the so-called 'Long Counter' originally developed in 1947 by Hanson and McKibben (Ref. 26), and improved in 1966 by DePangher and Nichols (Ref. 27). The Long Counter is a BF_3 proportional counter surrounded by two concentric cylindrical hydrogenous moderators, with a thermal neutron shield in-between. It has high sensitivity, good gamma discrimination, and wide energy response (from about 25 keV to 14 MeV). The most significant drawback of the counter is its non-isotropy of response; it is insensitive except for neutrons incident along the axis.

FIGURE 10: Cross-sections for the production of carbon-11 from carbon-12

As in the case of intermediate-energy neutrons, it is possible to moderate fast neutrons to thermal energies and apply any of the large number of thermal-detection schemes previously described. These instruments will be discussed in more detail in the section below on dose-equivalent rem-meters.

D. Relativistic Neutrons (Energies above 20 MeV)

These neutrons are almost never found at levels which can pose a potential hazard to personnel except at high-energy accelerators. Carbon-11 activation (20.4 minute half-life) is one of the most widely used techniques for measuring the high-energy neutron components of a radiation field. A carbon-rich detector is exposed to the radiation field and the decay positrons (0.97 MeV $E_{\max}$) are then counted by standard techniques. The threshold for ^{11}C excitation is about 20 MeV, and Fig. 10 (from Ref. 28) shows cross-sections for various particle types. Note that the (γ, n) cross-section is very small. Although the neutron cross-sectional behavior overemphasizes the region just above threshold, and high-energy charged particles can also produce ^{11}C , this technique is often useful for a

rough check on the possible contribution from high-energy neutron fluences. A quantity termed the "neutron flux density above 20 MeV" can be defined and estimated from the amount of Carbon-11 activity, assuming an average production cross-section of 22 mbarns (Ref. 28). The problem is to interpret this "measured" quantity in terms of dose equivalent, which depends upon the assumed spectrum of neutrons. For 20 MeV neutrons, the ICRP recommended conversion factor (Ref. 7) states that 4 neutrons $\text{cm}^{-2}\text{sec}^{-1}$ is equivalent to 1 mrem/hr (the number is 3.9 at 100 MeV). Various high-energy laboratories have recommended different conversion factors, which are summarized in the following table, reproduced from Ref. 28.:

Laboratory	Flux per mrem/hour
Brookhaven	4
Daresbury	4
CERN	10
Rutherford	3.6
Berkeley	$12.1/E(\text{MeV})^{1/4}$

This problem of exactly how to weight a measurement is inherent in all cases where activation is used.

Another high-energy detection technique depends upon fission. Ionization

chambers lined with bismuth are sometimes used for neutron measurements at very high energies (Ref. 1). Figure 11 (from Ref. 29) shows fission cross sections for U, Bi, Au, and Ta as a function of energy. Another technique is to coat thin evaporated layers of these materials on mica, which is subsequently scanned for fission events. To determine the energy spectrum, ratios of fission-probabilities are taken among the various materials (for example, U/Au, Bi/Au, U/Ta), and the different cross-sections of Fig. 11 are then used to determine the neutron energy spectrum.

Both foil activation and scintillation counting are commonly used for relativistic neutron detection; these have already been discussed under "fast neutrons".

E. Dose Equivalent Instruments

In this section, instruments will be discussed which are intended to provide a direct measurement of dose-equivalent (rem) over a wide range of neutron energies. Referring to Fig. 3, the neutron flux (particles $\text{cm}^2\text{sec}^{-1}$) is shown as a function of energy which provides proper weighting for dose-equivalent.

There are several different approaches to the direct measurement of rem. The most widely used method is the use of a suitably designed moderator surrounding a thermal-neutron detector. There are two geometries in common use: cylindrical and spherical. The cylindrical design was first described by Anderson and Braun (Ref. 30), who calibrated it using both thermal neutrons and 25 keV to 15 MeV neutrons. A similar instrument was evaluated by Block and Petrock (Ref. 31) who showed that in the intermediate-energy range the response was rem-proportional to within $\pm 10\%$ and directionally isotropic to $\pm 15\%$. Another design, by Leake and Smith (Ref. 32), was an improvement over the original Andersson-Braun instrument. The Block design was in turn modified by Tracerlab (now Trapelo West, a division of Lab for Electronics). Trapelo West markets the commercially available "SNOOPY" (see the Instrument Note).

FIGURE 11: Fission cross sections vs. energy of incident nucleon (neutron or proton) for targets of U, Bi, Au, and Ta.

The generic instrument design is as follows: a BF_3 proportional counter is surrounded by two cylindrical layers of polyethylene separated by a 200 mg/cm^2 layer of boron plastic. Similar disks cap the ends of the cylinder. The Andersson-Braun approach was to vary the two polyethylene thicknesses, and to drill a variable number of holes through the boron plastic layer, until the best empirical fit to the desired rem-response was achieved. Using a BF_3 tube, 2.5 cm in diameter and filled to a pressure of 600 torr, the counter was found (Ref. 31) to be insensitive to ^{137}Cs photons (660 keV) up to above 500 R/hour, with neutron sensitivities down to 2 mrem/hr full scale. Also, the energy response was within $\pm 10\%$ of rem-equivalent all the way from thermal to 15 MeV, except for a slight region of oversensitivity near 10 keV. This is shown in Fig. 12, taken from Ref. 30.

FIGURE 12: Neutron detection efficiency as a function of neutron energy. The figures give the counting rate relative to that from a counter with an ideal rem dose response of 19 cps/2.5 mrem h.

The cylindrical instrument is now a widely used survey instrument for mixed-field neutron dosimetry measurements (Ref.33). It is rugged, portable (about 11 kg), and sensitive. It has one important disadvantage: -the electronics associated with the BF₃ tube cannot follow extremely brief, high-intensity radiation bursts such as those around poor-duty-factor linear accelerators. A similar instrument is manufactured in Britain by Twentieth Century Electronics, Ltd, of Surrey. Another Andersson-Braun type detector has been reported by Leake (Ref.34). This instrument, specifically designed for use around pulsed-radiation sources, replaces the BF₃ counter with a BF₃-filled ion chamber run in current mode. The output is measured using a d.c. amplifier with a 30-second integrating time constant. However, the greatest sensitivity is about an order of magnitude less sensitive than that of the BF₃-counter instrument.

The use of a number of different detectors in spherical geometry was first described by Hankins (Ref.35). The basic instrument is commonly called the "Bonner sphere" (Ref.36): a ⁶LiI(Eu) crystal, 4mm in diameter and 4mm thick surrounded by polyethylene moderating spheres. "About 80% of incident thermal neutrons are absorbed in 1mm of ⁶LiI, so that thermal neutron detection is essentially a surface effect, whereas the efficiency for detection of gamma radiation and fast neutrons is roughly proportional to the volume of the crystal" (Ref.36). In Hankins' description

(Ref. 35) the detector is surrounded by a succession of polyethylene spheres, 2,3,5, 8, and 12 inches in diameter. Figure 13 from Ref. 35 shows the relative count rate vs. energy for the various sphere sizes, while Fig. 14 shows the estimated response of a single 10-inch sphere. While the gamma discrimination is slightly less impressive than for the cylindrical instrument, it is still very good: 5 R/hour of photons produces negligible response. That is quite sufficient for most purposes, since one seldom needs to measure 1 mrem/hr of neutrons in the presence of photon exposure rates as large as 5 R/hr.

FIGURE 13: Relative counting rate plotted against energy for moderating spheres.

FIGURE 14: Estimated response of a 10-inch sphere plotted against energy. The points shown are experimental points.

The directional sensitivity of the spheres is, of course, essentially isotropic except on the side where the electronic circuitry is located. From Fig. 14 it can be seen that if one can tolerate errors of about factors of two or three in the fast energy range, a single 10-inch sphere will suffice; the multi-sphere technique must be used when greater precision is required. These spheres are now commercially available, from both Ludlum and Nuclear Chicago (see the Instrument Notes section). Because of the use of LiI with resolution times of about 1 microsecond, these devices suffer from an inability to measure properly in poor-duty-factor situations, a failing they share with the cylindrical (BF₃ counter) instrument. Eberline manufactures a 9" sphere with a BF₃ counter as the detector.

Leake (Ref. 34) has reported a spherical-geometry detector with a moderator similar to the Andersson-Braun design, except that the boron plastic is replaced by cadmium between the polyethylene layers. The detector is LiI(Eu), similar to that of Hankins (Ref. 35) and Bramblett *et al.* (Ref. 36). Commercial manufacture has been undertaken by Isotope Developments, Ltd., of Reading, U.K. (Ref. 34).

Nachtigall and Rohloff (Ref. 37) have succeeded in extending the multi-sphere technique up to an energy of 50 MeV, using four spheres of diameter 2, 5, 11, and 18 inches.

Dose-equivalent instruments which employ several other principles of operation have been reported in the literature. One of these is the use of gas proportional counters sensitive to proton recoils. These instruments have been discussed above under "fast neutrons." They follow the desired response reasonably well up to about 15 MeV, but are insensitive to neutrons below about 100 keV. Three different efforts to develop so-called "Universal Dose Equivalent Instruments" (not specific to neutrons but rather intended for all types of radiation) have been discussed in the section Particle Accelerators. These are the approaches of LET spectrometry, columnar recombination in an ion chamber, and use of a liquid-filled ion chamber.

4. SOURCES FOR CALIBRATION OF NEUTRON INSTRUMENTS

Some of the general problems with calibration of instruments have been discussed in the introductory part of this volume. Here we shall focus attention on a few of the aspects specific to the use of neutron sources for calibration purposes.

The central problem in any calibration measurement is to understand the source of neutrons: its energy spectrum, directionality, presence or absence of equilibrium secondaries, and possible secondary sources from scatter of the primary neutron flux.

One common calibration technique in the fast-neutron region is the use of radioactive neutron sources from (α ,n) reactions. Considerable research has been done on a determination of energy spectra from these sources, and a review paper by Hanson (Ref. 40) contains an excellent summary of information up to 1960. More recent work is summarized in ICRU Report No. 13 (Ref. 5). The most common sources are Po-Be, Ra-Be, Pu-Be, and Am-Be. Figure 15, from ICRU 13, shows the energy spectrum of neutrons from ²⁴¹Am-Be as measured by several groups. The purpose of the figure is to show how much difference exists among various investigators for Am-Be; this situation is also true for each of the other sources mentioned. The spectral uncertainties imply that absolute accuracies better than about $\pm 10\%$ are probably not achievable with this method; but this is probably sufficient for most radiation protection work. One remaining problem is that the spectral studies have not generally extended below about 1 MeV, despite the fact that 20 to 30% of the neutrons in a typical source are in that low energy region (Ref. 5). The Am-Be source has one important advantage over the others, in that there is insignificant contamination from decay of daughters subsequent to Americium decay.

Fission is another common source of fast neutrons. Most data on energy spectra seem to fit the Maxwellian distribution function (Ref. 5), where E is the fission neutron energy and E_n is an experimental parameter:

$$dN(E)/dE \propto E^{1/2} \exp(-E/E_n)$$

Experimental numbers for E_n are in the range 1.2 to 1.6 MeV for several fission nuclei, such as ²³³U, ²³⁵U, ²³⁹Pu, ²⁴⁰Pu, ²⁴¹Pu, and ²⁵²Cf. A compilation of studies on fission spectra was done by Barnard *et al.* (Ref. 41), and summarized in ICRU No. 13 (Ref. 5).

Use of accelerators for production of nearly mono-energetic neutrons has been common; a wide variety of reactions is available leading to a variety of neutron energies. Figure 16 and Table 3, from ICRU No. 13 (Ref. 5), show the properties of some of the reactions used.

FIGURE 15: Energy distribution of ^{241}Am -Be neutrons according to different authors. The curves have been normalized to give the same total number of neutrons between energies 2.5 and 10.5 MeV: — Geiger and Hargrove (1964), proton-recoil counter telescope and time of flight; --- Greiss (1968), nuclear emulsions; Thompson and Taylor (1965), stilbene crystal with pulse-shape discrimination; - - - - Salgir and Walker (1967), proton-recoil spectrometer (silicon detector). The dotted low energy peak has been drawn according to the suggestion of Geiger and Hargrove.

Fig. 16 . Neutron energy spectra resulting from the bombardment of carbon, magnesium, sulphur and silicon with protons of several energies. (Borchers et al., 1964. Publisher: North Holland Publishing Company, Amsterdam.)

TABLE 3 — Monoenergetic neutron producing reactions (all energies in MeV)

Reaction	Q-Value	Threshold Energy	Minimum Neutron Energy at 0°	Maximum Neutron Energy at Bombarding Energy		Maximum Bombarding Energy for Monoenergetic Neutrons
				2 MeV	3 MeV	
$^2\text{D}(\text{d}, \text{n})^3\text{He}$	3.265	—	1.8 ^a	5.24	6.26	—
$^3\text{T}(\text{p}, \text{n})^3\text{He}$	-0.764	1.019	0.0639	1.20	2.21	—
$^3\text{T}(\text{d}, \text{n})^4\text{He}$	17.6	—	12.4 ^a	18.25	19.57	3.7
$^7\text{Li}(\text{p}, \text{n})^7\text{Be}$	-1.647	1.882	0.0294	0.228	1.3	2.38
$^{13}\text{C}(\text{d}, \text{n})^{14}\text{N}$	-0.281	0.328	0.00195	1.6	2.6	3.0
$^{48}\text{Sc}(\text{p}, \text{n})^{48}\text{Ti}$	-2.79	2.908	0.00140	—	0.112	3.67
$^{51}\text{V}(\text{p}, \text{n})^{51}\text{Cr}$	-1.562	1.565	0.00059	0.215 ^b	—	2.36
$^{68}\text{Cu}(\text{p}, \text{n})^{68}\text{Zn}$	-2.136	2.169	0.00051	0.214 ^c	—	2.29

^a Measured at 2.0 MeV, 150 degrees.

^b At bombarding energy = 1.762 MeV.

^c At bombarding energy = 2.368 MeV.

Sources of thermal neutrons are usually made by the moderation of any of the fast neutron sources just mentioned (Ref. 1). Alternatively, a thermal column from a reactor can be used. The fluence rate of any of these thermal sources can be calibrated by the foils-activation technique; gold foils are commonly used when high accuracies are required.

The directionality of the neutron source is sometimes difficult to determine, especially for thermal neutrons. The important problem here is the possible presence of scattered particles in addition to the primary flux. For accelerator-generated neutron beams the beam direction is known well, but beam halo is difficult to understand or reduce. For isotopic sources of small size, one can use a small detector to study departures from inverse-square fall-off of the flux. To minimize scattering problems, the physical layout of the calibration set-up should be carefully planned: for example, apparatus placed near a wall can be subject to large scattering corrections. A discussion of this, and of methods for measuring the possible scattering correction, can be found in ICRU Report No. 20 (Ref. 1).

In all calibrations, cross-comparison with standard instruments is extremely important. This is necessary to complement the understanding of the properties of the neutron source, because of possible unknown changes in the source.

As of this time, there does not exist a standard mechanism of calibrations for neutron instruments through any of the national standards bodies. Further, there are large regions in the neutron energy spectrum for which no well-understood neutron sources exist (Ref. 5): this is true in the region around 1 eV, and also near 2 keV. The available data are poor for many of the important cross-sections in the fast neutron region (e.g.,

ICRU Report No. 13 contains a summary of basic research (primarily cross-section measurements) needed to make possible accurate neutron instrument calibrations.

5. SUMMARY AND RECOMMENDATIONS

Neutron monitoring comprises two distinct tasks: (a) making detailed area surveys to determine the properties of the radiation fields; and (b) conducting routine surveillance monitoring. The latter should be supplemented with detailed surveys, undertaken occasionally and whenever there have been important changes in operating conditions.

The purpose of this section has been to enumerate the various classes of instruments, the conditions under which each class is appropriate and the changes in operation which require a detailed re-evaluation. The reader is directed to the original references for relevant details about specific techniques.

It is important to remember that whenever a new installation begins operation, it is essential to perform a detailed survey. Thus, separate instruments dominantly sensitive to thermal, intermediate, fast and possibly to relativistic neutrons are needed, as well as an instrument sensitive to photons. The relative weighting of neutrons in the various energy ranges can be assessed using a combination of techniques discussed in the text. There should by all means be an overlap in the means of measuring the various neutron energies to provide some redundancy of information. Also, these spectral studies are an important aid in studies for reducing levels for radiation protection purposes.

For routine surveillance an instrument whose response is proportional to dose-equivalent (rem) is recommended. Among those commercially available the cylindrical Anderson-Braun type is widely used, specifically that manufactured by Trapedo West. Also available are the Bonner-sphere instruments manufactured by Ludlum, Nuclear Chicago, and Eberline.

An instrument sensitive to thermal neutrons is frequently required: BF_3 and ^3He gas proportional counters, boron-lined chambers, $\text{LiI}(\text{Eu})$ scintillation detectors, as well as several other types are discussed. Activation of foils is another good technique for monitoring thermal neutrons because it is both simple and sensitive. A variety of foils applicable to a wide range of neutron energies is available commercially. All of these techniques are represented among the individual Instrument Notes.

The Indium/Cadmium technique, together with any of the thermal neutron detectors, can be used to provide a rapid assessment of fluxes of thermal- and intermediate-energy neutrons.

In the text no attempt has been made to comment on the commercially available instruments individually: the advantages and disadvantages are discussed in the classes of detectors (e.g., BF_3 tubes). However, sufficient information is not available at this time to make valid comparisons concerning the regions of applicability of many instruments. Also, we have attempted to give a brief discussion of some instruments and techniques which are not now commercially available, but are worthy of commercial development.

Finally a comment about electronics design should be made. It is generally true that monitoring instruments have not yet incorporated a number of the most recent advances in electronic technology. When exploited, these advances will lead to many improvements: Increased stability and reliability, reduced maintenance, smaller size,

weight and cost, and better means of data read-out. It is urgently recommended that these improvements be incorporated commercially wherever possible.

6. ACKNOWLEDGMENT

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F. INSTRUMENT NOTES: NEUTRONS

In this section are found Instrument Notes for those commercial instruments which are capable of measuring neutrons. The filing system lists Instrument Notes alphabetically by manufacturer. These notes include instruments that are appropriate for neutron spectroscopy as well as dose-equivalent measurements, neutron flux measurements, etc.

Neutron Dose-Equivalent Monitor
Eberline Radiation Monitor RM-16
Neutron Rem Detector, NRD

Class	Stationary
Principle of Operation	Polyethylene (cadmium-loaded) sphere of 9" diameter, BF ₃ tube inside, used with rate meter.
Sensitivity	NRD: (1) Energy response close to rem-equivalent from thermal to about 10 MeV. (2) Range 1 to 10,000 mrem/hour. RM-16: (1) Input sensitivity is charge sensitive, adjustable from approximately 2×10^{-13} to 10^{-11} Coulomb. (2) Range of four logarithmic decades. (3) Window width is adjustable from 0 to at least double the input sensitivity.
Sampling	Continuous
Performance	NRD: (1) Gamma rejection up to 500 R/hour dependent on high voltage. (2) About 50 counts/minute corresponds to 1 mrem/hour. (3) Plateau approximately 300 volts with a slope of 3% per 100 volts. (4) Directional response within $\pm 10\%$ (5) Operation voltage in typically 1600 to 2000 volts. RM-16: (1) Response time varies continuously with count rates. (2) Linearity within $\pm 10\%$ of reading typically, $\pm 25\%$ maximum. (3) High voltage regulated and adjustable from less than +200 volts to +2500 volts. (4) Temperature range is -1°C to 60°C ($+30^{\circ}\text{F}$ to $+140^{\circ}\text{F}$)

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RAD-NEU
Stationary Monitor
Eberline
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	<u>NRD-1</u>	<u>RM-16</u>
Requirements	Power: 105 - 125V, 60 Hz Size: 22.9cmD x 25.1cm overall height (9"x9 7/8") Weight: 6.25 kg (13 3/4 lb)	105 - 125V, 50-60 Hz and rechargeable batteries 13.6cmHx 26.4cmD x21.6cmW (5 3/8" x10 3/8"x8 1/2") 3.69 kg (8 lbs. 2 oz.)
Features	(1) Detector can be wall-mounted. Specify wall mounting bracket. (2) Cable up to 100 feet long from detector to electronics package (3) Alarm indication, high, red light and more penetrating tone; low, blue lamp	
References	Manufacturer's Specifications	
Cost	NRD Sphere w/cable \$ 760.00 RM-16 Rate Meter 1320.00 w/meter face 10472-B42	
Address	Eberline Instrument Corporation P.O. Box 2108 Santa Fe, NM 87501 (505) 471-3232

Neutron Monitoring Instrumentation

PRS-2P/NRD

Class	Portable
Principle of Operation	BF ₃ tube detector in spherical moderator measures neutron dose equivalent
Sensitivity and Range	0.2 mrem/hr to 10 rem/hr Typical rem response factors based on NBS Calibration: Energy (keV) 2 24 144 235 515 754 1054 Cpm per mrem/hr 215 240 80 70 35 35 35
Sampling	Continuous
Performance	May be used to measure approximate dose equivalent. Rejects gamma up to 500R/h depending on high voltage setting. Directional response within ±10%. Operational temperature from -18°C to 60°C (0 to 140°F).
Requirements	Power: Rechargeable Gel-Cell, 75 hrs./charge Size: 10cmWx24cmLx20cmH (4x9.5x7.75 inches) Weight: 2.3 kg (5 pounds)
Features	Operates as count rate meter with pre-selected level of precision or as scaler. Multiplier factor in count rate mode converts reading to mrem/hr, digital display.
References	Eberline General Catalog

INSTRUMENTATION
FOR ENVIRONMENTAL
MONITORING

RAD-NEU
Dose Equivalent
Eberline 2
Page 2

Cost	\$2160.00
Address	Eberline Instrument Corporation P.O. Box 2108 Santa Fe, New Mexico (505) 471-3232

Neutron Dosimeter
Harshaw DN105 PIN Diode

Class	Laboratory
Principle of Operation	The dosimeter operation is based on the fact that neutron damage in silicon tends to inhibit current flow. The diode is designed to be operated on the forward bias direction so that a current of 25 ma flows. A precision voltmeter is used to measure the voltage, V_f , required to establish the current. After irradiation the value of the voltage is found to increase by ΔV_f . Using calibration data similar to Fig. 1, the neutron dose in tissue may be determined.
Sensitivity and Range	Mean value of sensitivity, V_f/D^* 1.2 ± 0.1 mV/rad for AmBe neutrons
Sampling	Continuous
Performance	Mean value of V_f^* 1.20 ± 0.10 volts Randomness of V_f in group standard deviation 0.10 volts
Requirements	Size: $0.030'' \pm 0.002''$ $0.188'' \pm 0.020''$

While many instrument combinations may give good results, Harshaw recommends the following NIM module:

NC-28 Neutron Diode Analyzer
NS-12 Scaler
NQ-75 6-wide miniature NIM bin
737-216 Interconnect cable

Features	
References	Manufacturer's Specifications
Cost	Refer to factory
Address	The Harshaw Chemical Company 6801 Cochran Road Solon, OH 44139 (216) 248-7400

*Specifications at 25 ma

Neutron Radiation Detectors, $\text{BF}_3\text{-He}^3$

Class Portable, Laboratory

Principle of Neutron Detectors
Operation

Reference Manufacturer's Specifications

Address LND
 3230 Lawson Blvd.
 Oceanside, N.Y. 11572
 (516) 678-6141; Telex 14-4563

Quadrilateral BF_3 Counting Tubes

Model	Outside (cm)	Dia. (inches)	Overall (cm)	Length (inches)	Operating voltage	General	Price
220	1.9x1.2	0.75x0.75	26.0	10.25	1100	Spectrometry	\$525.00
221	2.9x2.5	1x1	26.0	10.25	1200	Spectrometry	525.00
223	3.8x3.8	1.5x1.5	36.2	14.25	1500	Spectrometry	720.00
224	5.1x5.1	2x2	36.2	14.25	1500	Spectrometry	655.00

Cylindrical ^3He Counting Tubes

Model	Outside (cm)	Dia. (inches)	Overall (cm)	Length (inches)	Operating voltage	General	Price
250	2.54	1.0	132.72	52.25	1300	Flux Mapping	\$480.00
2501	0.76	0.300	13.02	5.125	1900	Time of Flight	385.00
251	1.35	0.531	15.22	6.032	1000	Monitoring	335.00
2510	1.31	0.515	32.22	12.687	2200	Time of Flight	420.00
2511	1.31	0.515	21.75	8.562	2100	Time of Flight Spectroscopy	420.00
252	1.31	0.515	36.75	14.562	2200	Time of Flight Spectroscopy	450.00
2520	2.62	1.031	26.04	10.25	1275	Monitoring	485.00
2521	2.62	1.031	24.45	9.625	1275	High Temperature	485.00
2522	2.62	1.031	27.14	14.875	3400	Time of Flight Spectroscopy	750.00
2523	2.62	1.031	42.38	16.687	3400	Time of Flight Spectroscopy	690.00
2524	2.62	1.031	57.62	22.687	3400	Time of Flight Spectroscopy	835.00
2525	2.62	1.031	29.21	11.50	1300	Industrial Gaging	1075.00
2526	2.62	1.031	34.29	13.50	1300	Industrial Gaging	1055.00
2527	2.62	1.031	39.37	15.50	1300	Industrial Gaging	1270.00
253	5.16	2.031	26.04	10.25	1500	Monitoring	1480.00
2531	5.16	2.031	29.53	11.625	4800	Spectrometry	945.00
254	5.16	2.031	17.46	6.875	4800	Diffraction	1105.00
255	4.45	1.75	21.11	8.312	1500	High Temperature	1655.00

Quadrilateral ^3He Counting Tubes

Model	Outside (cm)	Dia. (inches)	Overall (cm)	Length (inches)	Operating voltage	General	Price
262	2.54x2.54	1x1	26.04	10.15	1500	Spectrometry	\$690.00
263	5.1 x5.1	2x2	36.8	14.50	2000	Spectrometry	1750.00
275	1.9 x5.1	0.75x2	11.1	4.375	1700	Beam Monitor	895.00
27510	1.9 x1.97			4.75	1900-2140	Beam Monitor	765.00

Cylindrical BF₃ Counting Tubes

Model	Outside (cm)	Dia. (inches)	Overall (cm)	Length (inches)	Operating voltage	General	Price
200	2.24	0.875	132.40	52.125	1200	Flux Mapping	\$455.00
201	1.35	0.531	15.40	6.062	1600	Monitoring	170.00
2011	1.35	0.531	19.70	7.75	1200	Monitoring	170.00
20110	1.32	0.519	19.70	7.75	1050	Survey Instruments	170.00
202	2.62	1.031	30.80	12.125	1100	General Purpose	195.00
2027	2.62	1.031	30.80	12.125	1100	General Purpose	195.00
2021	2.70	1.062	14.61	5.75	1600	AV/PDR-70	245.00
20210	2.62	1.031	35.48	13.97	1700	General Purpose	230.00
2022	2.62	1.031	20.00	7.875	1700	General Purpose	230.00
2023	2.62	1.031	40.32	15.875	1200	General Purpose	230.00
2029	2.62	1.031	40.32	15.875	1200	General Purpose	230.00
2024	2.62	1.031	60.64	23.875	1200	General Purpose	240.00
2025	2.62	1.031	30.48	12	1200	Reactor Source Range	230.00
2026	2.62	1.031	76.52	30.125	1200	Reactor Source Range	250.00
2028	2.62	1.031	37.78	14.875	1600	Diffraction	720.00
2033	5.16	2.031	40.32	15.875	2200	General Purpose	250.00
203	5.16	2.031	40.32	15.875	2200	General Purpose	250.00
2031	5.16	2.031	27.94	11	3000	AN/PDR-58	720.00
20310	5.16	2.031	64.45	25.374	2700	General Purpose	295.00
20327	5.16	2.031	30.56	12.375	1300	General Purpose	245.00
2034	5.16	2.031	30.56	12.041	1300	General Purpose	295.00
212	5.16	2.031	33.63	12.031	2500	Diffraction	295.00
2121	5.16	2.031	39.69	28.25	2500	Diffraction	980.00

¹⁰B Lined Proportional Counters

Model	(cm)	Max. Dia. (inches)	Overall (cm)	Length (inches)	Operating voltage	Sensitivity (cps/nv)	Price
230	1.35	0.531	As Specified		700	0.6	\$585.00
232	2.54	1.0	20.00	7.875	800	1.0	525.00
23210	2.54	1.0	38.74	15.25	950	4.0	585.00
23211	2.54	1.0	76.2	30.0	800	10	780.00
23011	1.27	0.50	29.61	11.656	600-900	0.6	525.00
23110	1.27	0.50	69.06	27.188	600-950	7.5	670.00
23113	1.27	0.50	160.82	63.313	600-950	7.5	835.00

Proton Recoil Fast Neutron Detectors

Model	(cm)	Max. Dia. (inches)	Overall (cm)	Length (inches)	Operating voltage	Price
270	4.10	1.615	11.75	4.625	1500-3200	\$785.00
2801	5.08	2.00	17.15	6.75	1400	655.00
2802	5.08	2.00	20.80	8.19	1400	720.00
2803	5.08	2.00	31.75	12.50	1400	805.00
2804	5.08	2.00	53.72	21.15	1400	850.00
281	3.81	1.50	25.40	10.0	2500	555.00
2821	2.54	1.0	17.15	6.75	1100	555.00
2822	2.54	1.0	20.80	8.19	1100	625.00
2823	2.54	1.0	31.75	12.50	1100	725.00
2824	2.54	1.0	53.75	21.15	1100	805.00
28114	5.1	2.0	70.00	27.75	2500	525.00
28115	5.1	2.0	85.34	33.6	2500	590.00

<u>Fission Counter</u>								
Model	(cm)	Max. Dia. (inches)	Overall (cm)	Length (inches)	Operating voltage	Remark	Range	Price
3075	0.64	0.25	2.54	1.0	200-800	Stainless	10^3 to 10^8 n	\$1135.00
3076	1.27	0.5	13.97	5.5	200-800	Aluminum	10^3 to 10^8 n	1225.00
3078	5.08	2.0	30.48	12	200-800	Aluminum	10^3 to 10^8 n	1485.00

<u>One-dimension Position Sensitive Proportional Counter</u>							
	Max. Dia. (cm)	Dia. (inches)	Overall (cm)	Length (inches)	Operating voltage	Anode	Price
25225	2.54	1.0	112.40	44.25	1000	quartz	\$3055.00
25228	2.54	1.0	112.40	44.25	1000	molecul---	3055.00
25229	2.54	1.0	43.18	17.00	1000	quartz	2075.00
25230	2.54	1.0	43.18	17.00	1000	molecul---	2075.00

Survey Meter
Ludlum Models 2 and 3

Class	Portable
Principle of Operation	Count rate meter for a variety of G-M and scintillator probes.
Sensitivity and Range	Sensitivity: See table on Ludlum Probes note Range: Model 2: 0-5 mR/hr, x0.1, x1, x10, linear Model 3: 0-2 mR/hr, x0.1, x1, x10, x100, linear 0-5,000 cpm, optional meter readout
Sampling	Continuous
Performance	Accuracy: $\pm 5\%$ of full scale Calibration Stability: Less than 5% variance to battery end point
Requirements	Power: Two "D" cells; 300 hr Size: 9 cm x 9 cm x 18 cm (3.4" Hx3.5" Wx7.0" L) exclusive of handle
Features	Pushbutton reset, adjustable high voltage output, speaker output
References	Manufacturer's specifications
Cost	Model 2: \$240.00 Model 3: \$270.00
Address	Ludlum Measurements Inc. 501 Oak Street Sweetwater, TX 79556 (915) 235-5494

Ludlum Probes

Model 42-1
Thermal Scintillator (Neutron)

Model 42-2
Fast Neutron Scintillator

Model 42-4
10" Sphere Neutron
Scintillator

Class Portable

Principle of ZnS(Ag) and $^6\text{LiI(Eu)}$ scintillators, and photomultiplier
Operation tubes, polyethylene sphere moderator

LUDLUM NEUTRON PROBES

Model	Type	Tube	Energy	Efficiency	Size	Weight	Cost
42-1	Scint. 1" diam NE 400 Boron Glass Detect. with ZnS(Ag) activ. n, α react.	RCA 6199	Thermal 0.025 eV	60 cpm = $1 \text{ n cm}^{-2}\text{sec}^{-1}$ γ rejection > 10 R/hr	7" long 2" diam	0.9 lb	\$275.00
42-2	Scint. 2" diam NE 404 Proton Recoil Defect. by ZnS(Ag)	RCA 6199	5 MeV	15 cpm = $1 \text{ n cm}^{-2}\text{sec}^{-1}$ γ rejection > 1 R/hr	7.5" long 2.3" diam	1 lb	275.00

42-4	Scint. 4 mm, 4 mm Li ⁶ I (Eu)	RCA 931A	Thermal to 7 MeV	Thermal 74 cpm = 1 mrem/hr 1 MeV 58 cpm = 1 mrem/hr γ rejection > 1.5 R/hr	10" sphere	18 lb	750.00
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Reference	Manufacturer's specifications
Address	Ludlum Measurements, Inc. 501 Oak Street Sweetwater, Texas 79556 (915) 235-5494

Neutron Spectrometer

Ludlum Model 42-5

Relative counting rate plotted against energy
for moderating spheres.

Class	Mobile														
Principle of Operation	Six different diameter spheres moderate various energy neutrons, LiI(Eu) crystal, light pipe and RCA 6199 photomultiplier tube to be used with any counting system capable of accepting 2 millivolt minimum signals and providing 900 V dc to the detector.														
Sensitivity and Range	<table><tr><th>Sphere Diameter (in.)</th><th>Energy Range</th></tr><tr><td>2</td><td></td></tr><tr><td>3</td><td></td></tr><tr><td>5</td><td></td></tr><tr><td>8</td><td>See Figure</td></tr><tr><td>10</td><td></td></tr><tr><td>12</td><td></td></tr></table>	Sphere Diameter (in.)	Energy Range	2		3		5		8	See Figure	10		12	
Sphere Diameter (in.)	Energy Range														
2															
3															
5															
8	See Figure														
10															
12															
Sampling	Continuous														
Performance	Accuracy, Efficiency, Response Time, etc. (see references)														
Requirements	Power: 900 V dc to be supplied by counting system Size: 112 cm H x 45 cm W x 52 cm D (45" x 18" W x 21" D) Weight: 55 kg (120 lbs)

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RAD-NEU
Spectrometer
Ludlum 3
Page 2

Features X-ray inspection of all spheres insures void-free moderators. A crank operated elevator assembly adjusts detector center from 44 to 73 inch working height.

References a) Manufacturer's specifications
 b) D. E. Hankins, "The Multisphere Neutron-Monitoring Technique," Los Alamos Report LA-3700, UC-41 Health and Safety, TID-4500, Los Alamos, New Mexico 1968.

Cost \$2250.00

Address Ludlum Measurements, Inc.
 501 Oak Street
 Sweetwater, TX 79556
 (915) 235-5494

Radiation Survey Meter

Ludlum Model 15

Class	Portable, handheld
Principle of Operation	Uses moderated BF ₃ detector for neutrons and end window GM detector for beta, gamma, and alpha, associated electronics of count rate meter, speaker, meter readout. The BF ₃ detector may be used in or out of the moderator providing capability for both fast and slow neutrons.
Sensitivity and Range	Radiation detected: neutron, beta, gamma, and alpha Range: 0-500 counts/minute with multiples of x1, x10, x100, x1000
Sampling	Continuous: push button switch for meter reset
Performance	Discrimination range: 2 to 60 millivolts Input impedance: 0.7 megohm High voltage: variable from 700 to 2400 Calibration stability: <15% variance to battery end point Meter accuracy: ±5% of full scale Response time: Fast: 3 seconds Toggle switch Slow: 11 seconds selection Neutron detector Gamma rejection: Function of discrimination point. Nominal 10R/hr. Moderator: One-inch thick paraffin enclosed by a 0.030-inch thick cadmium thermal neutron shield. Outer shell and fittings of aluminum End window GM detector Window: 1.5 to 2 mg/cm ² mica Window diameter: 3.18cm (1.25") Wall: 0.318cm (0.125") stainless steel Dimension: 3.49cm x13.34cmL (1.375" x5.25"L)
Requirements	Power: Two "D" cells, life: 7100 hrs. Size: 25cm x9cm x18cm (9.8" x3.5" x7") Weight: 3.6 kg (8 lbs)
Features	Calibration adjustment for each scale, temperature compensated. High voltage selector switch selects fixed 900 volts for GM detector or variable high voltage for BF ₃ and other detectors, calibration adjustment for each scale, temperature compensated.
Cost	\$775.00
Address	Ludlum Measurement Inc. 501 Oak Street Sweetwater, TX 79556 (915) 235-5494

Neutron Counters

N. Wood Models

Class

Portable, Laboratory

Principle of Operation

BF₃ Neutron proportional counters

MN-1

MN-1

MN-3

G-5-1

G-5-2

G-5-3

G-5-9

G-5-9

G-5-16

G-10-2

G-10-2A

G-10-5

G-10-5

G-10-5

G-10-12

G-10-12A

G-10-20

G-15-5

G-15-7H

G-15-34A

G-20-5

G-20-12

Model	Diameter	Active Length	Operating voltage	General	Price
MN-1	1/4"	1/4" to 5"	1100 to 2000 volts,	Pigtail lead.	Microneutron counters for
MN-3	1/4"	as specified.	for 20 to 60 cm. Hg. pressure of BF ₃ .	931/U connector (MN-3 has 5/16" casing up to 56" long.)	reactor instrumentation.
					\$105.00
					200.00
G-5-3	5/8"	3"	1400 to 2100 volts,	931/U connector.	Small general
G-5-9	or	9"	for 20 to 60 cm. Hg.	Caps same O.D.	purpose neutron
G-5-12	1/2" as	12"		as counter.	detectors.
G-5-16	specified.	16"			
					113.00
					113.00
					120.00
					126.00
G-10-2	1"	2"	1300 v. for 20 cm. Hg.	Pigtail lead.	Widely used in
G-10-2A	1"	2"	1800 v. for 60 cm. Hg.	931/U connector.	portable instruments.
G-10-5	1"	4 1/2"	1500 to 2200 v. for 20	HN 82-805	Standard multiple
G-10-12	1"	12 1/2"	to 60 cm. Hg. For	or	purpose neutron
G-10-20	1"	20"	lower oper. volt. specify small dia. anode.	931/U connector.	detectors.
					113.00
					126.00
					140.00
					155.00
					170.00
G-15-5	1 1/2"	5"	1700 to 2250 v. for 20	HN 82-805	Standard research
G-15-12	1 1/2"	12"	to 45 cm. Hg. Higher	or	counters. High sensitivity
G-15-20	1 1/2"	20"	pressure may be	931/U connector.	low background.
G-15-34A	1 1/2"	34"	specified as well as lower voltage.	[G-15-34A was developed for cosmic radiation studies (Simpson counter).]	
					140.00
					177.00
					197.00
					245.00
G-20-5	2"	5"	1900 to 2300 v. for 20	General purpose high sensitivity detectors.	
G-20-12	2"	12"	to 45 cm. Hg. Higher	Useful in moisture meters and environmental	
G-20-20	2"	20"	pressures and lower	monitoring. Connectors same as above	
G-20-38	2"	38"	voltages may be specified.	models.	
					162.00
					197.00
					225.00
					315.00

Prices listed are for brass or copper cathode, prices for aluminium cathode - add \$10.00; prices for stainless steel on request.

Prices listed for Model G (Enriched BF₃ - 97% Boron 10) are the same for Model F (Depleted BF₃ - 11% Boron 10).

Address N. Wood Counter Laboratory, Inc.
1525 E. 53rd Street
Chicago, IL 60615
(312) 324-1114

Neutron Monitor
Nuclear Enterprise Model NM-1

Class	Portable
Principle of Operation	Combined moderator/attenuation assembly, BF ₃ proportional counter, amplifier, meter readout visual.
Sensitivity and Range	Energy sensitivity: 0.025 eV to 10 MeV Range: 0.1 mrem/hr to 10 rem/hr, 5 decade log scale Gamma sensitivity: No significant response to fields up to 200 R/hr
Sampling	Continuous
Performance	Detector Output: normally 5 cps per 2.5 rem/h Response. Correct rem-dose response is maintained within 20% over energy range Temperature range: -10°C to 40°C Response time: ~60 sec at 0.5 mRem/hr reducing at higher rates. Scaler output. 2V negative, 2 μ sec wide pulse
Requirements	Power. Two Ever Ready Type PP9 and two Ever Ready Type PP7, ~ 80 hour life Size. Diameter (excluding handle and legs) 21.6cm (8.5") Length (excluding handle) 43.2cm (17") Weight 14.1 kg (31 lb)
Features	Remote scaler and recorder outputs provided
Cost	Available on request
Address	Nuclear Enterprises 931 Terminal Way San Carlos, CA 94070 (415) 592-8663

Neutron Dose Equivalent Monitor

Nuclear Enterprises Model MK 7

Class	Portable
Principle of Operation	The neutron detector is a polyethylene sphere with an inter-mediated perforated cadmium layer, at the center of which is a spherical ^3He proportional counter attached to a ratemeter with a visual log meter display
Sensitivity and Range	Energy range: thermal (.025 eV) to 7 MeV Dose Equivalent rate range: 0.1mrem/h to 1000 mrem/h
Sampling	Continuous
Performance	Correct rem response is maintained within a factor of 2%, in the thermal neutron energy range. At 14 MeV the response is down by a factor of 4 at higher energies the ratio falls but is typically 1000:1 at 10r/hr. Gamma rejection ratio; i.e., counts per rem/hr: counts per Roentgen/hr: better than 3000: 1(1r/hr, 100 keV). At higher energies the ratio falls but is typically 1000:1 at 10r/hr Output pulse: -5 to -15V pulse 3.2 cps/mrem/h
Requirements	Power: Batteries, 2 R20 (British type 02) Size: 38.0cmLx21cmWx25cmH Weight: 7.3 kg

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Features	Accessories a) 6 meter extension cable between detector assembly and indicator unit b) transport and storage case Output for remote scaler
References	Manufacturer's literature
Cost	Available on request
Address	Nuclear Enterprises, Inc. 931 Terminal Way San Carlos, CA 14070 (415) 592-8663 TELEX: 348-371

Neutron Monitor

NRC RemRad Model NP-2 "Snoopy"

Class	Portable, handheld
Principle of Operation	A moderator/attenuator assembly, BF ₃ detector, associated electronics used to convert detector pulse output to meter read-out in mRem/hour. Scaler output for measuring total dose.
Sensitivity and Range	Sensitivity: 6000 counts/mRem Energy Range: Thermal (0.025 eV) to 15 MeV Ranges: 0-2, 20, 200, 2000 mRem/hour linear Gamma sensitivity: No response to gamma radiation in fields up to 500 R/hr of ¹³⁷ Cs
Sampling	Continuous
Performance	Accuracy: ±10% of theoretical ICRP dose rate Temperature: -10°C to +50°C Detector resolving time: 1 μsecond Directionality: +20% Detector operating voltage: 1750 - 195 V dc Background: Less than 5 counts per minute
Requirements	Power: (3) "D" cells Size: 40cmL x24cmDia (16"L x9.6"D) Weight: 11.5 kg (25 lbs)
References	1. J. O. Anderson and J. Braun, "A Neutron Rem Counter", AE-132 2. Evaluation of a Neutron Rem Dosimeter, Hazards Control Quarterly Report #18, UCLRL 12/67. 3. Manufacturer's Specifications
Cost	\$1895.00
Address	Nuclear Research Corporation 1105 Industrial Highway Southampton, PA 18966 (215) 357-5015

Fast Neutron Survey Meter

Model E-1 D

Class	Portable, handheld, fast neutrons
Principle of Operation	Hurst chamber
Sensitivity and Range	Energy range: 0.2 to 14 MeV Integrate dose range: 0.05 millirep Ranges: 5, 50, 500 millirep/hr NOTE: 1 cps $\approx$ 1 milliroentgen equivalent physical (millirep) per hour
Sampling	Continuous
Performance	Accuracy: $\pm 10\%$ for Pu-Be neutrons
Requirements	Power: 6 "D" cells, 60 hours continuous Size: 25cm x 12cm x 15cm (10" x 4 3/4" x 5 5/8") Probe: 25cm x 5cm (Probe: 9 3/4" x 2"D) Weight: 3.6 kg (8 lbs)
Remarks	One count per second corresponds approximately to one milliroentgen equivalent physical (millirep) per hour.
Cost	On request
Address	Nuclear Research Corporation 1105 Industrial Highway Southampton, PA 18966 (215) 357-5015

Neutron Spectrometers

Ortec Spectrometers

Class	Laboratory			
Principle of Operation	Two closely spaced detectors with a thin layer of neutron sensitive material (⁶ LiF or ³ He) between, in a sandwich geometry. Neutrons incident on the layer give rise to charge particle pairs. The two particles are detected in the adjacent surface barrier detectors, and the Q-value of that reaction provides a bias against background events.			
Sensitivity		525-782	525-780	
	Neutron-Sensitive Material	150 μg/cm ² of ⁶ LiF on foreground detector	³ He gas (5 atm pressure)	
	Energy Range:	3 MeV and above	1-6 MeV	
	Resolution:	400 keV guaranteed 300 keV typical	100 keV guaranteed 84 keV typical	Thermal neutrons at 5 atm pressure
	Efficiency:	~10 ⁻³ ~7 x 10 ⁻⁷	~10 ⁻² ~3 x 10 ⁻⁶	for thermal neutron for 2.5 MeV neutron
	Performance	Temperature: Electronic Noise: Typically ≥ 60 keV FWHM total system noise Coincidence Resolving Time: 10 ns min, 30 to 40 ns typical Drift: about 5 keV/day at ambient temperatures		
Requirements	Power: Size: Two NIM bins (48.4cmW x44.5cmH x43.2cmD) -780 Detector (3.5cmDia x9.8cmL) -782 Detector (3.5cmDia x2.4cmL) Weight: 525-780 -- 50.5 kg (111 lb) 525-782 -- 44.8 kg (98.5 lb)			
Features	Components: Detector head, detector bias supply. Preamplifier; linear amplifier; precision pulse generator; timing single channel analyzers; sum-invert amplifier; fast coincidence; bias amplifier, pulse stretcher, and linear gate; delay amplifier, bin and power supplies, cables.			
References	Manufacturer's Specifications			
Cost	Model 525-780 ³ He detector head ----- \$7730.00 Model 525-782 ⁶ Li detector head ----- \$7380.00			
Address	Oak Ridge Technical Enterprises Corporation 100 Midland Road Oak Ridge, TN 37830 (615) 482-4411

³He Fast Neutron Spectrometer
Seforad-Applied Radiation Model FNS-1

Class	Detector
Principle of Operation	The detector is an ion chamber filled with ³ He gas at a high pressure together with a dense gas mixture to stop the product particle from the reaction: ${}^3\text{He} + n \rightarrow {}^1\text{H} + {}^3\text{H} + 764 \text{ keV}$. The electronics required is identical to that required by Ge(Li) spectrometry.
Sensitivity and Range	Energy range: 0.1 to 5.0 MeV Absolute efficiency at 1 MeV: 3×10^{-4} (point source at the side)
Sampling	Continuous
Performance	Energy resolution for thermal neutrons: better than 12 keV FWHM Energy Resolution for 1 MeV neutron: better than 25 keV FWHM Output: Polarity, negative Amplitude, 50 mV/MeV Impedance, 50 Ω Rise time, 0.2 to 5 microseconds (10 to 90%) Shaping regimes, 4 μsec Gaussian Fall time, about 1 msec.

Requirements Power: preamplifier, $\pm 24\text{V}$ DC from NIM Standard Amplifier
 anode bias, $+3000\text{V}$
 grid bias, $+850\text{V}$
 Size: ion chamber, $5\text{cmD} \times 50\text{cmL}$ (15cm active)
 preamplifier, $11\text{cmL} \times 8\text{cmW} \times 4\text{cmH}$
 carrying case, $67\text{cmL} \times 15\text{cmW} \times 15\text{cmH}$
 thermal neutron shield, $6\text{cmDia} \times 50\text{cmL}$
 spectrometer stand (collapsible), $45\text{cmL} \times 15\text{cmW} \times 20\text{cmH}$
 Weight: 7 kg

Features

References Manufacturer's Specifications

Cost \$10,000.00, includes ion chamber, preamplifier, thermal
 neutron shield, carrying case, pulse height response
 spectrum to thermal neutrons.

Address Peabody Scientific
 P.O. Box 2009
 Peabody, MA 01960
 (617) 535- 0444

Marketers for Seforad-Applied Radiation Ltd, Israel

Criticality Dosimeter (CD)
Reactor Experiments Cat. No. 710

Class	Portable
Principle of Operation	Utilization of neutron activation foils; after criticality accident, incident spectra may be broken down.
Sensitivity and Range	Neutron energy intervals (foils are sensitive only to neutron) which may be obtained Thermal -0.3 eV 1 MeV to 2.9 MeV 0.3 eV to 2 eV above 2.9 MeV 2 eV to 1 MeV (TLD for gamma 1 MeV doses greater than .5R included in wall mounted version)
Sampling	Continuous
Performance	Foils unaffected by temperature. Readout must be done on a GM counter or multi-channel analyzer. Foils include indium gold, sulfur, cadmium-covered indium, cadmium-covered gold, and cadmium-covered copper, and a TLD using ⁷ LiF in the wall-mounted version
Requirements	Power: None required Size: 2 1/4" x 2" x 1/8" Weight 1/2 oz.
Features	Inexpensive, sensitive dosimeter which is meant to be worn or used as an area monitor. The foils are sealed in a tough plastic package with a strong spring clip.
References	Manufacturer's Specifications

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Cost Available in boxes of 10
 Pocket version: 1-4 boxes: \$150.00
 5-9 boxes: 120.00
 10-24 boxes: 97.50 each
 Wall mounted version: 1-4 boxes: \$200.00
 5-9 boxes: 160.00
 10-24 boxes: 130.00 (Prices are for 10)

Address Reactor Experiments, Inc.,
 963 Terminal Way
 San Carlos, CA 94070
 Dean C. White, Technical Sales
 (415) 592-3355

Neutron Flux Integrator (NFI)
Reactor Experiments Cat. No. 700

Class Laboratory use; may be carried to site

Principle of Operation Utilization of neutron activation foils; then counted using an appropriate detector

Sensitivity and Range Sensitive to 2 keV to 20 MeV neutrons. Gives integrated dose.

Foil	Co	In	Au
Exposure Period	Several Weeks to a Year	Several Minutes to 4 Hours	Several Hours to 10 Days
Foil Size	2" dia. x 0.10"	2" dia. x .005"	2" dia. x .001"
Counting Method	Single or Multi-Channel Gamma Spectrometer Beta Counter		Beta Counter
Typical Sensitivity** (nvt per cpm)	1×10^7 n/cm ² *	1×10^3 n/cm ²	3.4×10^5 n/cm ²

*Based on using a 3"x3" NaI Crystal
**Sensitivity can be increased by using more than one assembly in
a single location.

The cobalt, indium, and gold foils can also be used by themselves,
outside of the NFI, to obtain the thermal neutron fluence.

Sampling	Continuous
Performance	Unaffected by reasonable environment. Energy response essentially uniform from 20 keV + 20 MeV Response characteristics are non-directional
Requirements	Power: None required Size: (6 1/2" Dx6 1/2" H) Weight: (8 lbs.)
Features	High sensitivity, high range, reliable, no high flux limitations
References	Manufacturer's specifications
Cost	\$245 not including foils
Address	Reactor Experiments, Inc., 963 Terminal Way San Carlos, CA 94070 Dean C. White, Technical Sales (415) 592-3355

Reactor Experiments, Inc.

Class Portable

Principle of Neutron dosimeter foils 12.7 mm (0.5") diameter
Operation

<u>Material</u>	<u>Cat. No.</u>	<u>Thickness</u> (mm) (inches)		<u>Nominal</u> <u>Purity</u> (percent)	<u>Price per</u> <u>Box of 10</u>
Aluminum	505	0.76	0.030	99.963	\$ 80.00
Aluminum	505A	0.13	0.005	99.991	65.00
Aluminum	505B	0.05	0.002	99.999	75.00
Ammonium-Sulfate	538A	3.2	0.125	99.998	85.00
Cadmium Cover Set	531A	1.0	0.040	99.9	110.00
Cadmium Cover Set	531	0.5	0.020	99.9	100.00
0.47% Cobalt-Aluminum	533A	0.013	0.005		200.00
Cobalt	506A	0.05	0.002	99.896	110.00
Copper	509A	0.25	0.010	99.998	75.00
Copper	509	0.13	0.005	99.998	80.00
Copper	509B	0.013	0.005	99.997	80.00
Dysprosium	581	0.025	0.001	99.995	145.00
7.90% Dysprosium-Aluminum	507B	0.013	0.005		215.00
0.155% Gold-Aluminum	546A	0.1	0.004		205.00
Gold	521B	0.025	0.001	99.995	140.00
Hafnium	553	0.05	0.002	99.957*	160.00
Indium	501	0.03	0.005	99.99	95.00
Iron	515	0.13	0.005	98.643	120.00
4.45% Lutetium-Aluminum	508	0.1	0.004	99.963	250.00
Magnesium	514	0.13	0.005	99.781	80.00

*Includes 2.94% Zirconium

Reactor Experiments, Inc.

Material	Cat. No.	Thickness (mm)	Thickness (inches)	Diameter (mm)	Diameter (inches)	Nominal Purity (percent)	Price per Box of 10
81.3% Manganese- Copper	516	0.05	0.002			99.722	\$ 85.00
Molybdenum	527	0.075	0.003			99.985	70.00
Neptunium-237	578	50 μg per cm^2		19	0.75		325.00 per foil
Nickel	513A	0.25	0.010			99.981	75.00
Nickel	513B	0.075	0.003			99.981	\$ 75.00
Niobium	575	0.13	0.005			99.837	100.00
Plutonium-239	576	30 μg per cm^2		19	0.75	99.22	325.00 per foil
Rhodium	540A	0.025	0.001			99.97	90.00
Scandium	580	0.13	0.005			99.64	255.00
Silver	511	0.13	0.005			99.96	85.00
Sodium Chloride	543B	2.0	0.075			99.99	\$ 85.00
Sulfur (1" diam.)	522A	4.75	0.187			99.99	80.00
Sulfur	522B	2.0	0.075			99.98	75.00
Tantalum	534	0.13	0.005			99.98	75.00
Thorium	512B	0.1	0.004			99.885	150.00 per 10
Tin	551	0.1	0.004			99.57	\$ 75.00
Titanium	535	0.25	0.010			99.876	140.00
Tungsten	526	0.15	0.006			99.96	115.00
Uranium (natural)	503	0.175	0.007			99.97	110.00
Uranium (depleted)	504L	0.025	0.001			99.958	145.00 for 5 foils
Uranium (enriched)	577	30 μg per cm^2		19	0.75	99.06	\$275.00 per foil
Vanadium	517	0.075	0.003			99.968	75.00
Zinc	552	0.13	0.005			99.99	75.00
Zirconium	541	0.13	0.005			99.82	75.00

Features Each foil is thoroughly cleaned and accurately weighted to 0.0001 g with a weight code embossed directly on the foil and 28 page "Foil Instruction Manual". Flux wires, fission foils, and cadmium tubing is also available.

Address Reactor Experiments, Inc.
963 Terminal Way
San Carlos, CA 94070
(415) 592-3355

Environmental Radiation Monitoring System
Sentry 1011

Class	Neutron/gamma detectors
Principle of Operation	High pressure spherical argon-filled ionization chamber, solid state electrometer, remote electronics station, central processing center.
Sensitivity and Range	Background to 10R/hr
Sampling	Continuous
Performance	Accuracy: $\pm 10\%$ Temperature: -24°C to 54°C Response time: 5 sec. Humidity: 0 to 100% (water proof)
Requirements	Power: 300 VDC Weight: Sensor housing 23 lb. Remote housing 50 lb.
Features	Low/intermediate sensors and electrometer; high level electrometer; remote station: transmitter/receivers; 16 remote stations; central processing center.
References	Manufacturer's Specifications
Cost	On request
Address	Reuter-Stokes 18530 S. Miles Parkway Cleveland, OH 44128 (216) 475-3434

Neutron Detectors

Model	Maximum Diameter	Maximum overall length	Thermal Neutron flux range	Voltage Range (volts)	Price
<u>$^{10}\text{BF}_3$ Proportional Counters</u>					
RS-P1-0403-102	1.31cm	30.63cm	...to $3 \times 10^4 \text{nv}$	1300 to 1600	\$ 410.00
RS-P1-0809-101	2.62cm	30.63cm	...to $3 \times 10^4 \text{nv}$	1300 to 1600	225.00
RS-P1-0809-201	2.62cm	30.80cm	...to $3 \times 10^4 \text{nv}$	1300 to 1600	265.00
RS-P1-0812-201					575.00
RS-P1-1613-144	5.16cm	40.32cm	...to $4 \times 10^3 \text{nv}$	2100 to 2800	290.00
RS-P1-1613-203	5.16cm	40.32cm	...to $4 \times 10^3 \text{nv}$	2100 to 2800	320.00
RS-P1-1613-205					675.00
<u>Helium-3 Proportional Counters</u>					
RS-P4-0203-201 (40atm.)	0.68cm	12.4cm	...to $5 \times 10^4 \text{nv}$	1850 to 2100	470.00
RS-P4-0406-212 (10atm.)	1.31cm	22.1cm	...to 10^4nv	1000 to 1200	500.00
RS-P4-0806-207 (4 atm.)	2.62cm	24.45cm	...to 10^4nv	1050 to 1250	485.00
RS-P4-0810-104 (10atm.)	2.62cm	34.3cm	...to $3 \times 10^3 \text{nv}$	1200 to 1500	810.00
RS-P4-0810-204 (10atm.)	2.62cm	34.3cm	...to $3 \times 10^3 \text{nv}$	1200 to 1500	845.00
RS-P4-0810-205 (4 atm.)	2.62cm	37.6cm	...to $4 \times 10^3 \text{nv}$	3100 to 3900	550.00
RS-P4-1808-207 (4 atm.)	5.75cm	26.57cm	...to $2 \times 10^3 \text{nv}$	1250 to 1450	1665.00
<u>Helium-4 Proportional Counters</u>					
RS-P5-0806-201 (10atm.)					320.00
RS-P5-0806-203 (15atm.)					320.00
<u>Fission Counters</u>					
RS-P6-0805-134					625.00
RS-P6-1608-110	5.16cm	30.80cm	...to 10^6nv	300 to 800	2075.00
RS-P6-2403-114	6.35cm	11.91cm	...to 10^9nv	300 to 700	1370.00
<u>B-10 Lined Proportional Counters</u>					
RS-P7-0805-127					180.00
RS-P7-0812-117	2.62cm	39.04cm	...to $2.5 \times 10^4 \text{nv}$	600 to 850	235.00
RS-P7-0826-126					350.00
<u>Compensated Ion Chambers</u>					
RS-C1-1210-135	3.89cm	34.45cm	10^4 to 10^{11}nv	200 to 700	3770.00
RS-C2-2514-115	8.02cm	60.32cm	2.5×10^2 to $2.5 \times 10^{10} \text{nv}$	300 to 800	2555.00
<u>Uncompensated Ion Chambers</u>					
RS-C2-0808-129	2.62cm	23.02cm	10^4 to 10^{11}nv	100 to 700	1630.00
RS-C2-2510-104	8.02cm	33.18cm	2×10^4 to $2 \times 10^{10} \text{nv}$	200 to 800	1305.00
RS-C2-2511-137	8.02cm	36.83cm	10^3 to 10^9nv	500 to 1000	on request
RS-C2-2599-125	8.02cm	317.82cm	5×10^2 to $5 \times 10^{10} \text{nv}$	200 to 1000	on request
RS-C2-2599-132					
<u>Fission Chambers</u>					
RS-C3-2510-114	8.02cm	33.18cm	...to 10^6nv	400 to 700	\$1370.00
<u>In-Core Fission Chambers</u>					
RS-C6-0201-231	0.478cm	5.56cm	...to $1 \times 10^{14} \text{nv}$	100V	on request
RS-C6-0201-288	0.648cm	5.24cm	...to $2 \times 10^{14} \text{nv}$	25 to 100V	on request

Address

Reuter-Stokes
18530 South Miles Parkway
Cleveland, OH 44128
(216) 475-3434

Neutron Dose-Rate Monitor
REM-PUG

Class	Portable, Survey
Principle of Operation	BF3 Proportional neutron detectors 10" polyethylene spherical moderator (LASL designed)
Sensitivity and Range	0.1 m Rem/hr to 10 Rem/hr log response
Sampling	Continuous
Performance	Follows Rem curve closely
Requirements	Power: 3D cells Size: 10" x 10" x 15" Weight: 23.5 lbs
Features	speaker; thermal to 7 MeV and higher; good Rem response
References	
Cost	\$1550.00
Address	Technical Associates 7051 Eton Avenue Canoga Park, CA 91303 (213) 883-7043

Neutron Monitor and Neutron Probes

Class Portable: Probes used with survey meters, scalars, ratemeters

Principle of ZnS(Ag) scintillators and photomultiplier tubes, polyethylene
Operation moderator (PUG-1N)

Specifications:

MODEL	TYPE	ENERGY	EFFICIENCY	SIZE	WEIGHT	COST
PNS-19	scintillation hornyak button n,p reaction	FAST energy response $\pm$ 30% from 1/4 MeV to 14 MeV	5 cpm per neutron/cm ² / sec. based on 5 MeV neutrons	2" diam x 8"L	1 lb.	\$345.00
PNS-20	scintillation Li 6 in ZnS(Ag) matrix neutron, alpha reaction	thermal	60 cpm/neutron per cm ² /sec.	2" diam x 8"L	1 lb.	\$425.00
PUG-1N	detection system includes PNS-20 plus moderator with polyethylene inner sleeve and boron loaded plastic outer shield	epithermal to fast		moderator approx. 5" diam. x 6"L	8 3/4 lb to- tal wt.	\$800.00
PUG-1AB Survey Meter with attached detection system						
Range: 0-500, 500, 50,000 500,000 cpm						

References Manufacturer's Specifications

Address Technical Associates
7057 Eton Avenue
Canoga Park, CA 91303
(213) 883-7043

Neutron Detectors
Series 9300 Texlium Detectors

Class	Detector for neutron spectrometry																								
Principle of Operation	Texlium detectors utilize the reaction $^3\text{He} + ^1_0\text{n} \rightarrow ^1_1\text{p} + ^3_1\text{H} + 764 \text{ keV}$. Various gas pressures Capacitance: 5 to 8 pf. Pulse output: ~ 0.28 $\mu\text{Coulombs}$ or equivalently 14 mV into 20 pf. Sensitivity: thermal - fast neutrons																								
Sampling	Continuous																								
Performance	Capacitance: 5 to 8 pf. Pulse output: ~ 0.28 $\mu\text{Coulombs}$ or equivalently 14 mV into 20 pf. Use time: ~ 3 μsec maximum Operating voltage: see Fig. 1 Slope: 2%/100 volts Operating temperature: -70°C to 200°C																								
Model Numbers																									
<table><tr><th>Filling Pressure</th><th>4 inch</th><th>6 inch</th><th>12 inch</th></tr><tr><td>1 atm.</td><td>9301</td><td>9315</td><td>9329</td></tr><tr><td>2 atm.</td><td>9303</td><td>9317</td><td>9331</td></tr><tr><td>4 atm.</td><td>9305</td><td>9319</td><td>9333</td></tr><tr><td>6 atm.</td><td>9307</td><td>9321</td><td>9335</td></tr><tr><td>10 atm.</td><td>9311</td><td>9325</td><td>9339</td></tr></table>		Filling Pressure	4 inch	6 inch	12 inch	1 atm.	9301	9315	9329	2 atm.	9303	9317	9331	4 atm.	9305	9319	9333	6 atm.	9307	9321	9335	10 atm.	9311	9325	9339
Filling Pressure	4 inch	6 inch	12 inch																						
1 atm.	9301	9315	9329																						
2 atm.	9303	9317	9331																						
4 atm.	9305	9319	9333																						
6 atm.	9307	9321	9335																						
10 atm.	9311	9325	9339																						
Requirements	Power: Requires 900-1400 volts dc between 2 - 10 atm Size: 2.54cm (1") outside diameter; (4.18") + active length Weight:																								
Features	The 9300 series are supplied with HN type detector. Detectors with diameter ranging from 1/4 inch to 2 inches with special envelope material such as aluminum or copper, and other custom designs are available upon special request.																								
References	Manufacturer's Specifications																								
Cost	Consult factory																								
Address	Texas Nuclear Division Ramsey Engineering Company Box 9267 Austin, Texas 78766 (512) 836-0801 Tele: 77-6413

Neutron Detector
Texas Nuclear Model 9341

Class	Detector
Principle of Operation	A neutron detector using the reaction $^3\text{He}(n,p)\text{T}$.
Sensitivity and Range	Sensitivity: thermal and fast neutrons Energy Range: 100 keV to 1 MeV, up to about 10 MeV using a pulse rise time discriminator. Active Length: 6 inches Resolution: 7 to 10%
Performance	Gamma rejection: up to several R/hr Operating voltage: ~1900 volts Temperature: -70°C to 200°F Filling gas: 4 atm. ^3He + 2 atm. Kr Plateau length: ~500 volts
Requirements	Power: Model 9341, ~1900 Vdc Size: 25cm (10.18 in. including HN connector) long Model 9341, one inch diameter Weight:
Features	
References	Manufacturer's Specifications
Cost	Model 9341 \$305.00
Address	Texas Nuclear Division Ramsey Engineering Company Box 9267 Austin, TX 78766 Tele: 77-6413

Neutron Dose Equivalent Monitor

Harwell Type 0949

Class	Portable
Principle of Operation	The neutron detector is a polyethylene sphere, with an intermediate perforated cadmium layer, at the center of which is a spherical proportional detector with a ^3He gas mixture combined with a ratemeter that has a liquid crystal display
Sensitivity and Range	Energy range: thermal to 7 MeV Doserate range: 1 mrem/hour to 10 rem/hr Integrated dose range: 1 mrem to 9999 mrem
Sampling	Continuous
Performance	20.8 cm diameter polyethylene sphere Accuracy: Within a factor of 2 (i.e., +100% to -50%) of the 1 ICRP recommendations (1963) for neutron energy Gamma sensitivity range: The gamma rejection ratio is >3000:1 for gamma energies up to 7 MeV Response time: 2.5 seconds or 20 seconds depending on choice of update period Polar response: To broad beam radiation, within $\pm 2\%$ irrespective of instrument orientation.
Requirements	Power: Two PP6 9 volt batteries, approximately 60 hr life Size: 32x 25 x21cm Weight: 6 kg
Features	Battery condition is continuously monitored and a low voltage is indicated by the appearance of two pulsating colors on the LCD.

References	Manufacturer's Specifications
Cost	Available on request
Address	AERE Harwell Instrumentation and Applied Physics Division Bldg. 347.3 Oxfordshire, OX11 0RA England Tele: (0235 24141, Ext. 4766 Attn: R.H.W. Sherwood

Neutron Dose Equivalent Rate Monitor
Harwell Type 0949-2

Class	Portable
Principle of Operation	The neutron detector comprises a 20.8cm diameter polyethylene sphere, with an intermediate perforated Cd layer, at the center of which is a spherical proportional detector with a ^3He gas mixture.
Sensitivity and Range	Rate Range: 1.0 mrem/hr to 10 rem/hr, dose equivalent Integrated Range: 0 - 9999 mrem Neutron energy range detected: thermal to 7 MeV
Sampling	Continuous
Performance	Gamma sensitivity: the gamma rejection ratio is greater than 3000:1 for gamma energies up to 7 MeV Response time: 1.0 mrem/hr to 10 rem/hr - 2.5 sec. or 20 sec. depending on chosen update period. Polar response: To broad beam radiation, within $\pm 2\%$ irrespective of instrument orientation Display: 4 digit liquid crystal display
Requirements	Power: 2 - 9W batteries; life $\sim$ 60 hours, intermittent use Size: $\sim$ 32x25x21cm Weight: $\sim$ 6 kg
Features	Battery condition is continuously monitored and a low voltage is indicated by the appearance of two pulsating colons on the LCD.

References	Manufacturer's Specifications
Cost	On application
Address	AERE Harwell Instrumentation and Applied Physics Division Bldg. 347.3 Oxfordshire, OX11 0RA ENGLAND Tele: Abingdon (0235) 24141, Ext. 4766

High Sensitivity Neutron Survey Meter

Victoreen Model 488A

Class	Portable, handheld
Principle of Operation	Uses a 4" boron-lined neutron proportional counter, moderator, amplifier, ratemeter, meter readout
Sensitivity and Range	Range: Thermal to 14 MeV Thermal neutrons, four ranges: 0-800 cpm 0-8,000 cpm 0-80,000 cpm 0-800,000 cpm These ranges correspond to 0-12 thermal neutrons/cm ² /sec, x1, x20, x100, x1000. Gamma Sensitivity: Completely insensitive to gamma radiation fields to 500 R/hr.
Sampling	Continuous
Performance	Precision: $\pm 10\%$ fullscale deflection Time Response Switch selected, 1.5 seconds, 5 seconds, 15 seconds Moderator. high density polyethylene encased in 1/32 in. cadmium shield 2.54cm (1 in.) wall thickness, overall moderator diameter 7.9cm (3 1/8"). Drift. less than 5% over 24 hours Temperature operating range: -30°C to +50°C -20°F to +120°F Temperature dependence. less than 0.05%/°C Humidity range: 0-99%, noncondensing
Requirements	Power: Two "D" cells Size: 30.5cmLx11.4cmWx25.4cmH (12"x4 1/2"x10") Weight: 3.85 kg (8.5 lbs)

INSTRUMENTATION
FOR ENVIRONMENTAL
MONITORING

RAD-NEU
Survey
Victoreen
Page 2

Features	The detector assembly may be utilized in three modes -- either bare tube, tube with polyethylene moderator, or tube with moderator and cadmium shield. With these combinations the 488A detects thermal neutrons, moderated fast and thermal neutrons, and only fast neutrons respectively.
Cost	\$1300.00
Address	Victoreen, Inc. 10101 Woodland Avenue Cleveland, Ohio 44104 (216) 795-8200

III.G. Personnel Dosimetry

1. Introduction
2. Design Considerations
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1. INTRODUCTION

In this section we shall discuss instrumentation used to measure integrated external radiation doses to personnel. The archetype of such instrumentation is the "film badge", but of course this discussion will be of broader scope; we shall deal with the general measurement problem of integrating dose instruments designed to be pinned to clothing, carried in pockets, worn on fingers, or otherwise portable without the user's constant attention or even constant consciousness.

There are several reasons why such instruments are used. Perhaps the most commonly expressed reason is related to legal liability: in routine use at low integrated dose levels, the dosimeter protects the employer against liability for negligence, and is in many cases required by law for just such a purpose. A second important motivation, and certainly the prime operational motivation, is the assessment of dose in the event of an accident, as this can be a crucial aid in post-accident treatment. Third, when doses may approach maximum permissible levels, dosimeters are often used to limit individual exposures. Finally, the psychological benefit of dosimetry is often cited; some individuals ignorant as to the true meaning of radiation dose somehow feel "safer" wearing a dosimeter, as if "big brother" were looking after them.

2. DESIGN CONSIDERATIONS

Each of these motivations places a different set of constraints upon the dosimetry system. Thus the whole question of accuracy is related both to the legal-liability situation and to the requirements imposed by decision-making in post-accident treatment. Also, the desirability of long-term dosimeter storage for legal purposes is sometimes cited. Further, the portability required of the dosimeter imposes size-and-weight restrictions, while cost considerations demand that large numbers of readings be possible at low unit-cost. All of these factors must be considered in any discussion of personnel dosimetry.

Obviously, "the more accurate the better". However, certain basic limitations prevent the attainment of extremely precise measurements. The most important is the problem of sampling. A dosimeter can only measure the radiation which strikes it, but the problem arises when one relates the dosimeter reading to an "equivalent whole body external dose." Several factors contribute here; for example, low-energy beta activity in the air can register in a sensitive dosimeter and yet be effectively shielded by a couple of layers of clothing. Also, the angle of incidence of the radiation is important: exposures from one side of the body may be misrepresented by some dosimeter

geometries, especially if the dosimeter is at the front while radiation is incident at the back of the body. Another problem occurs when dose is delivered to a part of the body (e.g., hands or feet) distant from the dosimeter.

Dosimetry thus presents a nearly impossible problem if accuracies in the few-percent range were to be required. Fortunately, no such need exists, from the viewpoint of either post-accident treatment or legal responsibility.

To illustrate these points, we shall quote from ICRP Publication 12 (Ref. 1):

"A dosimeter carried on the surface of the worker's body is best regarded as a sampling device. It provides a sample of the dose received at the surface of the body during the movement of the worker through his environment. It does not provide a direct measurement of the radiation dose to organs or tissues, except for the dose to the skin in the immediate vicinity of the dosimeter. Neither does it necessarily provide an adequate assessment of the situation in the workplace because it measures the dose only at one point on the body of each worker.

"It is only in exceptional cases, usually associated with substantial accidental exposures, that attempts need be made to estimate the actual organ doses, rather than doses at the surface of the body. The greatest uncertainties in the assessment of organ doses from routine monitoring are caused by the fact that a small number of dosimeters, often only one, has to be taken as representative of the exposure of the whole surface of the body. Unless the dosimeter results are representative, there is little point in conducting detailed calculations of the depth dose in a composite, finite medium -- the human body.

"The need to use many dosimeters and the complexity of the necessary interpretation to assess the actual organ doses prohibit the making of these assessments as part of routine individual monitoring. A simplified system of interpretation has to be adopted...and this will usually have the effect of assessing upper limits to organ doses rather than actual values...

"It has become conventional to note and record all radiation doses above the threshold of detection of the personal dosimeter. This detailed attention to low doses may, in some cases, focus unrealistic attention onto exposures which, if they cause any risk at all, cause risks which are trivial in comparison with other risks of life. Other exposures, often of the same order of magnitude, are not recorded,

either because they are below the threshold of detection or because the individuals are not monitored. The Commission [ICRP] has now recommended that individual monitoring and associated records are not needed if the doses are most unlikely to exceed 3/10 of the annual Maximum Permissible Doses. It is possible to make use of this recommendation to make individual monitoring simpler and more logical...

"In principle, the choice of an investigation level would correspond exactly to 3/10 of the appropriate annual Maximum Permissible Dose...

"In practice, however, such a precise equivalence cannot be achieved because most individual monitoring for external radiation is carried out over periods substantially shorter than a year. It is therefore recommended that, for the whole body, the investigation level for penetrating radiation should be set somewhat lower--50 mrem for a dosimeter issued for 2 weeks, 100 mrem for 1 month, or 300 mrem for 3 months. In the rare cases when a substantial part of the dose to an individual results from neutrons or from internal contamination, the investigation level should be applied to the total recorded dose in a control period rather than to each of the separate contributions...

"The uncertainties acceptable in routine individual monitoring should be somewhat less than the investigation level and can best be expressed in relation to the annual dose. The uncertainty in assessing the upper limits to the annual dose equivalent to the whole body or to the organs of the body...should not exceed 50%. Where these doses are less than 2 rems an uncertainty of 1 rem is acceptable. This uncertainty includes errors due to variations in the dosimeter sensitivity with incident energy and direction of incidence, as well as intrinsic errors in the dosimeter and its calibration."

For the purposes of our discussion here, suffice it to say first that the obvious accuracy limitations on external personnel dosimetry must be recognized by the user; second, that when high accuracies are required other methods must be called upon to complement the dosimeters of concern here; and finally, that most people realize that the need for great intrinsic accuracy is not the most significant factor in the choice of dosimeters.

What is important is an assurance that no substantial component of the absorbed dose is missed entirely by the dosimetry system

being used. Thus one needs to base a dosimetry system upon detailed information about the types of radiation likely to be encountered.

We shall concentrate here upon a discussion of the types of dosimeters; their relative sensitivities to various particle species and energy ranges; and their advantages and disadvantages.

3. DOSIMETER TYPES

For routine daily monitoring, there are two basic types of dosimeters in common use today: film and thermoluminescent dosimeters (TLD's). Film is the more commonly used, and historically has been the mainstay of personnel dosimetry systems. TLD's are now achieving increasingly widespread use. The most important subject of debate today in this field seems to concern the relative merits of film and TLD.

Several other types of dosimeters are also in use, usually for special applications. Among these are pocket ionization chambers, activation detectors, radiophotoluminescent (RPL) dosimeters, and thermally-stimulated-exoelectron-emission (TSEE) devices.

We shall attempt here to discuss the basic properties of each of these various types, as well as their various regions of applicability. We shall also indicate those new techniques which show possible promise for dosimetry, but still require extensive investigations and development.

There are two basic types of external radiation fields which are commonly monitored: beta/gamma and neutron fields. We shall discuss these separately, since the latter involves techniques and considerations completely different from the former.

4. GAMMA/BETA DOSIMETRY

A. Film

The basic property of film dosimeters is the ionization-induced darkening of emulsion. The intrinsic limitations of film are due to four problems: first, some (unavoidable) background darkening occurs in the absence of any ionization; second, film must be wrapped to avoid light leaks, and even for thin wrappings the dosimeter is typically insensitive to alpha particles, low energy betas, and low energy x-rays; third, the high atomic number of typical emulsions makes the response non-tissue-equivalent for low-energy photons; and finally, emulsion darkening ultimately saturates at high doses.

The advantages of film include the ability to re-scan emulsions after an exposure to determine details of dose depositions; use of filter shadowing and exposure-uniformity information to measure directionality of exposure and source distance; and the very wide variety of possible film types and grades (although in actual fact the variety of films now commercially available is quite limited).

Besides the physical or engineering considerations, two other properties of film are often cited: low unit cost, and long-term storage (for re-measurement and legal record-keeping).

Here we shall present data about some typical films to indicate the approximate ranges of various parameters. We shall not try to discuss and compare all of the various commercially available films and film services; because of their large number, that would be an almost insurmountable task. Suffice it to say that film badge services have come under the study of the International Atomic Energy Agency (IAEA) and the U.S. Public Health Service, and documents containing overviews of the important considerations have been in the open literature for several years (Ref. 2, 3, 4). Also, a 1962 book by Becker (Ref. 5) contains an excellent summary of the basic properties and considerations in film dosimetry.

The three basic operational characteristics to consider are accuracy, dynamic range, and energy response.

The intrinsic accuracy of film is quite good. In one study (Ref. 6) under carefully controlled conditions, and changing films every week, x-ray exposures to Kodak Type 3 film of 10 to 200 mR/week (0.5 to 10.0 R/year) have been determined to $\leq \pm 20\%$ for a single film and $\leq \pm 10\%$ (standard deviation) on a quarterly basis. For a single 10 mR exposure, standard deviations of $\pm 25\%$ are possible. Of course, such "carefully controlled conditions" are not typical of actual operational use. One very important quality-control practice is to set aside and develop a few blanks and controlled-exposure films with each new batch of emulsions, to determine if any changes in sensitivity have occurred. The main point to be made here is that intrinsic accuracy is not a significant contributor to error in the use of film dosimeters.

The dynamic range of the more sensitive films is typically from a few mrad to a few thousand mrad, with films available which respond up to the megarad region; the typical dosimeter system might use films of both the most and least sensitive types in order to cover as large a dynamic range as possible. Film's linearity of response (emulsion density vs. dose) is excellent, until saturation sets in at the high-dose end. This is illustrated

in Figure 1 (from Ref. 7).

FIGURE 1. Optical density against roentgen exposure plotted on linear grid for Kodak Commercial film and for Kodak Type M films in Ready Packs (from Ref. 7).

The humidity and temperature effects are significant, however. Studies of DuPont 555 and Kodak Type 2 films have shown rapid losses of up to 40% in sensitivity at high humidity (Ref. 8). The effect is probably on latent image formation, possibly from "the action of the various radiolytic products of water upon the silver halide crystal, affecting either the migration of conduction band electrons, the fraction of silver ions free to migrate to the sensitivity centers, or both" (Ref. 8). Figure 2, from Becker (Ref. 14) shows the effect of 75 to 95% humidity, 23 to 32°C on optical density for a typical film.

FIGURE 2. Effect of 75 to 95% humidity, 23 to 32°C on optical density of Kodak Type 2 Film, as a function of storage time (from Ref. 14).

The non-uniformity of photon sensitivity of films as a function of quantum energy is considered by some to be their greatest drawback. At low energies, the intrinsic sensitivity of emulsions is much larger than at medium energies, in some cases by big factors. Much effort has gone into the design of various types of filters to compensate for this energy dependence.

A typical example of the use of filters for decreasing the energy dependence is the work of Storm and Shlaer at Los Alamos (Ref. 9). Figure 3 shows the way in which the unfiltered response is modified by using filters with one element (Bi), two elements (Bi, Er) and six elements (Bi, Au, Ta, Er, Cd, and Nd). The idea is that the incident x-rays are attenuated by the K-edge absorption of each successive material, decreasing the effective dosimeter sensitivity. Figure 4 shows the measured response of four commercial films using this technique. Use of a lead frame back was necessary to eliminate a scattering problem in the plastic encasing the filters.

FIGURE 3. The photon response of DuPont 502 film unfiltered and filtered by one-, two- and six-element filters (from Ref. 9).

FIGURE 4. Response of DuPont 508 and 555, Eastman Type II and Ilford PM-1 films in the lead-frame plastic badge (from Ref. 9).

The conclusion of Storm and Shlaer is that it is possible to eliminate much variation in sensitivity at the low energies in film dosimeters, at least down to energies of a few tens of keV. It should be noted that multi-filter holders are relatively expensive, which has discouraged their wide use. At the other (high end of the energy scale, response becomes poor at energies high enough that particle-equilibrium is not established upstream of the dosimeter (of course, this is true for all dosimeters under consideration in this section).

B. TLD

Thermoluminescent dosimetry (TLD) is becoming more and more widely used for routine gamma/beta dosimetry. The basic physical principle is now reasonably well understood. Ionizing radiation incident upon a crystal elevates an electron from the valance band to the conduction band, leaving a hole in the valance band. The electron and the hole move through the crystal until they either recombine

or are trapped in metastable states, usually associated with impurity sites. When the crystal is heated, sufficient energy may be given to the electron (or the hole) so that it wanders around, eventually recombining with a counterpart hole (or electron). The recombination energy is released as visible light (a thermoluminescent photon).

A good description of basic TLD dosimetry can be found in the excellent book by Cameron, Suntharalingam, and Kenney (Ref. 10). Also, Cameron has compiled a bibliography on the subject (Ref. 11).

In perusing the literature on TLD, one is impressed by a tremendous amount of enthusiasm for TLD among its proponents. This impression can be had from even a brief study of the proceedings of recent conferences on the subject of TLD (Ref. 12, 13, 14). We would like to emphasize here that both film and TLD seem to have some advantages in particular situations, and can be used to complement one another for mixed-field dosimetry.

The advantages of TLD over film are several: (a) reproducible accuracies under laboratory conditions have been achieved at the $\pm 5\%$ level; (b) the dynamic range is large, with doses of from a few mrem to many thousand rem measurable in the same crystal; (c) directional uniformity surpasses that of film: it is easier to design a dosimeter to reduce the directional dependence to $\pm 20\%$ down to low x-ray energies; (d) fading is minimal: TLD's do not fade significantly with extreme climatic conditions; (e) energy response is more uniform at low x-ray energies; (f) TLD dosimeters are rugged and reusable.

The disadvantages include the destructive readout (once read out, the information is lost); the inability to determine directionality of dose deposition; and slightly higher costs than film. The commonest way to avoid the problem of data lost by accident is use of a back-up chip; this is now found in many systems.

The most common thermoluminescent crystal used today is lithium fluoride. There is extensive documentation of the properties of LiF; we shall try to show some of the typical data. For example, Figure 5 (Ref. 15) shows that the useful range of LiF extends up beyond 10^5 rem; a typical sensitive film is shown for comparison. The linearity of response is good, extending down to below 10 mrem on the low-dose end. The directional sensitivity is shown in Figure 6 (Ref. 15); again, the uniformity is excellent.

The energy response of a typical TLD is shown in Figure 7 (from Ref. 16). The enhanced sensitivity in the 20 keV region is much more easily handled by filtering techniques than is the (more extreme) problem with film.

FIGURE 5. Useful range of TLD and film (from Ref. 15).

Still another advantage of TLD is its sensitivity to betas. LiF can be fabricated in very thin layers, and requires little wrapping, so that beta dosimetry is quite feasible.

Perhaps the two main reasons cited at laboratories where TLD has not replaced film are its difficulties when neutrons are present, and its destructive readout. We shall discuss the neutron measurement problem separately below.

The destructive readout is intrinsic to the method. One heats up the TLD chip, measures the integrated signal, and is then ready to re-use the chip. Systems are now commonly used to semi-automate the process; however, all information is lost, and in the event of some failure there is no second opportunity to fall back upon, except the back-up chip already mentioned.

Another frequently cited "disadvantage" of TLD (compared to film) is the legal, record-keeping requirement. The argument is given that possession of the actual exposed film dosimeter itself is somehow intrinsically preferable to the mere record from a TLD reading. However, it is now generally recognized that well-maintained records are sufficient evidence in any legal situation (Ref. 17).

The "fading" problem is an important consideration if dosimeters are not changed often. There are many situations in which large groups

FIGURE 6. TLD dosimeter response vs. angle of incidence (from Ref. 15).

FIGURE 7. Energy Response of TLD-700 thermoluminescent material (from Ref. 16).

of employees are monitored with no expectation that any detectable exposure will occur. For example, at Lawrence Livermore Laboratory, semi-annual or annual processing cycles were desired for many employees whose exposure risk is minimal, but it was estimated that "the maximum period that film should be used is three months. In practice, a one-month exposure period was never exceeded because of the significant aging and fading characteristics of the film dosimeter" (Ref. 18). The fading of LiF, by contrast, is considered to be of only minor concern. However, in order to double-check for anomalous or incorrect readings, a back-up TLD chip (of calcium fluoride, CaF_2) is used at Livermore; it is read out only in the event of such an anomaly (Ref. 18).

In addition to LiF, several other TLD materials have been studied and applied in personnel dosimetry. Among those in use today are $\text{CaF}_2\text{:Mn}$, $\text{CaSO}_4\text{:Tm}$, $\text{CaSO}_4\text{:Dy}$, Al_2O_3 , and fluorite. Some of these are used in combination with or as back-ups for LiF. Some of them also have properties which are improvements over one or more of the less desirable characteristics of LiF.

For example, although LiF is subject to tribothermoluminescence (mechanically induced luminescence) and other possible spurious luminescence, $\text{CaSO}_4\text{:Dy}$ is significantly less susceptible: a factor of 30 less tribothermoluminescent response was measured in one study (Ref. 19). Also, this study showed excellent low-dose sensitivity for $\text{CaSO}_4\text{:Dy}$ (down to ~ 0.2 mR), which can be of occasional usefulness in dosimetry.

Another drawback of LiF is that its thermoluminescence is strongly dependent upon the

thermal history of the material (Ref. 20, 21), so that the annealing technique must be quite carefully controlled. This is much less true of $\text{CaF}_2\text{:Mn}$ and CaSO_4 dosimeters.

C. Pocket Ionization Chambers

A frequently-used back-up or auxiliary dosimeter is a pocket ion chamber, usually pen-sized. It is typically used to assess (*in situ*) the dose being received by a worker in a high-radiation-level environment. These instruments are understood to be for x and gamma radiation only (although many will respond as well to beta radiation), and to be capable of either direct read-out or immediate indirect read-out with a separate instrument.

Specifications for such dosimeters have recently been established by the American National Standards Institute, Subcommittee N13.5 (Ref. 22). The electrodes of the ion chamber form a capacitor which is charged to a pre-determined voltage; ionization occurring in the chamber surrounding the electrodes causes a decrease in the charging voltage which can be read by an electrometer; or alternatively causes motion of a movable electrode, which motion can be read directly.

A typical instrument today can measure exposure (Roentgen) to about $\pm 15\%$ of full scale for 200 mR full scale settings, and undergoes discharge (leakage) at rates of at most a few tenths of a milliroentgen per hour, for 200 mR full scale (Ref. 22). Since it is meant for use on at most a daily (8-hour or perhaps 24-hour) basis, it is a very useful instrument for daily or even hourly dose assessment. Pocket ion chambers can now be obtained in a variety of rugged, sealed, environment-insensitive design; their disadvantage is an almost total insensitivity to betas, alphas, and neutrons.

D. Other Dosimeters

Several dosimetry techniques are now in the research-and-development stage, and are not yet fully operational. Among the concepts being studied for gamma dosimetry are radiophotoluminescence (RPL) and thermally stimulated exoelectron emission (TSEE). Neither RPL nor TSEE is now used in any commercially available systems.

(i) RPL

In radiophotoluminescence, radiation produces defects in a solid (usually a glass); ultraviolet light can subsequently be used to produce fluorescence in the glass. The amount of fluorescence is related to the radiation absorbed. UV stimulation is not a destructive readout, however, which is a clear difference from

TLD. On the other hand, because dose is continually integrated an RPL dosimeter eventually saturates and must be discarded.

Figure 8 (Ref. 23) shows the absorption and luminescence response curves for one typical RPL glass.

FIGURE 8. Typical absorption and luminescence curves for an RPL glass: (A) absorption characteristics prior to irradiation, (B) absorption after irradiation, (C) luminescence produced by illumination in the absorption band (from Ref. 23).

Most RPL glasses suffer from fading over long periods of time. Another problem is the "pre-dose" RPL, which is the amount of response even before irradiation. Today, pre-doses equivalent to a few hundred mrad are about the lowest available (Ref. 24). Among the glasses being investigated are aluminum and lithium phosphates and lithium borate, all doped with silver phosphate. Figure 9 shows the calculated energy dependence of a lithium borate RPL glass containing various amounts of AgPO_3 . It can be seen that this material is quite energy-independent down to a few tens of keV. Among other drawbacks (Ref. 26) of glass dosimeters are the poor sensitivity to β 's of encapsulated glasses; and the relative insensitivity to low energy x-rays (≤ 50 keV).

Recent summaries by Cameron (Ref. 23), Becker (Ref. 24, 27) and Yokota (Ref. 26) can be consulted for more detailed information about RPL. We will summarize here by stating that RPL has several attractive features, which still require development work to produce a workable beta/gamma dosimetry system.

(ii) TSEE

Thermally-stimulated exoelectron emission (TSEE) is a process quite similar physically to thermoluminescence; indeed, many substances exhibit both TSEE and thermoluminescence. The basic concept is

that radiation impinging on the TSEE material elevates an electron from the valance band to the conduction band, leaving a hole in the valance band. The electron and hole move about until trapped in impurity sites. When the crystal is heated, the electron may pick up enough energy to overcome the work-function and be emitted: hence, thermally-stimulated exoelectron emission. The number of emitted electrons is measured in some counting or integrating device (e.g., G-M tube, ionization chamber).

Reviews of recent research in TSEE dosimetry have appeared in the two most recent conferences on TLD (Ref. 13, 14). Becker (Ref. 29, 30) has also recently reviewed the subject. Among the features which commend TSEE are extreme sensitivity (down to even the 10^{-6} R region); significantly reduced problems with poor crystal purity; low background and "pre-dose" characteristics; the small thickness of the sensitive layer (perhaps less than 10 nm); and the possibility of excellent energy response down to low x-ray energies, by the use of several materials. The disadvantages, at present, include poor "fading", so that readout must take place within at most a few days of exposure; and the more complicated read-out system required. At present, TSEE still requires considerable work on both the basic physics and the engineering details, before it can supplement the more widely-used systems.

FIGURE 9. Calculated energy dependence of a lithium borate RPL glass containing different amounts of silver phosphate (from Ref. 23).

(iii) Track Etching

This technique relies upon the fact that certain materials suffer local radiation damage from very high-LET radiation, the sites of which are preferentially attacked by certain etchants. This technique, which is principally used for neutron dosimetry, will be discussed below under "Thermal Neutrons".

5. NEUTRON DOSIMETRY

The problem of neutron personnel dosimetry is discussed separately from the gamma/beta problem for three reasons: First, a large fraction of radiation exposure involves no neutron fluxes at all; second, the present instrumentation for neutron dosimetry is in many important cases not adequate to the task, unlike that for gamma/beta dosimetry; and third, the dosimetry of high-LET radiations (of which neutron dosimetry is but one part) has been of increasing concern in recent years.

For details on the various energy divisions, properties, and relative importances of neutron fluences, the reader is referred to "Neutron Monitoring" elsewhere in this volume. In what follows here, we shall assume a familiarity with that section.

Baarli (Ref. 31) has discussed the problem of personnel dosimetry in mixed radiation fields around high-energy accelerators; the conclusion is drawn that present techniques are adequate only if the approximate composition and energy spectra of the component radiations are understood reasonably well.

At the present time, separate dosimeters are in common use for thermal and for fast neutron detection. Intermediate-energy personnel dosimetry is not often attempted, and high-energy neutron personnel dosimetry is performed using techniques (mainly, foils activation) already described in detail in "Neutron Monitoring."

There are basic physical reasons for the difficulty in monitoring intermediate-energy neutrons using a portable dosimeter. Thermal neutrons can be captured in materials with high thermal-neutron cross-sections, and fast neutrons can be detected by the charged secondaries produced in (n,p) interactions in hydrogenous (or other low-Z) material. However, neutrons in the intermediate-energy region (between ~0.5 eV and ~200 keV) are not easily detected except by the use of a moderator and a thermal-neutron detector.

As discussed in "Neutron Monitoring," relatively few situations exist in which a

FIGURE 10. Response of NTA packets to neutrons of various energies (from Ref. 32).

substantial fraction of the dose is from thermal neutrons; the main exception is near a thermal column at a reactor. However, thermal-neutron detectors are important because many indirect methods of assessing higher-energy neutron fluences actually rely on thermal-neutron detectors.

We shall begin by discussing fast-neutron and then thermal-neutron dosimeters, and finally body backscattering and albedo dosimeters.

A. Fast Neutrons

The basic technique most commonly used for fast neutrons (~200 keV to ~20 MeV) is the nuclear track emulsion. A typical emulsion of this type is Eastman Kodak Type A (often called "NTA") emulsion, which has been described in detail by Lehman (Ref. 32) and Becker (Ref. 5). NTA is a special fine-grained film, typically 25 to 35 microns thick. Neutrons are detected by the trail of ionization from charged particles released by one of three mechanisms: (a) elastic (n,p) interactions with hydrogen, which register in NTA above about 0.4 MeV; (b) exoergic (n,p) reactions with nuclei, which register in NTA below about 10 eV and then only on nitrogen nuclei; and (c) inelastic nuclear interactions resulting in a "star", significant about 20 MeV.

The energy response of NTA is shown in Figure 10, from Lehman (Ref. 32). The sensitivity is seen to be significant only for fast and thermal neutrons. Standard practice is to report the response of neutron film in tracks per unit area, as seen through a microscope. There are difficulties in relating the measured quantities to neutron flux (and hence to dose-equivalent), mainly because some of the response comes from interactions in the emulsion wrapping, and some from interactions in the emulsion itself; varying emulsion thicknesses (batch to batch) complicate calibration procedures. One problem with this class of detectors is that it is susceptible

FIGURE 11. Estimated NTA film response (from Ref. 34).

FIGURE 12. NTA film badge response as a function of neutron incidence angle (from Ref. 35).

to hydration at high humidities, which increases the hydrogen content and hence the fast-neutron sensitivity. This and other problems (temperature effects, storage problems) have been discussed by Lehman (Ref. 32) and others (Ref. 33, 34, 35). Figures 11 (Ref. 34) and 12 (Ref. 35) show the fast-neutron response of NTA film as a function of incident angle.

Figure 13 (Ref. 35) shows the effect of 70% relative humidity on latent image stability of NTA film. Desiccation followed by the heat-sealing of Kodak NTA Type B film in moisture-proof bags has been successfully used for several years by a British group. Their unpublished results (Ref. 45) show that the 2- to 4-week latent-image half-life of NTA film can be extended to about 8 weeks using this technique. The bags are made of a laminate of paper/0.012 mm Al foil/polyethylene. Marshall and Stevenson (Ref. 36) have considered the edge-effects problem in scanning of emulsion, while Oshino (Ref. 37) has studied the energy dependence of NTA worn on a phantom.

FIGURE 13. Effect of 70% relative humidity on latent image stability of NTA film (from Ref. 35).

Although fast neutron dosimetry has been carried out for almost two decades using nuclear track emulsions, it is clear that some features are in need of improvement. However, in situations where neutron spectra are known to be dominated by fast neutrons, and where high humidity is not a problem, this technique can be used with reasonably good results.

Two other methods of fast neutron personnel dosimetry involve foils activation or fission fragment foils, in either case resulting in darkening of film by charged particles (from decay or fission fragments). These will be touched on briefly in the next section, on thermal neutron detectors.

B. Thermal Neutron Detectors

Thermal neutron personnel dosimetry is commonly performed today with one of three means: thermoluminescent dosimeters, fission foil dosimeters, or film surrounded by a foil which is activated and then decays.

The commonly used TLD method relies upon two isotopic compositions of lithium fluoride, ^6LiF and ^7LiF . Lithium-6 has a thermal-neutron cross-section of about 960 barns for the process $^6\text{Li}(n,\alpha)^3\text{H}$; the cross-section of lithium-7 is smaller by a factor of about 25,000. The ^7LiF TLD responds to photons with the characteristics already described, while the ^6LiF responds to both photons and thermal neutrons. Harvey et al. report that the response of a ^6LiF TLD to 1 rem of thermal neutrons is approximately 80 times the response to 1 R of gamma radiation (Ref. 28), and that the minimum detectable thermal neutron dose is ≤ 0.5 mrem. However, "it is necessary to calibrate each batch of ^6LiF separately, as significant differences in sensitivity may occur between batches, and the ratio of neutron to gamma sensitivity is not constant" (Ref. 28). Fast neutron sensitivity of LiF is insignificant. "The response in equivalent gamma rads per fast neutron tissue rad is roughly proportional to neutron energy, ranging from 1% at 1 MeV to 10% at 10 MeV" (Ref. 28).

Many of the basic advantages of and problems with TLD's are identical to those of TLD's in general, which have already been discussed.

The fission foil technique exploits the property of certain crystalline materials, plastics, and glasses, whereby radiation damages the materials so that certain etchants will preferentially attack the damaged areas. In particular, some materials only produce "tracks" (etched pits or holes) when the linear energy transfer (LET) exceeds some critical rate. Neutrons themselves, and gammas, do not provide the minimum LET, while fission fragments do; hence, the fission fragment technique, in which a fissionable foil is placed next to a plastic

detector, and the integrated thermal neutron fluence subsequently determined by etching and microscopic scanning or electronic counting. Typical track sizes of 0.1μ are observed. Becker (Ref. 38) has prepared a review of track etching.

These dosimeters are relatively unaffected by humidity, high temperature, and mechanical shock. Using ^{235}U , detection limits of as low as 10^{-6} rem have been reported for thermal neutrons (Ref. 30); and using other nuclides (^{238}U , ^{232}Th , ^{237}Np) with thresholds in the MeV region, detection limits in the 0.5 to 1 mrem range for fast neutrons are also reported. A discussion by Unruh gives some of the important considerations in applying this technique (Ref. 39). Another discussion by Widell (Ref. 40) indicates that mica is an excellent substance for image formation, but that residual trace uranium in some natural micas produces a high latent background, which must be "erased" before use by baking for several hours at 500°C .

Still another thermal-neutron technique is activation. One approach uses film surrounded by cadmium, with emulsion darkening from the gammas due to the reaction $^{113}\text{Cd}(n,\gamma)^{114}\text{Cd}$ (Ref. 34). Unruh, et al. (Ref. 39) describe a system in which a series of three foils is activated, with subsequent exposure of emulsion by foil decay. The three foils are tin-rhodium (sensitive to gammas, thermal, and "body moderated" fast neutrons), cadmium-rhodium (gammas, "body moderated" fast neutrons), and tin-iron (mainly gammas). The various radiations are determined independently using three film-darkening measurements, and by solving a set of simultaneous algebraic equations.

C. Albedo Neutron Dosimeters

A fundamental consideration in neutron personnel dosimetry is that of backscatter or albedo. We quote from Hankins (Ref. 41):

"When the human body is exposed to neutrons, some of the incident neutrons are backscattered to create a flux of neutrons of various energies leaving the body. These neutrons are called albedo neutrons, and a dosimeter placed on the body to measure this flux of back-scattered neutrons is called an albedo-neutron dosimeter. Such dosimeters are usually designed to detect thermal neutrons, and when cadmium is used to eliminate incident thermal neutrons, the dosimeter response is largely from the thermal neutrons that return from the body."

The power of albedo dosimetry is that it has sensitivity in the energy range where very few other techniques work: for intermediate-

energy neutrons (~ 0.5 eV to ~ 200 keV). There are many situations in which a significant fraction of the dose equivalent is from intermediate-energy neutrons, and in which personnel dosimetry is desired.

Unfortunately, many of the albedo-neutron dosimeters in present use have a significant energy dependence, which decreases their usefulness in situations where the neutron energy spectrum is unknown. Where the spectrum is known, correction factors can be applied to compensate for energy dependence and arrive at a reasonably accurate ($\pm 30\%$) dose-equivalent measurement.

A recent study by Hankins (Ref. 41) examined the way in which various design parameters affect dosimeter response. Hankins used small TLD chips (^6LiF) encased in various thicknesses of 2" diameter polyethylene (CH), all surrounded by cadmium. He studied the effects of placement of the TLD in front of, within, and behind the CH, and the effect of removing the cadmium on the back of the badge. His most important conclusion was that almost none of his parameters significantly affected the rather large energy-dependent sensitivity. On the positive side, he found that adding CH increased the dosimeter sensitivity; that with total encasement of cadmium the sensitivity is not significantly affected by distance away from the body, at least up to 3 cm; that sensitivity is greatly affected by shielding thickness between source and dosimeter; that if cadmium is not present on the back (body side), most of the response is from albedo rather than incident neutrons, except for thermal neutron exposures on a totally cadmium-enclosed dosimeter; and that the placement of both ^6LiF and ^7LiF chips in the same or equivalent positions is critical.

The energy dependence as found by Hankins shows a monotonic decrease of a factor of about 10 between 200 keV and 3 MeV neutron incident energies, with comparably high sensitivities for the intermediate and 200 keV cases.

Similar results have been reported by Hoy (Ref. 42) at Savannah River, where one of the main monitoring problems is around a plutonium production plant, with a typical $^{238}\text{PuF}_4$ neutron spectrum mainly between 0.3 and 25 MeV. The dosimeter (a 2"-diameter two-section concentric hemisphere, separated by cadmium, using ^6LiF and ^7LiF TLD, and worn on a belt) gave adequate dose-equivalent measurements.

Another albedo dosimeter has been developed in the United Kingdom (Ref. 28). TLD's (^6LiF and ^7LiF) are housed in a plastic cover loaded with boron carbide. The design criterion was as follows: for an unshielded 1/velocity ($1/v$) detector at the surface of the body, the response to the albedo from incident thermal

neutrons exceeds that to the albedo from incident 10 keV neutrons by a factor of about 10. The dosimeter was designed to modify this response to decrease the thermal and epithermal sensitivity: hence the boron. The resulting sensitivity is shown in Figure 14 (Ref. 28). Note that the sensitivity is reasonably flat from thermal energies to 10 keV, which is a wide range indeed.

FIGURE 14. Response of albedo dosimeter worn on the body as a function of incident neutron energy (from Ref. 28).

6. SPECIAL DOSIMETRY PROBLEMS

Although the most common monitoring problem in personnel dosimetry is whole-body exposures, there are often requirements for dosimetry in other situations. Perhaps the most familiar of these is measurements of doses to the extremities (particularly to the hands).

For this purpose, several specially designed classes of integrating dosimeters have been developed. Thus, dosimeters to be worn on the fingers are now in common use, and dosimeters for mounting in shoes have also been designed.

Here we will not treat these in detail; instead the reader is referred to the literature. The main operational difficulties with such instruments are related to the validity of the measurement: the accuracy of the particular dosimeter used is by no means the most important consideration. For example, dose may be accrued through contact with radioactive material, a common problem in radiochemistry laboratories. Here even finger dosimeters can give incorrect results in many cases, since low-energy betas can give surface skin doses undetectable by some dosimeters.

The main point to be made here is that users must investigate very carefully the particular radiation field being monitored, to assure that the measured dose is meaningfully related to the dose received.

A discussion of finger dosimetry can be found in Ref. 43 and 44.

7. SUMMARY

The aim of this section has been to discuss instrumentation for the measurement of integrated external radiation doses to personnel. We have attempted to outline the types of measurements needed; the accuracies required; the intrinsic limitations imposed by the sampling problem; and the capabilities and drawbacks of presently existing instrumentation.

We will summarize by reiterating several points made in the main text:

a) The accuracy required of personnel dosimetry systems is not excessively great: the ICRP (Ref. 1) has recommended maximum uncertainties of less than 50% (or less than 1 rem, whichever is greater) in assessment of annual dose equivalent.

b) One key requirement of any dosimetry system is to make sure that no significant component of the radiation field goes completely unmeasured: in cases of large accidental exposure, such an omission would be regrettable indeed.

c) For routine monitoring of external beta/gamma radiations, adequate instrumentation exists today. While improvements are

definitely possible (and in some cases imminent, or at least probable from the current research program in dosimetry), the basic task of beta/gamma dosimetry is now being performed reasonably well.

e) There is still need for a dosimeter useful in those situations dominated by x-rays below about 15 keV.

f) Thermal-neutron dosimetry can now be performed adequately with any of several dosimetry systems.

g) Fast-neutron dosimetry has largely relied for more than a decade upon the nuclear-track-emulsion technique, which suffers from several important drawbacks. Research on a number of possible alternate systems ought to be encouraged.

h) Intermediate-energy neutrons (~0.5 eV to ~200 keV) are still a difficult measurement problem. Albedo dosimeters are not now sufficiently well understood to allow their use in neutron fields with unknown energy spectra, and no other systems now seem capable of performing this task adequately.

i) More work is required to develop a simpler way of assessing the energy spectra of external radiation fields, to complement the measurement of dose-equivalent. For cases of accidental exposure this is particularly important, and has not (in our opinion) been given sufficient attention.

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G. INSTRUMENT NOTES: DOSIMETRY

The following pages contain descriptions of instruments used for monitoring doses received by personnel who are near sources of radiation. In most cases the instrument is to be worn on the person and integrates the exposure received over fairly long time periods. Descriptions are arranged alphabetically by manufacturer and include film badges, thermoluminescent detectors, ionization chambers and chirping GM tubes.

Direct Reading Pen Dosimeters

Models 862, 883, 866, 608, 619, 686

Class	Personal dosimeter																												
Principle of Operation	Ionization chamber, quartz fiber electrometer, capacitor, and microscope																												
Sensitivity and Range	<table><tr><th>Model</th><th>Range</th><th>Number of Divisions</th><th>Smallest Scale Division</th></tr><tr><td>862</td><td>0-200 mR</td><td>20</td><td>10 mR</td></tr><tr><td>883</td><td>0-500 mR</td><td>25</td><td>20 mR</td></tr><tr><td>866</td><td>0-1 R</td><td>20</td><td>50 mR</td></tr><tr><td>608</td><td>0-10 R</td><td>20</td><td>500 mR</td></tr><tr><td>619</td><td>0-100 R</td><td>20</td><td>5 R</td></tr><tr><td>686</td><td>0-600 R</td><td>30</td><td>20 R</td></tr></table>	Model	Range	Number of Divisions	Smallest Scale Division	862	0-200 mR	20	10 mR	883	0-500 mR	25	20 mR	866	0-1 R	20	50 mR	608	0-10 R	20	500 mR	619	0-100 R	20	5 R	686	0-600 R	30	20 R
Model	Range	Number of Divisions	Smallest Scale Division																										
862	0-200 mR	20	10 mR																										
883	0-500 mR	25	20 mR																										
866	0-1 R	20	50 mR																										
608	0-10 R	20	500 mR																										
619	0-100 R	20	5 R																										
686	0-600 R	30	20 R																										
Sampling	Continuous																												
Performance	Calibration accuracy: $\pm 10\%$ @ midscale, ^{60}Co and ^{137}Es Rate dependence: $\pm 5\%$, up to 10,000 R/hr Energy dependence: $\pm 15\%$ of ^{60}Co response from 80 keV to ~1.3 MeV Electrical leakage: 2% per day max. at room temperature Operating temperature range: 29°C to 54°C - (-20°F to 130°F)																												
Requirements	Power: Charger 906 on 1 battery Size: 11.43cmL (4 1/2"); 1.42cm (9/16") diameter Weight: 35.44g (1 1/4 oz.)																												
Features	Hermetically sealed																												
References	Manufacturer's Specifications																												
Cost	Consult factory																												
Address	Dosimeter Corporation of America 6106 Interstate Circle Cincinnati, OH 45242 (513) 793-6051

Special Purpose Dosimeter

Models 886, 884, 609

Class	Personal direct reading dosimeters			
Principle of Operation	Ionization chamber, quartz fiber electrometer, capacitor, microscope.			
Sensitivity and Range	<u>Model</u>	<u>Range</u>	<u>Scale Divisions</u>	<u>Calibration Accuracy</u>
	Neutron-insensitive gamma only, Model 886	0-200 mRad	10 mRad	±10% (γ)
	TE Fast Neutron and Gamma, Model 884	0-200 mRad	10 mRad	±10% γ, ±10%(n)
	Thermal Neutron, Model 609	0-120 mRem	5 mRem	±20%(n)
Sampling	Continuous			
Performance	Rate dependence: ±5%, up to 10,000 R/hr Energy dependence: ±20%, 80 keV to 2 MeV Electrical leakage: 2% per day max. Zero voltage: less than 200 V dc, max. Operating temperature range: -29°C to 54°C (-20°F to 130°F) Altitude: up to 7.620m (25,000 ft) Humidity: up to 98%			
Requirements	Power: charged by external power source, charger Model 806 Size: 1.42cm Dia (9/16"), 11.43cm max. length (4 1/2") Weight:			
Features	Hermetically sealed aluminum wall construction			
References	Manufacturer's Specifications			
Cost	Consult factory			
Address	Dosimeter Corporation of America 6106 Interstate Circle Cincinnati, OH 45242 (513) 793-6051

Personnel Radiation Monitor

Model RT-1A

Class	Personal dosimeters
Principle of Operation	Halogen quenched geiger tube detects exposure rate, aural indications of field strength
Sensitivity and Range	Indications begin at ~0.1 mR/hr and increase proportionally in rate with increasing field. In high fields, individual chirps cease and a continuous tone is emitted. Operation has been tested up to 5000 R/hr. Sensitivity: 150 chirps per milliroentgen (25 chirps per minute in 10 mR/hr field)
Sampling	Continuous
Performance	Operating temperature range: -23°C to 54°C (-10°F to 130°F)
Requirements	Power: 9 V NEDA #1604 battery, ~300 hour life Size: 3.02cm thick x6.03cmW x7.9cmH (1 3/16" x 2 3/8" x3 1/8") Weight: 0.17 kg (6 oz.)
Features	1. Improved resistance to humidity 2. Gasket sealed aluminum case provides rugged, weather resistant package
References	Manufacturer's Specifications
Cost	\$165.00
Address	Eberline Instrument Corporation P.O. Box 2108 Santa Fe, New Mexico 87501 (505) 471-3232

TLD Dosimetry Service

Eberline

Class	Thermoluminescent Dosimetry Service (TLD)
Principle of Operation	TLD dosimeters with various parameters are supplied, returned for processing, readout, and results reported. Readout is by heating and observing thermoluminescence.
Sensitivity and Range	Exposure, mrem: 5 10 50 300 3,000 30,000 Standard Deviation: 20% 10% 4% 4% 4% 4%
Features	(1) Badge has two TLD chips, behind 10 and 285 mg/cm ² , to enable calculation of beta or non-penetrating x-ray dose. (2) Dosimeter contains natural lithium. (3) Wrist, ankle, ring badges also available. (4) Environmental badges (5 LiF chips) available in weatherproof bag. (5) Quarterly exchange is typical. Other cycles available.
References	Manufacturer's Specifications
Cost	Consult manufacturer
Address	Eberline Instrument Corporation P.O. Box 2108 Santa Fe, NM 87501 (505) 471-3232

TLD Reader
Eberline TLR-6

Class	Laboratory
Principle of Operation	Thermoluminescent materials are heated in the heater panel and the light released from the material, which is proportional to the dose received, is integrated and displayed on the front panel digital readout.
Sensitivity and Range	1 mR to 10^5 R in two ranges: 1 mR to 100 R 1 R to 10^5 R
Sampling	Each TLD requires 10 second pre-heat and 10 second read cycle
Requirements	Power: ac line operation Size: (30H x25W x45cmD) 12" x10" x18" Weight: (16 kg) 36 lb
Features	(1) 5-digit display (2) Glow curve signal available on rear banana jack, 0 to 10 mV (3) Background less than one mR, referred to 35 mg LiF (4) All solid-state circuitry
References	Manufacturer's Specifications
Cost	\$4400.00
Address	Eberline Instrument Corporation P.O. Box 2108 Santa Fe, NM 87501 (505) 471-3232

TLD Materials

Class	Detector
Principle of Operation	After the TLD has been exposed to radiation, it can be heated and it will luminesce. By proper calibration, the amount of light output is proportional to the radiation intensity.
	<u>Materials Available</u>
	(1) <u>Lithium Fluoride (TLD) as -</u> -Powder - TLD-100, 600, 700 -Extrusions - TLD-100, 600, 700 in several geometries -High Sensitivity Ribbons - TLD-100, 600, 700 in several geometries -Cleaved Crystals - TLD-100, 600, 700 in several geometries
	(2) <u>Calcium Fluoride as -</u> -CaF ₂ :Dy crystals in several geometries -CaF ₂ :Dy powder -CaF ₂ :Mn powder -CaF ₂ :Mn chips in several geometries
	(3) <u>Lithium Borate as -</u> -Li ₂ B ₄ O ₇ :Mn powder -Li ₂ B ₄ O ₇ :Mn solid dosimeters in several geometries
	(4) <u>Calcium Sulphate as -</u> -CaSO ₄ :Mn powder -CaSO ₄ :Dy powder
	(5) <u>TLD Configurations including</u> -Bulb Dosimeters -Card Dosimeters
References	Manufacturer's Specifications
Cost	Consult factory
Address	The Harshaw Chemical Company 6801 Cochran Road Solon, OH 44139 (216) 248-7400

Specifications:

Property/Type	LiF	Li ₂ B ₄ O ₇ :Mn	CaF ₂ (TLD-200)	CaSO ₄ :Mn ^{1,2}	CaSO ₄ :Mn ^{1,2}	CaSO ₄ :Dy
Density (gm/cc)	2.64 (powder ~1.3)	~2.4 (Ribbons) ~1.2 (Powder)	3.18	3.18	2.61	2.61
Effective Atomic No. (Z) for photoelectric absorption	8.2	7.4	16.3	16.3	15.3	15.5
TL Emission Spectra	3500-6000 Å (4000 max.)	5300-6300 Å (6050 max.)	4400-6000 Å (5000 max.)	Peaks at 4835 Å at 5765 Å	4500-6000 Å	4800 Å 5700 Å
Temperature of main TL glow peak	195°C	200°C	260°C	180°C	110°C	220°C
Efficiency at ⁶⁰ Co relative to LiF	1.0	0.15	10	30	70	20
Energy Response 30 keV/ ⁶⁰ Co	1.25	0.9	~13	~12.5	~10	~12.5
Useful Range	mR-3x10 ⁵ R	< 50mR-10 ⁶ + R	mR-3x10 ⁵ R	10μR-10 ⁶ R	μR-10 ⁴ R	μR-~10 ⁵ R
Fading	Negligible* 5%/yr at 20°C	5% in 3 months	10% in first 24 hours 15% total in 2 weeks	10% in first 24 hours 16% total in 2 weeks	50% in first 24 hours	2% in 1 month 8% in 6 months
Physical Forms	Powder ex- truded teflon, glass encap- sulated bulbs, cleaved cry- stals	Powder, Solid Chips, Teflon	Powder, teflon, pressed, glass encapsulated bulbs	Powder, Crystals, Glass bulbs	Powder, Teflon	Powder, Pressed Ribbons

*Post-irradiation, pre-evaluation anneal for 10 minutes with LiF and 20 minutes with CaF₂(TLD-200) at 100°C normalizes these materials and eliminates fading. Other TLD materials include BeO:Mn; CaSO₄ (rare earth), Al₂O₃(Mn), SrF₂(Mn), CaSO₄(Tm), Mg₂SiO₄(Tb).

¹Harshaw CaSO₄ is 80-200 mesh and dispensable; Con-Rad material very fine.

²The high sensitivity TLD materials are all best handled, used, and stored in opaque containers to prevent fading from light exposure.

TLD Dosimetry System

Harshaw Model 2271

Class	Laboratory																										
Principle of Operation	Thermoluminescent Dosimetry (TLD) system. TLD chips are read out by heating which generates luminescence, detected by a phototube. System is automated. TLD chips are lithium fluoride (LiF).																										
Sensitivity and Range	<u>Exposure (Dose) Range</u> <table><tr><td>Gamma or X-rays</td><td>10mR to 10^5 Rad</td></tr><tr><td>Beta</td><td>40mRad to 10^5 Rad</td></tr><tr><td>Thermal Neutrons</td><td>5mRem to 10^5 Rem</td></tr><tr><td>Environmental</td><td>> 1mRad</td></tr></table> <u>Detectability (No background, 90% confidence level)</u> <table><tr><td>$^{60}\text{Cobalt}$ Gamma</td><td>10mR $\pm$ 5mR</td></tr><tr><td>Natural Uranium Beta</td><td>40mRad $\pm$ 20 mRad</td></tr><tr><td>Thermal Neutron</td><td>5mRem $\pm$ 2.5 mRem</td></tr><tr><td>Environmental</td><td>1mR $\pm$ 0.5mR</td></tr></table>	Gamma or X-rays	10mR to 10^5 Rad	Beta	40mRad to 10^5 Rad	Thermal Neutrons	5mRem to 10^5 Rem	Environmental	> 1mRad	$^{60}\text{Cobalt}$ Gamma	10mR $\pm$ 5mR	Natural Uranium Beta	40mRad $\pm$ 20 mRad	Thermal Neutron	5mRem $\pm$ 2.5 mRem	Environmental	1mR $\pm$ 0.5mR										
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Thermal Neutron	5mRem $\pm$ 2.5 mRem																										
Environmental	1mR $\pm$ 0.5mR																										
Sampling	TLD card containing 2 chips can be read out in 40 seconds																										
Performance	<u>Card Configurations</u> <table><tr><td>Type G-1 Gamma (No Neutrons)</td><td>2 TLD-100, 1/8" x 1/8" x0.035"</td></tr><tr><td>Type G-7 Gamma (Neutrons)</td><td>2 TLD-700, 1/8" x 1/8" x0.035"</td></tr><tr><td>Type BG-1 Beta/Gamma (No Neutrons)</td><td>2 TLD-100, 1/8" x 1/8" x0.035"</td></tr><tr><td>Type BG-7 Beta/Gamma (Neutrons)</td><td>2 TLD-700, 1/8" x 1/8" 1/8" x 1/8" x0.035"</td></tr><tr><td>Type NG-67 Neutron/Gamma</td><td>One each TLD-600 and TLD-700, both are 1/8" x 1/8" x0.035"</td></tr><tr><td>Type G-2 Gamma (Environmental)</td><td>2 TLD-200 ($\text{CaF}_2:\text{Dy}$) 1/8" x 1/8" x0.035"</td></tr><tr><td>Type K-8 Gamma (kilorad levels)</td><td>2 TLD-800 ($\text{Li}_2\text{B}_4\text{O}_7:\text{Mn}$) 1/8" x 1/8" x0.035"</td></tr></table> <u>Uniformity of Response</u> <table><tr><td>$^{60}\text{Cobalt}$ Gamma</td><td>$\pm 15\%$ at 100mR</td></tr><tr><td>Natural Uranium Beta</td><td>$\pm 20\%$ at 100mRad</td></tr><tr><td>Thermal Neutron</td><td>$\pm 25\%$ at 100mRad</td></tr></table> <u>Maximum Error (90% confidence level)</u> <table><tr><td>$^{60}\text{Cobalt}$ Gamma</td><td>$\pm 7\text{mR}$ or $\pm 15\%$ whichever is greater</td></tr><tr><td>Natural Uranium Beta</td><td>$\pm 20\text{mRad}$ or $\pm 20\%$ whichever is greater</td></tr><tr><td>Thermal Neutrons</td><td>$\pm 2\text{mRem}$ or $\pm 25\%$ whichever is greater</td></tr></table>	Type G-1 Gamma (No Neutrons)	2 TLD-100, 1/8" x 1/8" x0.035"	Type G-7 Gamma (Neutrons)	2 TLD-700, 1/8" x 1/8" x0.035"	Type BG-1 Beta/Gamma (No Neutrons)	2 TLD-100, 1/8" x 1/8" x0.035"	Type BG-7 Beta/Gamma (Neutrons)	2 TLD-700, 1/8" x 1/8" 1/8" x 1/8" x0.035"	Type NG-67 Neutron/Gamma	One each TLD-600 and TLD-700, both are 1/8" x 1/8" x0.035"	Type G-2 Gamma (Environmental)	2 TLD-200 ($\text{CaF}_2:\text{Dy}$) 1/8" x 1/8" x0.035"	Type K-8 Gamma (kilorad levels)	2 TLD-800 ($\text{Li}_2\text{B}_4\text{O}_7:\text{Mn}$) 1/8" x 1/8" x0.035"	$^{60}\text{Cobalt}$ Gamma	$\pm 15\%$ at 100mR	Natural Uranium Beta	$\pm 20\%$ at 100mRad	Thermal Neutron	$\pm 25\%$ at 100mRad	$^{60}\text{Cobalt}$ Gamma	$\pm 7\text{mR}$ or $\pm 15\%$ whichever is greater	Natural Uranium Beta	$\pm 20\text{mRad}$ or $\pm 20\%$ whichever is greater	Thermal Neutrons	$\pm 2\text{mRem}$ or $\pm 25\%$ whichever is greater
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Residual TL Effect (Following
a 100mR evaluation)

Equivalent Gamma	5mR
Equivalent Beta	10mRad

Reproducibility (Repeatability)

Same card with 100mR ^{60}Co	$\pm 5\%$
Same card with 100mRad U_{nat}	$\pm 10\%$
Reference Light Source	$\pm 3\%$

Evaluation Time (Two TLD's
per Card)

Both TLD's	< 40 seconds per card
One TLD	< 30 seconds per card

Features

- (1) System can accommodate 250 TLD cards (2 chips each) automatically
- (2) BCD identification of each card also automated
- (3) Manual mode also available
- (4) TLD reader, integrating picoammeter, flow regulator, flow meter, teletypewriter all included.
- (5) Other systems are available; 3000, 2000, 2000C, 2000P. They can evaluate powders, solids, or bulk.

Reference Manufacturer's Specifications

Cost Refer to manufacturer

Address Harshaw Chemical Company
6801 Cochran Road
Solon, OH 44139
(216) 248-7400

Film Badge Service

Landauer Gardray

Class	Personnel Dosimetry Service
Principle of Operation	Uses film with two separate emulsions for low and high dose readings of x-ray, beta, and gamma exposures, readout by Landauer after mail-exchange.
Sensitivity and Range	Sensitivity: Exposure range, 10mR to 10 ⁵ R TLD Energy range: 7 1/2 KVP (3 to 4 keV) to 1.33 MeV 700R same TLD Energy dependence: 15 keV to 1.3 MeV Dose rate dependence: None measured
Sampling	Continuous
Performance	Accuracy: University of Michigan: National Sanitation Foundation Ann Arbor, Michigan
Features	1. Lower energy beta detection possible with less wrapper mass (~ 25 mg/cm ²) 2. Fast neutron and thermal monitoring also available. 3. Vapor-sealed film strip insures freshness, improves low dose measurements and minimizes environmental effects.
References	Manufacturer's Specifications

Cost	Number of Badges per shipment	G* Regular (x-ray - beta-gamma)		
		Weekly	Biweekly	Monthly
	1	1.80	2.00	2.25
	2	1.20	1.30	1.45
	3	1.00	1.05	1.10
	4 or 5	0.85	0.90	1.00
	6 - 11	0.75	0.80	0.85
	12 - 30	0.70	0.75	0.80
	31 - 60	0.60	0.65	0.75
	61 -100	0.55	0.60	0.65

Number of Badges per shipment	p* Fast Neutron (plus x-ray, γ , β)	
	Weekly	Biweekly
1	2.75	2.85
2	2.05	2.20
3	1.80	1.85
4 or 5	1.65	1.70
6 - 11	1.55	1.60
12 - 30	1.50	1.55
31 - 60	1.35	1.40
61 -100	1.25	1.30

Address R. S. Landauer Jr. and Co.
Glenwood Science Park
Glenwood, IL 60425
(312) 755-7000

*Price each badge, each time packet is exchanged.

TLD Ring Landauer Dosimetry Service

Class	TLD Ring Dosimetry Service			
Principle of Operation	LiF (TLD) dosimeters on ring holder readout by Landauer after mail-exchange			
Sensitivity and Range	Sensitivity: Exposure range: 10mR to 100,000R Energy Range: Energy independent above 100 keV <40% energy dependence for lower energies down to 18 keV			
Performance	Accuracy: Standard error < 25% (10mR), < 10% (> 100mR) Tissue Equivalence: $Z(\text{LiF}) \div Z(\text{Tissue}) = 1.096$			
Features	1. Weekly, biweekly, monthly schedules 2. Dosimeters can be furnished sealed in sterile tubing.			
References	Manufacturer's Specifications			
Cost	Number of Badges per shipment	U* TLD Ring (X-ray - beta - gamma) <u>Weekly</u> <u>Biweekly</u> <u>Monthly</u>		
	1	2.35	2.40	2.75
	2	1.75	1.80	1.90
	3	1.55	1.60	1.65
	4 or 5	1.40	1.45	1.50
	6 - 11	1.30	1.35	1.40
	12 - 30	1.25	1.30	1.35
	31 - 60	1.20	1.25	1.30
	61 -100	1.10	1.15	1.20
Address	R. S. Landauer Jr. & Co. Glenwood Science Park Glenwood, IL 60425 (312) 755-7000			

*Price each badge, each time packet is exchanged.

X-Ray Dosimeter

Nuclear Associates Model 06-002, 012, 050, 099

Low Energy
Direct Reading
Dosimeter

Class	Portable, Pocket										
Principle of Operation	0.03" Plastic walls allow low energy x rays to penetrate the ion chamber.										
Sensitivity and Range	<table><thead><tr><th><u>Model</u></th><th><u>Range</u></th></tr></thead><tbody><tr><td>06-002</td><td>0-200 mR</td></tr><tr><td>06-012</td><td>0-1.5 R</td></tr><tr><td>06-050</td><td>0-5 R</td></tr><tr><td>06-099</td><td>0-100 R</td></tr></tbody></table>	<u>Model</u>	<u>Range</u>	06-002	0-200 mR	06-012	0-1.5 R	06-050	0-5 R	06-099	0-100 R
<u>Model</u>	<u>Range</u>										
06-002	0-200 mR										
06-012	0-1.5 R										
06-050	0-5 R										
06-099	0-100 R										
Sampling	Continuous										
Performance	Accuracy: $\pm 15\%$ of true dose for 30 keV X-rays Energy Dependence: $\pm 10\%$ of 30 keV reading for 15 to 100 keV $\pm 20\%/-10\%$ of 30 keV reading for 30 to 225 keV Leakage: 2% of full scale/24 hr										
Requirements	Power: Use charger Size: Four in. length x 9.16 in. diameter Weight: About 2 oz.										
References	Manufacturer's Specifications										
Cost	\$86.25 - except 06-012 \$100.00										
Address	Nuclear Associates 100 Voice Road Carle Place, N.Y. 11514 (516) 741-6360

Direct Reading Dosimeters

Nuclear Associates

Class

Portable, Pocket, Fountain Pen Size

Principle of
Operation

A quartz fiber electrometer, compound microscope calibrated
readout.

Sensitivity
and Range

Energy Sensitivity:

Model	Range	Cost 1-9 each
06-869	200 mr	\$60.00
883	500 mr	56.50
866	1 r	71.25
608	10 r	71.25
638	200 r	71.25
686	600 r	71.25

INSTRUMENTATION
FOR ENVIRONMENTAL
MONITORING

RAD-DOS
Gamma, X-Ray
Nuclear Associates 2
Page 2

Sampling	Continuous	
Performance	Accuracy: $\pm 10\%$ of dose Leakage: 2% per 24 hours	
Requirements	Power Use charger Size: 4" L x 9/15" D Weight: About 2 ounces	
Features	Optional Charger, Model 06-912 Optional Calibrator, Model 06-200	\$79.50 42.50
References	Manufacturer's Specifications	
Cost	See chart above	
Address	Nuclear Associates 100 Voice Road Carle Place, N.Y. 11514 (516) 741-6360

Personal Digital Dosimeter
Nuclear Associates Model 06-502

Class	Portable - Belt Clip
Principle of Operation	Halogen-quenched GM Tube and digital display
Sensitivity and Range	Range 0-999 mR - Digital Display 43 beeps per mR
Sampling	Continuous
Performance	Accuracy: $\pm 10\%$ with Co ⁶⁰ at 1 R/hr
Requirements	Power: 9 volt transistor battery Size: 6.35cmx11.43cmx2.54cm (2 1/2"x4 1/2"x1") Weight: 170gms (6 oz.)
Features	Wide Range 0-999 mr
References	Manufacturer's specifications
Cost	\$190.00
Address	Nuclear Associates 100 Voice Road Carle Place N.Y. 11514 (516) 741-6360

Rateometer
Prima I, II

Model	Prima I 05-100	Prima IIa 05-205
Radiation Detected	Gamma, X	Gamma, X
Detector	G.M. Tube	G.M. Tube
Energy Dependence	200 keV - 1 MeV	80 keV - 1 MeV
Ranges	Background to 2 R/hr	Background to 10 R//hr
Response Time	Not applicable	
Accuracy	±30%	±30%
Features	Audible Output	Audible Output
Power Required	4.05V mercury cell, 6 mo.	9V alkaline cell, 6 mo.

Size	2 cm D x 13 cm (0.75" D x 5" L)	6.5cmx9cmx2. (2.5"x3.6"x0.84")
Weight	100 grams (3.5 ounces)	120 grams (5 ounces)
Price	\$450.00	\$145.00

References

Manufacturer's specifications

Address

Nuclear Associates
100 Voice Road
Carle Place, N.Y. 11514
(516) 741-6360

Primadose II - Alarming Dosimeter

Nuclear Associates Model 05-153

Class	Portable Belt Clip
Principle of Operation	Compensated Halogen-quenched GM Tube with readout separately.
Sensitivity	Exposure Alarm Set Points 12, 25, 50, 100, 200, 400 or 800 mR High Dose Rate Alarm 10-20 R/hr
Sampling	Continuous
Performance	Accuracy: $\pm 10\%$ at Cs-137 Energy Response: $\pm 20\%$ from 80 keV to 1.2 MeV Operating Environment: 0°C to 55°C at 0-90% relative humidity
Requirements	Power: 9 Volt Alkaline Battery Size: 9.2 cm x 6.4 cm x 2.1 cm (3.6" x 2.5" x .84") Weight: 120 gm (4 oz)

INSTRUMENTATION
FOR ENVIRONMENTAL
MONITORING

RAD-DOS
Nuclear Associates 5
Page 2

Features	Wide Range Optional Reader, Model 95-154, Cost: \$395.00
References	Manufacturer's specifications
Cost	\$225.00
Address	Nuclear Associates 100 Voice Road Carle Place, N.Y. 11514 (516) 741-6360

X-Ray Dosimeter

Nucleus Model 06-002, 050, 099
(Bendix)

06-050
Low-Energy Dosimeter

Class	Portable, pocket								
Principle of Operation	Plastic walls allow low energy x-rays to penetrate the ion chamber								
Sensitivity and Range	<table><tr><th>Model</th><th>Range</th></tr><tr><td>06-002</td><td>0-200 mR</td></tr><tr><td>050</td><td>0-5 R</td></tr><tr><td>099</td><td>0-100 R</td></tr></table>	Model	Range	06-002	0-200 mR	050	0-5 R	099	0-100 R
Model	Range								
06-002	0-200 mR								
050	0-5 R								
099	0-100 R								
Sampling	Continuous								
Performance	Accuracy: $\pm 15\%$ of true dose for 30 keV x-rays $\pm 10\%$ of keV reading for 15 to 100 keV $\pm 20\%$ -10% of 30 keV reading for 30 to 225 keV Leakage: 2% of full scale/24 hours								
Requirements	Power: Use charger Size: 4" in length Weight: About 2 ounces								
Cost	\$82.00								
Address	Nucleus, Inc. P.O. Box R Oak Ridge, Tenn. 37830 (615) 482-4041

Direct Reading Dosimeter
Nucleus Model 06-884, 886

Class	Portable, pocket
Principle of Operation	Tissue equivalent dosimeter, measures both fast neutron and gamma dose in rads, based on collisions between fast neutrons and atoms of soft tissue with the empirical formula $C_5H_4O_{18}N$.
Sensitivity and Range	Neutron energies: 20 keV to 15 MeV Exposure: 20 mrad to 600 rads Gamma energies: 80 keV to 3 MeV.
Sampling	Continuous
Performance	Accuracy: 20% of actual dose for neutrons 10% of actual dose for Cobalt-60 and Cesium-137 gammas 15% of actual dose for gammas from 80 keV to 3 MeV Leakage: Unavailable
Requirements	Power: Use charger Size: 4 inches, length Weight: 2 ounces
Features	To measure the first-collision dose of fast neutrons, subtract the neutron insensitive dosimeter reading (Model: 06-886, \$115.00) from the tissue equivalent dosimeter reading
Cost	\$160.00 each
Address	Nucleus, Inc. P.O. Box R Oak Ridge, Tenn. 37830 (615) 482-4041

Film Badge Dosimetry Service
Radiation Detection Company

BETA RAY SENSITIVITY OF FRONT FILM
IN R-D FILM PACKET

FILM DENSITIES UNDER R-D BADGE
FILTERS FOR 0.3R EXPOSURE

Class	Film Badge Dosimetry Service		
Principle of Operation	Film is distributed to user, developed after exposure, exposure reading reported. Film is sensitive to gamma, x-ray, and beta radiation, with four filters.		
Sensitivity and Range	<u>Radiation</u>	<u>Energy Range</u>	<u>Exposure or Dose Equivalent</u>
	X, Gamma	25 keV to 70 keV	0.005 to 50 R
		70 keV to 3 MeV	0.01 to 1000 R
	Beta	Above 1 MeV	0.01 to 1000 rem
	Filters Open Window Region: 0.125" plastic 0.032" aluminum: 0.015" cadmium plus aluminum 0.030" lead plus aluminum		
Features	1. Films cycle weekly, biweekly, monthly 2. Reports mailed to user within three days of receipt of film 3. Ring dosimeters also available 4. Neutron dosimeters using several activation foils also available		
References	Manufacturer's Specifications		
Cost	Upon request		
Address	Radiation Detection Company 162 Wolfe Road, Sunnyvale, CA 94086 (408) 735-8700

TLD Reader

Radiation Detection Mark IV Model 1100

Class	Laboratory
Principle of Operation	TLD reader for thermoluminescent dosimeters. Accepts a wide variety of powdered or solid TLD's, mainly designed for LiF. Constant voltage heater supply, chopper stabilized operational amplifier.
Sensitivity and Range	Sensitivity: 10 mR to 50,000 R for LiF
Performance	Accuracy: 10 mR to 100 mR better than ± 10 mR 100 mR to 1 R better than $\pm 10\%$ Above 1 R better than $\pm 5\%$ Readout Time: 10 to 15 seconds
Requirements	Power: 115 VAC, 60 cycles Size: 12.4" height, 14.1" width, 18.6" depth (31 x 36 x 47 cm) Weight: 42 lb (19 kg)
Features	1. Removable planchet for powder samples 2. Internal reference light 3. Provision for recording glow curves 4. Inert gas flow to suppress nonradiation-induced luminescence
References	Manufacturer's Specifications
Cost	
Address	Radiation Detection Company 162 Wolfe Road, Sunnyvale, CA 94088 (415) 967-2837

TL Badge System
RMC TL Badge: Model UD-800 Series
Automatic Reader: Model UD-710 Series
Magazine Changer: Model UD-730 Series

Construction of the model UD-802A, a typical TL badge, and the energy responses of each element.

Class	Laboratory System
Principle of Operation	Thermoluminescent Dosimetry (TLD) system. TLD chips that integrate an exposure to gamma and x-rays are readout by heating which generates luminescence. This released luminescence is detected by a phototube. System is automated. TLD chips are $\text{Li}_2\text{B}_4\text{O}_7: \text{Cu}$ and $\text{CaSO}_4: \text{Tm}$
Sensitivity and Range	Radiation detected: X, γ -rays 10 keV to ~ 10 MeV Measurement range: 1 mR - ~ 1000 R
Sampling	Periodical readout of integrated skin dose on TLD chips that are number coded.

Performance

Fading

Elements (phosphors):	Li ₂ B ₄ O ₇ : Cu	< 10% 1 month
	CaSO ₄ : Tm	< 1% 1 month
Spurious signal:	Undetectably small	
Badge-feed system:	With magazines, 50 badged magazine	
Reading method:	Heating	
	Optical-heating method	
TL measurement:	Pulse counting method (for small signals); current method (for large signals)	
Preheating and annealing:	Functions incorporated	
ID Code Reading:	Optical reading	
Processing speed:	20 seconds/badge	
	3 hours/500 badges	
	(utilizing magazine changer)	

Requirements

Power:	115 ± 10% VAC, 220 ± 10% VAC, 50-60 Hz, 500VA
Size:	TLD chip, 3mm dia.x15mg/cm ² density; on hanger, 49x23x6mm
	TLD automatic Reader, Model UD-710 Series, 543mmWx830mmHx360mmD
	Magazine changer, Model UD-730 Series, 485mmWx790mmHx360mmD
Weight:	7g, 13g with hanger (badge)
	Model UD 701 Series, 77 kg (156 lbs)
	Model UD 730 Series, 61 kg (123 lbs)

Features

- Easy handling
- Resistance to mechanical shock
- Low-energy x-rays and γ-rays can be detected separately by utilizing the energy characterization of the phosphor.
- Equipped with an interface for transmitting data to outside equipment such as computer or printers
- ID codes of large capacity can be punched
- Lock mechanism prevents accidental contamination of detection port
- Self-checking function such as checking of optical path, sensitivity of PM tube, heating stability, etc.
- Microcomputer control.

References

Manufacturer's Specifications

Cost

Address

Radiation Management Corporation
3508 Market Street
Philadelphia, PA 19104
(215) 243-2950

Alarming Digital Dosimeter

PDR-1b and PDR-1c

Class	Portable
Principle of Operation	GM tube, audio output (chirp) and digital readout indicates dose accumulated--settable alarms
Sensitivity and Range	One chirp each mR--PDR-1b One chirp each tenth mR--PDR-1c Digital dose from 1 mR to 9999 mR--PDR-1b Digital dose from .1 mR to 999.9 mR--PDR-1c
Sampling	Continuous
Requirements	Power: 4 nicad miniature rechargeable batteries--with charger Size: 2.25" x0.813" x4.25" Weight: 7 oz.
Features	Nicad rechargeable batteries Can be run off A.C. Memory protection--bias cells take over to protect memory in event of discharge or failure of main batteries Settable alarms
References	Manufacturer's Specifications
Cost	PDR-1b: \$275.00 PDR-1c: \$285.00
Address	Technical Associates 7051 Eton Avenue Canoga Park, CA 91303 (213) 883-7043

Pocket Radiation Monitor

Model PM-1

Class	Portable
Principle of Operation	GM tube, audio output (chirps) indicates dose rate
Sensitivity and Range	Sensitivity: 3 chirps per minute per mR/hr
Sampling	Continuous
Performance	Accuracy: Operating temperature range: -20°C to +55°C
Requirements	Power: Single "transistor" battery (9V) Size: 6cm x 11cm x 2.7cm (2.4" x 4.3" x 1.1") Weight: 0.2 kg (0.44 lb)
Features	Rugged Long battery life
References	Manufacturer's Specifications
Cost	\$100.00
Address	Technical Associates 7051 Eton Ave. Canoga Park, CA 91303 (213) 883-7053

TLD Dosimetry System and Badge Service
Teledyne Radi-Guard and Model 8300

Multi area dosimeter
4 main readout areas
(1 2 3 4) and 4 backup
areas (1a 2a 3a 4a)

Rear view of dosimeter with
identification numbers 2
dots insure dosimeter is
properly inserted in reader

Standard personnel
badge filter array

RADI-GUARD DOSIMETER

Model 8300 Reader

Class	Thermoluminescent Dosimetry (TLD) System
Principle of Operation	TLD dosimeters with various parameters are read out by heating and observing thermoluminescence. Badge service is provided. Standard Badge (see figure) has following filters: <ol style="list-style-type: none"> 1. 2 mm plastic + 1 mm Al 2. open 3. 1 mm plastic + 1 mm Cu + 1 mm Al 4. 1 mm plastic + 1 mm Pb + 1 mm Al
Sensitivity and Range	<p>(a) <u>Personnel</u></p> <p>With ^{137}Cs exposure, maximum error (90%) = ± 8 mR or $\pm 10\%$, whichever is greater. Dosimeter is 15% loaded LiF.</p> <p>(b) <u>Environmental</u></p> <p>Standard dosimeter is $\text{CaSO}_4 \cdot \text{Dy}$, 30% loaded.</p>
Features	<p>(a) <u>Model 8300 Reader</u></p> <ol style="list-style-type: none"> (1) Integrator linear to $\pm 1.5\%$ over entire range (2) 6-decade display

Features (cont.)

- (3) Glow curve output (0 to 8 volts) provided, with automatic range changing
- (4) Inert gas flow is used
- (5) Built-in annealing feature with on-off switch
- (6) Fast calibration (2 seconds)
- (7) 4 programmed temperature cycles available, cycle times 12 to 18 sec
- (8) Switch can enable reading of LiF, Ca:SO₄:Dy, and finger dosimeters
- (9) Model 9100 automatic system is compatible

(b) TLD Dosimeters

- (1) Rods (1 mm diameter, 6 to 150 mm long) available as ⁶LiF, ⁷LiF, CaSO₄:Mn, Li₂B₄O₇:Mn
- (2) Tape (25 mm wide, 0.4 mm thick) available in ⁷LiF
- (3) Discs of various sizes available in ⁷LiF, ⁶LiF, Li₂B₄O₇:Mn, CaF₂:Mn, CaSO₄:Mn

(c) Dosimetry Service

- (1) Issue period: 1, 2, 4 weeks, 1 month or 3 months
- (2) Radi-Guard dosimeters (see figure) are used
- (3) Calibration is done with ¹³⁷Cs
- (4) Zero control badges supplied free with each shipment

References

1) Manufacturer's Specifications

Cost

Model 8300 Reader \$7,990.00
(Other models, automatic and manual, are also available).

Dosimeter Readout Service:

Badges per Service Shipment	Regular Dosimeter	Thermal Neutron Dosimeter	Quarterly Summary AEC Form 5
1-9	\$1.70	\$2.50	\$1.00
10-24	1.20	2.25	.75
25-99	.95	2.00	.50
100 & up	.85	1.75	.25

Initial Set-up or new name charge: \$.75 each
Lost badge charge 2.50 each
Minimum billing per quarter 30.00

Address

Teledyne Isotopes
50 Van Buren Avenue
Westwood, N.J. 07675
(201) 664-7070

Direct-Reading Dosimeters

Victoreen Model 541R/541L

Class	Portable
Principle of Operation	Pocket ion chamber directly read on built in scale Radiation detected: gamma, x-ray Energy response: 541R: 30 keV to 2 MeV 541L: 17 keV to 200 keV (see figure)
Sensitivity and Range	Energy sensitivity: $\pm 10\%$ from 55 keV to 2 MeV, gamma $\pm 15\%$ from 45 keV to 2 MeV, gamma Range: 0 - 200 mR Geotropism: less than 1 minor scale division
Performance	Leakage rate: 2% per 24 hours Precision: within 10% of full scale Environmental effects: temperature limits -10°C to 50°C humidity limit: 0-99%, noncondensing Typical energy response: see figure

Requirements	Power: Model 2000A charger Size: 10.2cmL, 1.27cmD (4"L, 1/2"D) Weight: 0.014 kg (0.5 oz.)
Features	1) Uses Model 2000A charger 2) Meets ANSI N13.5 - 1972
Cost	541R - \$60.00, 541L - \$85.00
Remarks	The mR scale is built into the dosimeter and is illuminated by an external light source.
Address	Victoreen, Inc. 10101 Woodland Avenue Cleveland, Ohio 44104 (216) 795-8200

Personal Dosimeter Reader/Charger

Victoreen Model 687D Minometer II

Class	Laboratory		
Principle of Operation	Radiation penetrating a pocket chamber capacitor causes current to leak off in proportion to the amount of radiation. The resulting voltage drop is calibrated in milliroentgens and is read on the Minometer II.		
Sensitivity and Range	Full scale ranges: 0-40 mR, 0-200 mR, 0-100, 0-500 mR depending upon dosimeter used		
Performance	Instrument precision: Within 10% of full scale at standard temperature. Operating temperature: +10°C to +35°C Display: 10.2cm(4") meter.		
Dosimeters	Model	362 (0-200mR)	365 (0-500mR)
	Radiation Detected	Gamma & X-ray	Gamma & X-ray
	Gamma Energy Dependence	within $\pm 15\%$ from 30 keV to 2 MeV	within $\pm 15\%$ from 40 keV to 2 MeV
	Cs ¹³⁷ Calibration Leakage	$\pm 15\%$ less than 5%/day	$\pm 15\%$ less than 5%/day
	Range	0 - 200mR	0 - 500mR
	Overall Dimensions	14cmL, 1.27cmDia (5-1/2"L, 1/2"Dia)	14cmL, 1.27cmDia (5-1/2"L, 1/2"Dia)
	Temperature Dependence:	within 0.16%/°C	Same
	Humidity limits:	0-60% normal, up to 80% for limited time	Same

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Beta, Gamma, X-ray
Victoreen 2
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Weight	0.014 kg (0.5 oz.)	0.014 kg (0.5 oz)
Price	\$18.00	\$18.00

Requirements Power: 100-130 VAC, 50 or 60 Hz, 15 VA
 Size: 27.3cmLx19.1cmWx13.3cmH (10.75"Lx7.5"Wx5.25"H)
 Weight: 4.1 kg (9 lbs.)

Features Reader and charger for pocket chambers

Cost \$970

Address Victoreen Inc.
 10101 Woodland Avenue
 Cleveland, Ohio 44104
 (216) 795-8200

Thermoluminescent Dosimeters

LiF and CaF₂:Mn

Class	Detectors
Principle of Operation	Based upon principle of thermoluminescence whereby radiation dose is related to light output as material (LiF or CaF ₂ :Mn) is heated.
Sampling	Integrated dose until moment of readout
Specifications	See page 2
Requirements	TLD reader to readout dose accumulated Power: } Size: } relates to TLD reader Weight: }
Features	TLD readers are also available from Victoreen
Remarks: LiF:	<p>Lithium Fluoride is supralinear above approx. 400 R See Figure. Gas Flow required below 400 mR.</p> <p>LiF sensitivity depends upon previous exposure history and the annealing technique used. For short term measurement cycles (within 1 week), the recommended anneal cycle consists of: a high temperature dedose, a controlled cooling or quench plus a post irradiation anneal to empty unstable low temperature traps. Victoreen TLD Readers include automatic programming of this anneal cycle.</p>
References	Manufacturer's Specifications
Cost	Contact Manufacturer
Address	Victoreen, Inc. 10101 Woodland Avenue Cleveland, OH 44104 (216) 795-8200

Specifications:
LiF:

Model Numbers			Configuration	Dose Range in		Dose Rate Dependence R/Sec.	Size in mm
LiF	⁶ LiF	⁷ LiF		Model 2800	Model 2810		
2600-6	2600-7	2600-8	Mini	1mR- 15,000R	1mR-10 ⁵ R	Negligible to 10 ⁹	12x1.4
	2600-54	2600-55	Chip	1mR-	<1mR-		3.2x3.2x0.9
2600-70			Chip	3,000R	10 ⁵ R	<10% to 10 ¹¹	0.7x4.5
2600-58	2600-56	2600-57	Square Rod				6x1x1
4.8x10 ⁻⁸	7.0x10 ⁻⁸	9.0x10 ⁻¹¹	(Thermal neutron response)R/n/cm ² : ⁶⁰ Co Rads				
2.0x10 ⁻¹⁰	2.0x10 ⁻¹⁰	2.0x10 ⁻¹⁰	(Fast neutron response)PoBeR/n/cm ²				

CaF₂:Mn:

Model	Configuration	Dose Range		Dose Rate Dependence R/Sec.	Linearity	Size in mm
		Model 2800	Model 2810			
2600-12	Mini-Round Glass tube	< 10mR- 6,000 R	< 10mR- 100,000 R	Negligible to 10 ⁹	5%-10 ⁵ R	12x1.4
2600-14	Chip					3.2x3.2x0.9
2600-45	Bulb	< 5mR- 6,000 R	< 0.1mR- 100,000R	<10%-10 ¹¹	3%-10 ⁴ R	4.3x11
2600-50	Square Rod	< 10mR- 6,000 R	< 10mR- 100,000 R			6x1x1
Neutron response R/n/Cm ² : ⁶⁰ Co Rads Thermal = 1.2x10 ⁻¹⁰ Fast (PoBe) = 2.3x10 ⁻¹⁰

LiF

$\text{CaF}_2:\text{Mn}$

Mini-Gard Dosimeter and Reader

XETEX Model 403B/404A

Class	Personal, portable
Principle of Operation	A small halogen quenched, filtered GM tube detector that read-outs in chirps or continuously when preselected level reached. Also has a semiconductor memory that can be read out separately with a reader.
Sensitivity and range	Radiation detected; X, γ -rays Dose range: 1-4096mR Energy dependence: +15% from 70 keV to 1.3 MeV Maximum rate: 15R/h for rated accuracy
Sampling	Continuous intermittent readout
Performance	Alarm range: 4-1024 in binary steps Accuracy: $\pm 15\%$ for dose alarm or stored dose (referenced to ^{137}Cs) Memory: stores dose data until protected button depressed
Requirements	Power: Dosimeter, 1 - 9V battery (MN1604); Reader, 115V, 60 Hz, 10W. Size: Dosimeter, 57mm x 110mm x 25mm (2 1/4"x4 11/32"x1"); Reader 21.8x15.5x10.2cm (8.6"x6.1"x4") Weight: Dosimeter, 0.17kg (6 oz.) w/battery Reader, 1.1 kg (2.5 lb).
Features	Accumulated dose can be erased, but will be retained even if turned off Dosimeter is housed in a rugged metal box with a spring clip for attachment to the belt or pocket.
References	Manufacturer's Specifications
Cost	Dosimeter: \$170.00 Reader: 345.00
Address	XETEX Inc. 660 National Ave Mountain View, CA 94043 (415) 964-3261

Digital Dosimeter
Xetex Model 409A-1,2

Class	Portable			
Principle of Operation	A halogen-quenched, filtered G-M tube with associated electronics and LED display (409A).			
Sensitivity	Radiation detected: X- and γ -ray			
Model	Dose Increment (mR)	Dose Range (mR)	Alarm Range (mR)	Rate* Capability (R/hr)
409A-1	1	1-9999	4-1024	50
409A-2	0.01	0.01-99.99	0.04-10.24	1
402A-1**	1	1-999	4-1024	10
402A-2	0.01	0.01-99.99	0.04-10.24	0.3
Sampling	Continuous			
Performance	Accuracy: $\pm 15\%$ of indicated dose referred to ^{137}Cs Energy dependence: $\pm 15\%$ from 60 keV + 1.3 MeV Operating temperature: $0^\circ + 50^\circ\text{C}$			
Requirements	Power: MN1604 Alkaline Battery (life ~300 hrs.) Size: 7.6cmx10.5cmx3.2cm (3"x4 1/8"x1 1/4") Weight: 0.19 kg (7 oz) including battery			
Features	Units continue to respond at high rates. Internal unit provides audible signal (beep) for each dose increment and continuous tone for alarm.			
References	Manufacturer's Specification			
Cost	\$265.00 (409A-1) \$150.00 (402A-1) \$275.00 (409A-2) \$150.00 (402A-2)

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Digital Dosimeter
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Address XETEX, Inc.
 660 National Ave.
 Mountain View, CA 94043
 (415) 964-3261

*For stated accuracy; †409A only

**Models 402A-1 and 402A-2 are slave display units only. They have no
integral detectors and do not respond to radiation.

Digital Dosimeter

XETEX Model 415A

Class	Portable
Principle of Operation	Energy compensated halogen-quenched GM tube detector and all solid state electronics with 4 digit LED digital display.
Sensitivity and Range	Detector dimensions: 8mm eff. length and ~51/2mm diameter Radiation detected: X and gamma ray Range: 0-9999mR in 1 mR increments Accuracy: $\pm 15\%$ of indicated dose referred to ^{137}Cs Detector wall thickness: 80-100 mg/cm ²
Sampling	Continuous, readings push button
Performance	Energy dependence: $\pm 15\%$ from 60 keV to 1.3 MeV gammas Operating temperature: -20°C to 50°C Maximum rate: 15R/h (typical) for stated accuracy
Requirements	Power: Battery (6 months life w/MN1604 alkaline in normal use) Size: 57 x 110 x 25mm (2 1/4" x 4 11/32" x 1") Weight: 0.17 kg (6 oz.)
Features	1) Audible signal chirping with dual rate switch to select 1 to ~40 chirps per mR; continuous above 1500 R/hr 2) Go-nogo test lamp (LED) lights when battery is still good 3) Will not read zero in a high radiation field but will continue chirping above 1500 R/hr. 4) Controls: on-off switch, battery-test/display button, chirp rate switch, reset button. 5) Memory: dosimeter stores dose until protected button is depressed.

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References	Manufacturer's Specifications
Cost	\$187.00
Address	XETEX, Inc. 660 National Ave. Mountain View, CA 94043 (415) 964-3261

"Accu-Chirp" Radiation Monitor

XETEX Model 416A

Class	Portable, personal
Principle of Operation	Uses a halogen quenched Gm tube as detector and gives an audible signal, chirping with dual rate switch to select approximately 1 to 40 chirps per mR (at ^{137}Cs energy)
Sensitivity and Range	Radiation detected: X-ray and gamma Energy range: 20 keV up Saturation rate: Continues to operate above 1500 R/hr.
Sampling	Continuous
Performance	Operating temperature: -20°C to 50°C Controls: On-off switch, battery test button, chirp rate switch Accuracy: Energy dependence:
Requirements	Power: 1 - 9V battery (MN1604) Size: 57x110x25mm (2 1/4"x4 11/32"x1") Weight: 0.17 kg (6 oz.)
Features	Packaged in a rugged, anodized aluminum case with positive clip
References	Manufacturer's Specifications
Cost	\$115.00
Address	XETEX, Inc. 660 National Ave. Mountain View, CA 94043 (415) 964-3261

Radioactive Gas Monitor

JLI Model 1125

Class	Portable
Principle of Operation	<p>A positive displacement pump draws air through a sub micron filter and electrostatic precipitator removing interfering particles, ions, humidity and smoke.</p> <p>Ionization within the chamber is detected and amplified by an electrometer circuit which is calibrated to read μ curies/M^3.</p>
Sensitivity and Range	<p>Energy Sensitivity:</p> <p>Tritium Ranges: 100, 1000, and 10,000 μCi/M^3</p>
Performance	<p>Accuracy: $\pm 15\%$ full scale</p> <p>Zero drift: after 5 minute warmup less than 5% FS in 30 minutes; less than 3% FS in 4 hours</p> <p>Time Constant: Approximately 20 seconds</p> <p>Gamma Compensation: Up to 5 mR/hr for uniform radiation field</p> <p>Decontamination: Reads less than 5 μCi/M^3 in 5 minutes after 15 minute exposure to 1000 μCi/M^3 of Tritium gas.</p> <p>Airflow: Approximately 2 liters/min.</p> <p>Operating Temperature $+40^\circ\text{C}$ to $+150^\circ\text{C}$</p>
Requirements	<p>Power: 109-121 V ac, 50-65 Hz, 5 W, or battery</p> <p>Size:</p> <p>Weight:</p>
Features	<p>Automatic changeover to battery operation on power failure. Military type construction. Suitable for other Beta emitting fission product gases; C^{14}, A^{41}, R_n^{222}, S^{35}, K_r^{85}, X_e^{133}, X_e^{135}</p>
References	<p>J.R. Waters, JLI-506, "Pitfalls and Errors in the Measurement of Tritium in Air"</p>

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RAD-NUC
Gas
Johnson Labs
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Cost	\$1995
Remarks	Other models available. Instrument use high above sea level will require a correction factor due to reduced air density in chamber. "Designed for military and other rugged requirements and does not have adequate sensitivity for health monitoring" (Private communication, L. Gevins) Carrying case and straps included.
Address	Johnson Laboratories Inc. 3 Industry Lane Cockeysville, Md. 21030 (301) 666-9500

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RAD-NUC
Tritium
Nuclear Enterprises
Dec. 1971

Tritium in Air Monitor
Nuclear Enterprise Model 1762

Class	Portable, hand-held
Principle of Operation	Ion chamber, amplifier, meter read-out
Sensitivity and Range	Energy sensitivity: Range 10 to 15,000 pCi/cm ³ Interferences:
Sampling	Continuous
Performance	Meter accuracy: Drift: 2% Temperature range:
Requirements	Power: Size: Weight:
Features	
Cost	
Address	Nuclear Enterprises Inc. 935 Terminal Way San Carlos, California 94070 (415) 593-1455

Tritium in Air Gas Monitor
Nuclear Enterprise Model 1762A

Class	Portable, hand-held radioactive gas monitor												
Principle of Operation	A dual concentric gamma compensated ionization chamber, which enables measurements of HTO (tritiated water vapor) and other radioactive gases in air. Current from the chamber is measured. A dust filter and an electrostatic precipitator is used for the removal of small ionization particles and gaseous ions to ensure that the current measured is due the actual radioactive disintegrations within the ionization chamber.												
Sensitivity and Range	Range 10-50, 150, 1500, and 15,000 pCi/cm ³ of HTO in air <table><tr><th>Gas</th><th>Meter Reading for 1 MPC</th></tr><tr><td>HTO</td><td>5 pCi/cm³</td></tr><tr><td>Argon 41</td><td>4 pCi/cm³</td></tr><tr><td>Krypton 85</td><td>10 pCi/cm³</td></tr><tr><td>Radon</td><td>21.5 pCi/cm³</td></tr><tr><td>Xenon 133</td><td>40 pCi/cm³</td></tr></table>	Gas	Meter Reading for 1 MPC	HTO	5 pCi/cm ³	Argon 41	4 pCi/cm ³	Krypton 85	10 pCi/cm ³	Radon	21.5 pCi/cm ³	Xenon 133	40 pCi/cm ³
Gas	Meter Reading for 1 MPC												
HTO	5 pCi/cm ³												
Argon 41	4 pCi/cm ³												
Krypton 85	10 pCi/cm ³												
Radon	21.5 pCi/cm ³												
Xenon 133	40 pCi/cm ³												
Performance	Accuracy $\pm 20\%$ of reading above 1 MPC (MPC of HTO in air is 5 pCi/cm ³ in England) Temperature coefficient $\pm 1\%$ of full scale per 10°C change												
Requirements	Power Mallory Batteries - RM 42-R, (500 hrs), TR 134-R, (400 hrs), TR-134-R, 2 ea. (200 hrs) Everyready Batteries - Type 126, 3 ea., motor, (8-9 hrs) Size 29.8 cm H, 20.3 cm W, 64.5 cm L (11.75", 8", 25 25") Weight 12.25 Kg (27 lbs)												
Cost													
Address	Nuclear Enterprises Ltd 935 Terminal Way San Carlos, Calif. 94070 (415) 593-1455

Iodine 131 Monitor
Nuclear Enterprise NE 8423

Class	Portable
Principle of Operation	Gamma detector and three discriminators select I-131 energy fission products, meter read-outs.
Sensitivity and Range	Energy Sensitivity: Range: 0.007 μ Ci to 0.65 μ Ci Interferences:
Sampling	Continuous, sample size 7 cm dia., 6 cm thick
Performance	Accuracy: Temperature range: Response time:
Requirements	Power: 12 V d.c. car battery Size: Weight:
Features	"Subtraction" electronics displays I-131 in presence of higher and lower energy fission products. Rate meters display channels A,B,C, and B-(A+C)
Cost	
Address	Nuclear Enterprise Inc. 935 Terminal Way San Carlos, California 94070 (415) 593-1455

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RAD-NUC
Iodine
Nuclear Measurement
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Air Radiation Monitor
Nuclear Measurements AM-21

Class	Field
Principle of Operation	The monitoring system consists of two parts, the pump system and the detecting system. Housed within a lead shield assembly is a gamma scintillation detector used for detecting gamma emitters below 1 MeV and a particulate collector; as the particulate matter builds up, the detector converts the ionizing events into pulses which are counted on ratemeter, also air is entering and leaving the lead housing at a flow rate from 3-10 cfm through the filters for deposition of particulates.
Sensitivity and Range	Count rate will be 318 cpm for Iodine ¹³¹ at 10% yield, 5 cfm airflow rate, and 10^{-10} $\mu\text{c/cc}$ constant activity.
Sampling	Airflow through filters and charcoal cartridge
Performance	Ratemeter accuracy: $\pm 10\%$ of actual input cartridge Time constant: 200 seconds at 50 c/m to 1 second at 50,000 c/m Stability: Drift not more than $\pm 2\%$ daily, $\pm 20\%$ annually except in narrow window spectrometry
Requirements	Power: 90-130 V ac, 60 Hz, 5-7 amp Size: 53 cm X 1 M x 1 M (21" x 39" x 39") Weight: 250 Kg (550 lbs)
Features	Graphic recorder provision, alarm controls
Cost	\$4390 with Esterline-Angus recorder
Remarks	Other air monitors available similar to the above are given in the following table.
Address	Nuclear Measurement Corporation 24260 North Arlington Avenue Indianapolis, Indiana 46218 (317) 546-2415

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RAD-NUC
Iodine
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NMC Air Monitors

MODEL	RADIATION MONITORED	PRICE	DETECTORS	RECORDER	FIXED FILTER SA - 20 SERIES
AM - 2A	Alpha	\$2740	SC-2a	1 Single Channel or 1 Rustrak	1 C/M
AM-3A		\$3220	SC-2A	1 Single Channel or 1 Rustrak	
AM-2B	Beta	\$2960	SC-2B	1 Single Channel or 1 Rustrak	21 C/M
AM-3B	Beta	\$3440	SC-2B	1 Single Channel or 1 Rustrak	
AM-2D	Gross Beta-Gamma	\$2860	DGM-2	1 Single Channel or 1 Rustrak	21 C/M
AM-3D	Gross Beta-Gamma	\$3415	DGM-2	1 Single Channel or 1 Rustrak	
AM-2F	Alpha, Beta, Gamma	\$3010	FGM-2	1 Single Channel or 1 Rustrak	30 C/M
AM-3F	Alpha, Beta, Gamma	\$3580	FGM-2	1 Single Channel or 1 Rustrak	
AM-2G	Gamma	\$3780	SC-2-2S	1 Single Channel or 1 Rustrak	Gross 128 C/M SPEC. 10 C/M
AM-3G	Gamma	\$4410	SC-2-2S	1 Single Channel or 1 Rustrak	
AM-2I	Iodine	\$3890	SC-2-1S	1 Single Channel or 1 Rustrak	Gross 97 C/M SPEC 10 C/M
AM-3I	Iodine	\$4530	SC-2-1S	1 Single Channel or 1 Rustrak	
AM-22I	Iodine + Alpha or Beta or Gamma	\$5450	SC-2-1S and DGM-2 or PC-21-FW	1 Dual Channel or 2 Rustrak	Iodine 10 C/M GROSS 97 C/M
AM-33I		\$6015		"	
AM-2P	Alpha or Beta or	\$3310	PC-21-FW	1 Single Channel or 1 Rustrak	a < .1 C/M b 30 C/M
AM-3P	Alpha + Beta	\$3880	PC-21-FW	"	
AM-22P AM-33P	Alpha + Beta separately but simultaneously	\$3780 \$4360	PC-21-FW PC-21-FW	1 Dual Channel or 2 Rustrak	a < .1 C/M b 30 C/M
PAPM-11	Plutonium	\$4580	SC-2A	"	10 C/M

NMC Air Monitors

MODEL	MOVING FILTER SA-30 SERIES	MOVING FILTER SA-30 SERIES	FIXED FILTER SA-3L	MOVING FILTER SA-30 SERIES	MOVING FILTER SA-3L
AM-2A	-	-	20%	-	-
AM-3A	1 C/M	1 C/M	-	20%	11%
AM-2B	-	-	32%	-	-
AM-3B	21 C/M	40 C/M	-	32%	18%
AM-2D	-	-	10%	-	-
AM-3D	21 C/M	30 C/M	-	10%	5.7%
AM-2F	-	-	35%	-	-
AM-3F	30 C/M	30 C/M	-	32%	17%
AM-2G	-	-	19%	-	-
AM-3G	GROSS 128 C/M Spec. 10 C/M	GROSS 120 C/M Spec. 35 C/M	-	19%	8%
AM-21	-	-	17% GROSS 3% IODINE	-	-
AM-31	GROSS 97 C/M Spec 10 C/M	IODINE 30 C/M Gross 115 C/M	-	10% GROSS 15% Iodine	10% GROSS 15% Iodine
AM-221	-	-	17% GROSS 3% IODINE	-	-
AM-331	IODINE 10 C/M Gross 97 C/M	IODINE 30 C/M Gross 115 C/M	-	17% GROSS 3% Iodine	17% GROSS 3% Iodine
AM-2P	-	-	α 37% β 37.5%	-	-
AM-3P	α <.1 C/M β <30 C/M	α <1 C/M β <40 C/M	-	α 37% β 37%	α 20% β 22%
AM-22P	-	-	α 37% β 37%	-	-
AM-33P	α <10/M β <30 C/M	α <10/M β <40 C/M	-	α 37% β 37%	α 35% β 32%
PAPM-11	10 C/M	-	-	35%	-

Continuous Gas Monitoring
Trapelo Model MGP-1A

Class Laboratory

Principle of Operation Gas sampler, gamma scintillation or G-M tube detector, filter, ratemeter, alarm, and pump

Sensitivity and Range

SENSITIVITIES OF TRACERLAB MG-1A AND MG-2A GAS SAMPLERS
FOR NET SIGNAL EQUAL TO BACKGROUND

Activity	Type and Energy (MeV)		Half-Life	Min. Detectable Conc. $\mu\text{C}/\text{cc}$	
	Gammas	Betas		MG-1A*	MG-2A**
O^{19}	1.3 (64%) 0.2 (96%) 0.1 (4%)	4.5 (30%) 2.9 (70%)	30 Seconds	7×10^{-7}	3.8×10^{-7}
A^{41}	1.37 (100%)	1.24 (100%)	110 Minutes	6.25×10^{-7}	5.5×10^{-7}
Br^{82}	γ decay scheme in doubt	0.46 (100%)	36 Hours	-	7.6×10^{-7}
Kr^{85} I^{131}	0.54 (0.35%) 0.36 (87%) 0.64 (9%) 0.72 (3%)	0.69 (100%) 0.61 (87%)	9.4 Years 8.1 Days	3.5×10^{-4} 1.7×10^{-6}	9.6×10^{-7} 1.3×10^{-6}
Xe^{133}	0.08 (100%)	0.34 (100%)	5.3 Days	4×10^{-6}	2×10^{-6}

* With Tracerlab MD-5B Scintillation Detector (250 cpm background, 3" lead shielding).

** With Tracerlab MD-12C Large Area Beta-Gamma G-M Detector (40 cpm background 2" lead shielding).

Sampling 2 to 10 CFM

Performance Continuous operation

Requirements	Power: 110/220 V ac, 50-60 cps, 1 phase, 900 watts Size: 82 cm W, 82 cm H, 62 cm D (32.5" W, 32.5" H, 24.5" D) Weight: About 430 Kg (150 lbs)
Features	Remote control & Indication. Particulate sampler optional.
Cost	\$5,000
Address	LFE Corporation Trapelo Division 1601 Trapelo Road Waltham, Mass. 02154 (617) 890-2000

MGP-1A system

schematic Diagram MAP 1B MGP 1A Particulate and Gas Monitoring System

Continuous Particulate Monitor

Trapelo Model MAP-1B

Schematic diagram, MAP-1B.

Class

Laboratory

Principle of
Operation

Airborne particulate sampler, moving filter, beta detector,
ratemeter, alarms, control panel, enclosure, and pump

Sensitivity
and Range

Detector	Activity	Half-Life	Energy and Type in MEV	Min. Detect. Conc. p.p. cc	Back-ground	Not Signal
MD-1B	C ¹⁴	5500y	0.15β ⁻	2.5 X 10 ⁻¹¹	25 cpm	25 cpm
4 mg/cm ²	Ti ⁴⁴	3.5y	0.76β ⁻	1.0 X 10 ⁻¹¹		
Beta-Gamma	Sr ⁹⁰ -Y ⁹⁰	20y	0.54β ⁻ , 2.2β ⁻	4.5 X 10 ⁻¹²		
GM	Na ²²	2.6y	0.54β ⁻ , 1.28γ	1.0 X 10 ⁻¹¹		
	P ³²	8.1d	0.36γ, 0.61β ⁻	1.0 X 10 ⁻¹¹		
	Co ⁶⁰	5.3y	0.31β ⁻ , 1.3γ, 1.1γ	8.0 X 10 ⁻¹²		
	Cs ¹³⁷	33y	0.52β ⁻ , 1.1β ⁻	1 X 10 ⁻¹¹		
	Pm ¹⁴⁷	2.6y	0.22β ⁻	1.5 X 10 ⁻¹¹		
MD-3B	Alpha Scintillation	138d	5.3α	1 X 10 ⁻¹²	3 cpm	4 cpm
MD-4B	C ¹⁴	5500y	0.15β ⁻	1 X 10 ⁻¹²	20 cpm	20 cpm
Beta Scintillation	Ti ⁴⁴	3.5y	0.76β ⁻	5 X 10 ⁻¹²		
MD-5B	Ce ¹⁴⁴	5.3y	1.1γ, 1.3γ	9 X 10 ⁻¹¹	250 cpm	200 cpm
Gamma Scintillation	P ³²	8.1d	0.36γ, 0.61β ⁻	1 X 10 ⁻¹⁰		

Measured at air flow rate of 10 scfm, for otherwise uncontaminated filters.

Requirements

Power: 110/220m 50-60 cps, 1 phase, 1000 watts
Size: 82 cm W, 101 cm H, 62 cm D (32.5"W, 40" H, 24.5"D)
Weight: About 270 Kg (600 lbs)

Cost

\$6,000

Address

LFE Corporation
Trapelo Division
1601 Trapelo Road
Waltham, Mass. 02154
(617) 890-2000

Fixed Filter Air/Iodine
Monitor-Victoreen Model 840-2

Class	Laboratory and Power Reactor Site Monitoring
Principle of Operation	Air is pumped through a charcoal filter and the iodine particulates are counted.
Sensitivity and Range	From 10^{-12} to 10^{-6} $\mu\text{C}/\text{cc}$ at 4 cfm Model 843-2 Beta scintillation detector Model 843-3 Gamma scintillation detector Model 843-4 G-M detector
Sampling	Air pump system, 4 cfm air flow, collection eff: 95%
Performance	Meter accuracy: $\pm 2\%$ fullscale Time constant: 10^6 c/m - 1 minute 10^6 c/m - 0.2 sec. Temperature range: 0°C - 50°C
Requirements	Power: 110 Vac, 10 amps, 60 Hz, single phase (others available on request) Size: 114.3 cm H, 61 cm W, 72.5 cm D (45" H, 24" W, 28.5" D) Weight: 227 Kg (500 lbs)
Features	Single channel analyzer capabilities for iodine monitoring, and airborne particulate monitoring or both simultaneously; alarms for alert, high radiation flow rate, and failure, log and linear scales.
Reference	Manufacturer's specifications

Cost	Upon request
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Address Victoreen Instrument Division
10101 Woodland Avenue
Cleveland, Ohio 44104
(216) 795-8200

System Sensitivity with Models 843-2 Beta and 843-3 Gamma, Scintillation Detectors.

System Sensitivity with Model 843-4 G-M Detector

LASERS

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1. Introduction

Lasers were developed during the early 1960's and rapidly became available commercially. The hazard of a direct beam to eyes and skin was recognized early, and rudimentary precautions (e.g., eye protection goggles) soon spread throughout the industry. During their first decade, lasers have been improving in size, efficiency, energy, price, power and available wavelength. Consequently, over 50,000 had been sold by 1968, and the number being manufactured per year is increasing. New applications are constantly being proposed which will greatly increase the number of individuals exposed to lasers.

A listing of the characteristics of some lasers is given in Table 1 which was taken from Read (Ref. 1, Table 1.3, p. 24).

The amount of energy absorbed by a human retina varies according to wavelength, and has a broad peak at about 580 nm. Since lasers have a narrow spectrum compared to the visible band, it is reasonable to consider the Maximum Permissible Exposure (MPE) at or near 580 nm as being representative of the maximum sensitivity of the retina.

An excellent summary and justification for selecting MPE levels is given by Sliney as Chapter 7 in Laser Application in Medicine and Biology (ref. 2). Also, a comprehensive compilation of the biological data by Van Pelt et al. in a preliminary draft titled "A Review of Selected Bioeffects Threshold for Various Spectral Ranges of Light" (Ref. 3) provides a basis for selecting MPE levels.

2. Hazard Levels

It is only recently that efforts have been undertaken to define the hazard levels from reflected, refracted, diffused, and dispersed beams for legal purposes. Even relatively low power levels are dangerous to the human eye, because the focussing action of the eye can concentrate the energy on a retinal area which may be 10^5 times smaller than the entrance aperture of the pupil (Ref. 7).

A day-time or contracted pupil has a diameter of about three millimeters (area of 7 mm^2), while a night-time or fully dilated pupil has a diameter of about seven millimeters (area of 38 mm^2). A summary of the relationship between pupil and retinal image area is available in Reference 2, p. 185.

Tissue injury is assumed to be primarily a thermal effect, and studies are under way to determine the sensitivity of different sites (e.g., retina vs. cornea) to various wavelengths. The threshold of retinal damage is a strong function of the time rate of deposition of energy. In addition, various members of the same species will have different sensitivities to the same laser.

Hazard thresholds are often quoted in the research literature as that level at which 50% of the exposed population will exhibit damage. The minimum hazard threshold, where no damage is indicated in any of the specimens, may be a factor of 2 or more below the 50% level. Also, the functional damage threshold, where an eye exhibits a localized loss of sight, must be distinguished from a microscopic alteration of tissue structure detectable by ophthalmoscopic observations or histochemical techniques.

TABLE I
Laser Characteristics

Laser	Wavelength (μ)	CW power (W)	Peak pulse power (W)	Pulse energy (J)	Pulse rate (pps)	Beam angle (mrad)
He-Ne	0.6328	0.001-0.1	—	(rarely used pulsed)	—	0.2-1
Ar	0.4880 0.5145	1-10	20	mJ	60-200	0.5-1
Kr	0.6471 0.5681 0.5208 0.4762	0.5	2	mJ	60	0.5-1
CO ₂	10.6	10-5000	10	0.1-5	~100	~1-4
N ₂	0.3371	Not CW	10	—	1-100	2-20
Ruby	0.6943	1	to 10^6	1-500	60	1-10
Nd:Glass	1.06	—	to 4×10^4	1-1000	1-10	0.5-10
Nd:YAG	1.06	1-250	to 10	0.01-1	to 10000	2-20
Cr:As	0.84-0.9	1-10	1-100	1 mJ	100-1000	20-400

3. Biological Effects

A recent book edited by M.L. Wolbarsht (Ref. 2) is a collection of nine chapters written by twelve authorities in the field. It provides a compilation of the state of knowledge in this area and an extensive guide to the literature. Laser effects on normal and tumor tissue, organelles, macromolecules, teeth, various organs and ocular damage studies are described. It is within the scope of this study to document the biological hazard only to the extent that it affects the instrumentation. The biological sensitivity influences the MPE which in turn is a factor in determining the necessary instrument sensitivity.

The usual problems of extrapolating animal data to human response are present. One approach (Ref. 8) is to select the most sensitive organ (eyes) of a similar animal (monkeys) and measure the sensitivity of as many subjects as are necessary to establish a 50% Retinal Damage Level, RDL (50% of the subjects exhibit damage, within 50 minutes after exposure).

Definitions of an "absolute" threshold and an "average" threshold (50% level) are given in Reference 3 as well as the following quotes (p. 19), "Laser light may be relatively more hazardous than light from other sources because the radiant properties (monochromaticity, coherence) which can be produced by laser action allowing better focussing of the incident light upon the retina, and thus produce a greater hazard. Note that not all lasers produce monochromatic coherent light of low divergence, and that not all lasers of the same output power pose equal hazard"; (p. 20) "The principle mode of damage is dependent upon the tissue involved and the energy of the incident light quanta. Infrared light usually effects damage by a general increase in tissue temperature whereas ultra-violet light tends to cause more specific photochemical reactions. Both modes are present in the visible portion of the spectrum."; (p. 24) "Indeed the process of vision itself is dependent upon molecular adsorption of photons with an associated molecular denaturation.

4. Maximum Permissible Exposure

Sliney tabulated seventeen MPE recommendations in Reference 2, p. 174 and it has been reproduced here as Table 2. For continuous wave exposure to ruby lasers by dark adapted (fully dilated) eyes, six of the researchers agreed on recommending $1 \mu\text{W}/\text{cm}^2$ for the MPE.

The American National Standards Institute (ANSI) has formed the "Z-136 Standards Committee on the Safe Use of Lasers and Masers"

to recommend an MPE. Their report should be published in 1972.

Figure 1 is a simplification which shows the MPE levels currently recommended by the American Conference of Governmental Industrial Hygienists (ACGIH, Ref. 9) and the Lawrence Livermore Laboratory (LLL, Ref. 10). Also, Ham's data (Ref. 6) has been included as a reference of the threshold for retinal damage. The data cover the tremendous range of 12 orders-of-magnitude, and are intended to provide the reader with an appreciation of the current state of knowledge and controversy in this area. Some recent experimental work (Ref. 7) presents damage levels of rhesus monkeys, including a comparison of experimental data with calculated thresholds based on a simple thermal model.

A large number of variables, including how soon the eyes were examined after exposure, the animals used, the statistical uncertainty, and the absorption of the transmitting medium (usually air) will contribute to a recommended MPE. After including a reasonable safety factor, a recommended MPE can be proposed as a U.S. Standard for direct illumination of human eyes. The recommended MPE for direct viewing of a diffuse surface reflection would then also have to be determined, along with skin levels for differently pigmented skins. The long-term low-level exposure problem has not yet been sufficiently documented (Ref. 4).

Figure 1 also contains a datum point for direct sun exposure, arbitrarily plotted as a continuous wave. A generally accepted solar constant (Ref. 12), defined as the amount of solar radiation reaching the average earth orbit outside the atmosphere, is $0.137 \text{ W}/\text{cm}^2$. If one assumes that about 75% reaches sea-level, then the direct sun exposure is approximately $100 \text{ mW}/\text{cm}^2$. This number is only included as an order-of-magnitude reference since many injuries have resulted from looking at or near noon sun, particularly when viewing an eclipse, or (worse yet) using binoculars.

The sun can appear as a $160 \mu\text{m}$ diameter point on the retina, whereas a laser can be focussed to an image of about $20 \mu\text{m}$. As the image size decreases, the power density (W/cm^2) increases, which means that the probability of burning the retina is increased.

The "optical gain" of a human eye can be defined as the ratio of cornea spot size to retinal image area. It is typically 100,000 to 200,000. Considering this magnification factor, the retinal irradiance could be as high as $1000 \text{ W}/\text{cm}^2$ from a small 2 mW helium-neon laser (Ref. 7). Several factors influence

the optical gain of an eye for any particular situation:

- a. Pupillary diameter
 1. background illumination
 2. size of the light source(s)
 3. degree of collimation of light being emitted by the source
- b. Distance between the eye and the light source
- c. Distance at which the eye is focused
- d. Overall acuity of the eye and any corrective eye wear.

The safe irradiance levels at the cornea are determined by calculating the radiant exposure (J/cm^2) at the cornea, which is below the minimum energy density at the retina required to produce detectable damage.

The skin exposure ceiling value for continuous wave lasers recommended by the ACGIH is $1.0 W/cm^2$ (Ref. 9).

The difference between collimated coherent and non-coherent radiation, the wavelength sensitivity of the eye, and the pulsed nature of some lasers are among the considerations that must be kept in mind when attempting to evaluate the possible hazard from a laser.

5. Measurement Considerations

A variety of instruments are available that measure the power of laser beams. However, the problems encountered in measuring the environmental or industrial hazard due to dispersed and reflected beams require an instrument to meet more stringent design specifications. The following list of items needs to be considered when designing a laser hazard monitor:

- a. Sensitivity: The instrument should be sensitive to at least $1 \mu W/cm^2$ for low-level occupational exposure to continuous wave lasers.
- b. Bandwidth: The instrument should cover the ultraviolet, visible, and near infrared regions. A variety of easily mounted optical filters would provide selective sensitivity to different wavelengths.
- c. Interferences: If the instrument has the required sensitivity and bandwidth, then it will respond to room lamps, sunlight, and specular reflections. Two measurements, with and without the optical filter, will provide the user with both the total intensity and the contribution from one wavelength.

d. Geometric Acceptance: The solid angle of sensitivity should be similar to the human eye. If the acceptance is too narrow then a significant reflection may not be detected; if it is too broad then it is difficult to find the source.

e. Dynamic Range: Since a specular reflection may be 10% (or more) of the energy in the beam, the instrument should be able to measure high doses without burning out.

f. Modulation: Continuous wave and pulsed lasers may be modulated, which will affect the temporal rate of deposition of energy. Both average (for CW) and peak (for Q-switched) readings should be provided.

g. Frequency Response: The peak frequency response (maximum repetition rate) and the highest measurable duty cycle should be graphically described, illustrating the 1 db and 3 db points.

h. Biological Weighting: The instrument should allow a comparison to the current Maximum Permissible Exposure guidelines. A readout in units of "hazard factor" should be available. For example, a green section indicating exposures less than a decade below the MPE threshold, a yellow section up to the MPE level, and a calibrated red area above the MPE level.

i. Accuracy: Since the difference between "safe" and "unsafe" is not a precise discontinuity, the instrument does not require extremely high accuracy (e.g. $\pm 15\%$).

j. Engineering Features: The usual objectives of all survey instruments (e.g., lightweight, rugged, compact, rechargeable, inexpensive) should be provided. Options (e.g., audible alarms, and recorders, may be useful for some applications.

k. Calibration: An independent calibrated source should be available to allow a complete check of the instrument's operation from sensor to the readout, for fail-safe operation.

l. Aperture stop: The aperture stop should be similar to a fully dilated human eye (7 mm dia.) over the visible light range (400 to 1400 nm). For wavelengths from 0.1 to 400 nm and from 1400 to 10^6 nm the stop should be 1 mm in diameter. The relative sensitivity per unit area should be constant within $\pm 10\%$ across the aperture stop when mapped with a 1 mm diameter beam.

6. Instrument Comparisons

Table 3, compiled by Zimmerer in September 1970 (Ref. 13) contains a list of manufacturers of meters which measure power in laser beams. A revised edition of Table 3 titled "Update on Power Measurement" by R. Zimmerer was published in the March 1972 issue of Laser Focus magazine.

These tables provide a convenient listing of some manufacturers' in this field. The reference list in Honey's paper (Ref. 5) has also been reproduced here; it contains remarks to guide the interested reader.

It can be assumed that accidental exposure to a direct laser beam will be avoided by proper design of the installation; safety design to avoid accidental exposure to reflected light cannot be so easily assumed.

As summarized in the next section, no completely satisfactory survey instrument exists for monitoring reflected light from lasers at the power levels of interest. Unfortunately, even if such an instrument did exist, its use would require great care. The danger lies in the premise that if a survey of a powerful laser showed that the reflected-light situation was "safe", a careless or cavalier attitude might develop; it takes but one accidental specular reflection or the slightest change in optical arrangement to cause serious injury.

It should therefore be emphasized, even while discussing measurement instrumentation, that a "safe" laser installation requires above all common sense and proper safety engineering.

Table 3 allows one to easily and rapidly compare the commercially available laser beam monitors. The table lists power, spectral range, price, and company addresses.

A few other companies were found in addition to those cited in the table: Gamma Scientific Inc., 3777 Ruffin Road, San Diego, California 92123, manufactures systems and calorimeters for light measurement; Sylvania Electronic Systems, P.O. Box 188, Mountain View, California 94040 has a Model 502 detector; and Laser Precision Corporation, 5 West Whiteshore Street, Yorkville, N.Y. 13495, sells pyroelectric detectors.

Only two instruments were found during this survey which were specifically designed for surveying of hazard levels. Unfortunately,

neither of these instruments completely fulfills the requirements for a survey meter. We shall discuss them separately:

A. The "Laser Hazard Monitor" made by International Laser Systems (I.L.S.), comes remarkably close to being an ideal survey meter. Designed partly under contract with the U.S. Public Health Service Bureau of Radiological Health (BRH), it attempts to incorporate the temporal pulse-power spectrum (see Figure 1) from 10^{-8} second to one second pulse durations. Unfortunately, it lacks sufficient sensitivity to measure power levels much below Ham's curve in Figure 1.

The instrument incorporated Ham's curve directly into its electronic circuitry and allows a direct measurement (by suitable choice of selectable temporal interval) of the power level as compared to this curve. The readout is in terms of "hazard factor" from 0.1 to 1000, where "hazard factor" is defined as the ratio of measured laser power to the biological reference curve. The bandwidth covers the visible range and extends to the 1060 nm line from the Neodymium laser. The response is corrected for the spectral sensitivity of the human eye; one possible source of error is the implicit assumption that the damage threshold as a function of wavelength is the same as the detectable spectral distribution for all the different possible damage sites in the eye.

It is most unfortunate that this instrument's sensitivity is inadequate since many of its features are the most advanced in this field. One other problem is the lack of a calibrated source to augment the battery test circuit; this feature would provide a much-needed fail-safe calibration feature.

B. The "Laser Energy Evaluator" manufactured by United Detector Technology has the required sensitivity (down to $0.1 \mu\text{J}$) but does not have the temporal range feature. However, it does have a fail-safe system which is very important and is better than simply a battery test circuit. A Light Emitting Diode (LED) provides a calibrated radiation source to completely check the accuracy and operation of the instrument from the sensor to the meter. This is a crucial feature when trying to determine if a radiation level is really hazardous, since the decision depends on the accuracy and reliability of the measurement. The next question is, "How accurate and stable

is the calibration source?" The frequency distribution will remain stable since it is a result of electrons changing states which is a property of the material and its doping. The intensity should remain reasonably stable, but this is a function of temperature and conversion efficiency. LED's appear to be the most stable, longest-lived light source yet produced. Even if the calibrator drifts, the user will be alerted since the meter will not read the proper calibration level.

Another problem with the Laser Energy Evaluator is its directional sensitivity. The probe is meant to be hand-held, but the acceptance cone appears to be so narrow that it cannot easily be held in position, unless aligned on a support (e.g., a tripod). Its spectral response (450 to 900 nm) covers the region of most laser applications; however, it does not extend to the 1060 nm Neodymium laser.

7. Summary and Recommendations

Instrumentation for measuring the energy or power in laser beams is readily available (e.g., see Table 3). Only one instrument can measure the biologically weighted hazard factor, but it lacks sufficient sensitivity to measure the low energy from dispersed and reflected lasers.

Researchers in the field are still debating where the Maximum Permissible Exposure levels should be set. New lasers that produce high power bursts in picosecond pulses, and long-term low-level exposures have not yet been studied.

There is no single instrument available with the required sensitivity, temporal response, fail-safe operation, and biological weighting required to completely evaluate the environmental hazard from lasers. International Laser Systems' hazard monitor has most of the required features, lacking only sensitivity.

The most important need appears to be incorporation of high-sensitivity into a device whose philosophy is similar to that of the I.L.S. instrument. Electronic compensation for the biologically-weighted temporal response curve is an extremely valuable feature. A very desirable feature is a measurement which reads directly in "hazard factor" (ratio of measured level to the MPE level). A direct method for calibration (perhaps using an LED source) is a necessary fail-safe feature. Another design criterion is sufficient angular acceptance to permit hand-held use. Extremely high accuracy is not required (10 to 15% measurement error is quite sufficient), and any design should sacrifice accuracy beyond this level if such a requirement would compromise other features or increase the cost.

8. Acknowledgment

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Lloyd D. Stephens, Lawrence Berkeley Laboratory

Arthur Vassiliadis, Stanford Research Institute

Maximum Permissible Exposure (MPE) at cornea to prevent retinal damage from visible and near infrared lasers for a fully dilated (7 mm dia.) human eye and Retinal Damage Level (RDL) for direct illumination.

Figure 1

TABLE 2

TABLE II A Comparison of Maximum Recommended Corneal Exposure Levels for Three Operating Modes at the Ruby Laser Wavelength for Intrabeam Viewing by the Dark-Adapted Eye

Author and organization	Data source ^a	Safety factor	Pupil (mm)	d_r (μ)	Continuous wave (CW) (W/cm ²)	Non-Q-switched (J/cm ²)	Q-switched (J/cm ²)
Straub (1963), Harry Diamond Laboratories	MCV	100	7.1	16	—	5×10^{-8}	—
Jones and Montan (1964), U. Calif. Lawrence Radiation Laboratory	MCV	80	7.1	11.4	2.5×10^{-6}	2.5×10^{-8}	2.5×10^{-9}
Schlickman and Kingston (1965), MIT Lincoln Laboratories	MCV	10	3	10	—	5×10^{-7}	—
Weston (1965), Ministry of Aviation, Great Britain	MCV OSU	≥ 10	6	8.5	5×10^{-5}	2×10^{-8}	2×10^{-9}
Goldman and Hornby (1965, 1966), Cincinnati Children's Hospital	MCV	—	—	—	1×10^{-6}	1×10^{-7}	1×10^{-8}
Electronic Engineering Association Great Britain (1966)	MCV OSU	—	—	—	1×10^{-6}	1×10^{-8}	1×10^{-9}
Flint (1966), Martin-Marietta Corp., Orlando, Florida	MCV	10	7.5	16	—	3.2×10^{-7}	3.2×10^{-8}
Vos (1966), Institute for Perception, RVO-TNO, Netherlands	MCV	1	6	5	1×10^{-3}	1×10^{-5}	5×10^{-8}
Kaufman (1956), General Precision Corporation	MCV	20	7.1	10	6.8×10^{-7}	6.3×10^{-8}	5.3×10^{-9}
Slaney (1967), Slaney and Palmisano (1968), USAFHA	MCV	≥ 10	7	10	1×10^{-6}	1×10^{-7}	1×10^{-8}
Beauchamp and Pantell (1967), Stanford Univ.	MCV	—	7.1	—	2.5×10^{-6}	2.5×10^{-8}	2.5×10^{-9}
Nevada Operations of the USAEC (1967)	MCV	—	—	—	3×10^{-6}	1×10^{-8}	0.1 w/cm ² peak
American Conf. of Governmental Industrial Hygienists (1968)	MCV	> 10	7	10	1×10^{-5}	1×10^{-7}	1×10^{-8}
Wilkening (1968), Cincinnati Laser Safety Conference	MCV	> 10	7	10	1×10^{-6}	1×10^{-7}	1×10^{-8}
Armed Forces-National Research Council Committee on Vision (1968)	MCV	—	—	—	1×10^{-6}	1×10^{-6}	1×10^{-7}
U.S. Department of the Air Force (1968)	SRI	—	7.1	—	2.5×10^{-2}	2.5×10^{-4}	1.9×10^{-6}
U.S. Departments of the Army and the Navy	MCV	^c	7	—	1×10^{-6}	1×10^{-6}	1×10^{-7}

^a The referenced source of biologic data were from three institutions: the Medical College of Virginia, MCV; the Ohio State University, OSU, and the Stanford Research Institute, SRI.

^b CW level is for the argon laser specifically.

^c Safety factors of 2 to 10 are recommended for applications, thus lowering the listed levels.

Note: Table II provides an overall picture of the variation in present safe levels specified at the cornea. Since most sets of safe levels vary with laser type and environmental conditions, one situation was taken for comparison, namely the ruby laser wavelength for the dark-adapted pupil. Many sets of standards provide safe levels for indoor, nighttime and daytime conditions. Essentially all (with the exception of Schlickman and Kingston) provide levels that may be used for the nighttime, dark-adapted pupil. When the pupil size was not specified as in the data of Straub, Jones and Montan, Kaufman, Beauchamp and Pantell, and the U.S. Department of the Air Force, their safe power or energy into the eye was divided by the area of a dark-adapted pupil—0.4 cm²—corresponding to a pupil size of 7.1 mm for presentation in Table II. The levels expressed in Table II assume that the pupil is smaller than the laser beam diameter at the eye. If the laser beam diameter is smaller than the pupil, as in the case of some CW lasers, power or energy entering the eye would be a more appropriate expression for the safe level. For comparison, the source of biological data, the safety factor, the pupil diameter and the retinal image (d_r) which were described in the reference as being utilized in the derivation of the level are also provided.

TABLE 3
Laser Power Meters (Ref. 13)

Manufacturer	Instrument	Power	Spectral Range (μm)	Price	Output	Comments
Central Semiconductor 1501 North Arden Dr El Monte, Calif 91734	CSC 11, CSC 12	10^{-1} to 0.2 W/cm^2	0.3 to 11	\$20 to \$230	BNC or terminal	Silicon photodiode, selected type 110C silicon cell with various calibrations
Cintra Inc 440 Logue Ave Mountain View, Calif 94040	Thermal Radiometer Model 202 with Probe 2221	1 to 10,000 W	Visible to microwave	\$4,190	Digital	Calorimeter air cooled
Coherent Radiation 932 East Meadow Dr Palo Alto, Calif 94303	Model 212	10^{-6} to 10 W	0.3 to 1.1	\$875 to \$975	Analog meter	Silicon photodiode battery powered
	Model 201	0.1 to 100 W	0.3 to 30	\$925	Analog meter	Coated disc absorber, air cooled
	Model 205	0 to 20 W	0.3 to 30	\$800	Analog meter	Uses same readout as 201
	Model 206	2 to 200 W	0.3 to 30	\$1,995	Digital panel meter	Uses 201 head
	Model 213	10 to 100 W	10.6	\$1,250	Analog meter	Water cooled, con- centric disc absorbers
Diversified Scientific Instruments Inc 2474 Tea Garden St San Leandro, Calif 94577	DSI Power Meter	0 to 5 mW	0.3 to 1.1	\$55	Built in meter	Silicon self- powered
EG&G Inc. 183 Brockline Ave Boston, Mass 02215	Model 580 Radiometer with accessories	Uses beam samplers	0.35 to 1.2	\$2,500	Analog meter	Uses photomultiplier tube
ElectroTest Instruments 2834 Colorado Ave Santa Monica, Calif 90406	Model 50 Laser Power Meter	0 to 50 mW	0.4 to 11.5	\$125	Built in meter	Silicon, photovoltaic
Eppl Laboratory Inc Box 419 Newport, RI 02841	Laser Thermopile	0.25 W maximum	0.1 to 40	\$525 to \$735	80 mV/W	Self powered, 1/8 inch aperture, water- cooled
Hadron Inc (Control Data/TRG) 800 Shattuck Dr Westbury, NY 11590	TRG 107, 108 Ballistic Thermopile	10^{-3} to 500 joules, 500 MW maxi- mum pulse power	0.3 to 15	---	35 μV per joule	Twin cone thermopile, 103 has photodiode monitor
	TRG 117, 118 Ballistic Thermopile	1,000 joules 500 MW maxi- mum pulse power	0.3 to 15	---	22 μV per joule	Twin cone thermopile, 118 has photodiode monitor
Hy-Cal Engineering Co 12105 Los Nietos Rd Santa Fe Springs, Calif 90670	Laser Calorime- ters 1840	0.01 to 1 W	0.3 to 6	----	1 V/W	Has cooling fins and water cooling
	C 1839B	0 to 1 W	"wide"	--	Shielded leads	Cone absorber
	C 1829A	600 W maximum	6 to 14	--	Shielded leads	One inch mirror over 90% reflectivity deviates beam 90° Analog meters avail- able for use with the calorimeters
International Light Inc Dexter Industrial Green Newburyport, Mass 01950	IL600 Research Photometer with proper accessories	10^{-9} to 6, watts or joules	0.3 to 1.1	\$900 to \$2,500	Analog meter	Silicon photodiode or photomultiplier tube
Jodon Engineering Associates Inc 145 Enterprise Dr Ann Arbor, Mich 48103	Optical Power Meter	0.02 to 100 mW	0.3 to 1.1	\$298	Built in meter	Silicon self- powered
Korad Dept Union Carbide Corp 2520 Colorado Ave Santa Monica, Calif. 90406	K-J2 Laser Calorimeter	0.05 to 100 joules, 100 MW per cm^2 maxi- mum pulse power density	0.5 to 11	\$945	20 μV /joule	Liquid cell calorimeter
	K-J3 Laser Calorimeter	0.1 to 500 joules, 500 MW per cm^2 maximum pulse power density	0.5 to 11	----	5 μV /joule	Liquid cell calorimeter
	K-D1	Sample beam with diffuse reflector	0.3 to 1.1	\$1,650	2 ma/W	Uses photomultiplier tube for 0.3 nano- second rise time
	K-PM	1 to 500 W	1 to 11	\$1,095	Analog meter	Coated metal absorber
Laser Associates Ltd 172 Bradford Rd Slough, Bucks, England	Model 42 Calorimeter	1 to 10^3 joules	----	\$900	200 μV /joule	Broad band dual cones, used with Model 43 battery powered indicator unit
	Model 43 Light Meter	0.2 to 2 mW full scale	0.35 to 1.15	\$675	Built in meter	Silicon photodiode, very fast response
Light West Box 257 Van Nuys, Calif 91408	Model D 1 Laser Power Detector	0.1 to 100 mW	0.3 to 1.1	\$132	Built in meter	Silicon self powered

TABLE 3 (cont.)
Laser Power Meters (Ref. 13)

Manufacturer	Instrument	Power	Spectral Range (μm)	Price	Output	Comments
Molelectron Corp 930 Thompson Place Sunnyvale Calif 94086	Model Si Bafometer		1 μm to 4 mm	\$585	1.2 $\times 10^5$ V/W at low chopping frequencies	Silicon vacuum mounted and liquid helium cooled
Optics Technology Inc 901 California Ave Palo Alto, Calif 94304	Model 615 Laser Power Meter	0.03 to 1,000 mW	0.3 to 1.1	\$445	Built in meter	Silicon self powered
Perkin Elmer Corp 131 Danbury Rd Wilton, Conn 06897	Model 5000 Power Meter	0.3 to 100 mW	0.45 to 1.0	\$395	Built in meter	Silicon photodiode
Quantronix Corp 225 Engineers Rd Smithtown, N Y 11787	Model 500 & 501 Energy Receiver	1,000 joules or 5 $\times 10^9$ W	0.3 to 1.1	Model 500 with 504 readout	5 mV/joule	Folder cone absorber 501 has photodiode monitor 504 is a c line powered meter
	Model 502 & 505 Power Receiver	1 mW to 30 W	0.3 to 1.1	less than \$1,200	5 mV/W	Coated absorber 505 has photodiode monitor
Scientech Inc 1724 14th St Boulder, Colo 80302	LTM 10 and LTM 20 Disc Calorimeter	0.1 to 30 W	0.4 to 2.0	\$785 to \$1,185	100 $\mu\text{V/W}$ or battery powered analog meter	4 inch or 3 inch diameter disc with absorbing coating and 100 ohm manganin heater for d-c sub stitution power calibration
	LTM 36 Disc Calorimeter	1 mW to 3 W	0.4 to 2.0	\$375	100 mV/W	1 inch diameter disc with absorbing coating and 40 ohm manganin heater for d-c substitution power calibration A.C. line powered meters available at \$275 and \$350
Scientifica and Cook Electronics Ltd 40-48 High St Acton, London W3 England	Model LPM 101 Laser Power Monitor	0.3 to 100 mW	0.45 to 1.05	\$240	Built in meter	Silicon photodiode
Spectra Physics Inc 1250 West Middlefield Rd Mountain View Calif 94040	Model 401C Power Meter	0.03 to 100 mW	0.45 to 1.05	\$355	Analog meter	Silicon self powered
Spex Industries Inc 3880 Park Ave Metuchen N.J. 08840	Model 1448 Laser Power Meter	250 mW maximum	0.44 to 0.65	\$400	Built in meter	Silicon self powered
TRW Instruments 139 Illinois St El Segundo Calif 91408	Model 74A Laser Pulse Detector	0 to 12 W at 0.5 μm	0.3 to 1.1		BNC, 50 ohm	Silicon photodiode less than 1 micro second rise time
United Detector Technology 1732 21st St Santa Monica, Calif 90404	Digital Laser Power Meter	10 ⁻⁹ to 10 W	0.3 to 1.1	\$840	Digital with BCD	Silicon photodiode

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Laser Hazard Monitor
SWRHL Helium-Neon Survey Meter

Class	Portable survey meter for 634.8 nm.
Stage of Development	Prototype finished in November 1970, not intended for commercial production.
Principle of Operation	Photoconductive detector regulates the current to a meter
Sensitivity and Range	Only for continuous wave irradiance at 632.8 nm. (He-Ne) Minimum sensitivity $10\mu\text{W}/\text{cm}^2$ Apertures dia = 1.46 cm 1.12 cm 0.56 cm 0.21 cm
Sampling	Continuous
Performance	Accuracy: 7% change from 0°C to 50°C Stability: Unknown
Requirements	Power: Three 9V transistor radio batteries, 10 hr minimum Size: 3" x 4" x 5" Weight: unknown
Features	Audible indicator allows operator to devote his attention to aligning the detector with respect to the radiation being measured.
References	BRH/SWRHL document 70-2, Nov. 1970, by R. W. Peterson, W. F. Van Pelt, and H. F. Stewart
Cost	Electronic Parts about \$60 Optical housing unknown
Remarks	Zener circuit regulates the voltage; battery test circuit not rechargeable; no "live zero" circuit; entire aperture must be uniformly illuminated; ambient light is also measured.
Address	Bureau of Radiological Health U.S. Public Health Service Rockville, Maryland 20852 Attn: R.W. Peterson, W.F. Van Pelt, H.F. Stewart (301) 443-2965

INSTRUMENTATION
FOR ENVIRONMENTAL
MONITORING

RAD-LAS
Laboratory
E.G. & G.
Mar. 1972

Laser Measurement System

E.G. & G. Model 580-20A

NOMINAL CALIBRATION CURVES -
DETECTOR HEADS

Class

Laboratory

Principle of
Operation

An optical detector head uses a vacuum photodiode to convert irradiance to current for the Model 580-11A or 580-12 (Digital) indicator units which measures current and charge. Three standard detectors are available S-1, S-5, and S-20.

Sensitivity
and Range

Spectral Range 350 to 1200 nm - S-1
 200 to 775 nm - S-5
 350 to 800 nm - S-20

Performance

580-20A DETECTORS - NOMINAL PERFORMANCE CHARACTERISTICS

Detector Type	Surface	Spectral Range $\mu\mu$	Irradiation Inc. $\mu\mu$ Range Full Scale	Incident Power Range Full Scale
580 20A	S 20 with photometric filter	Intensity	1.3×10^{-2} to 3.3×10^{-6} W/cm ²	1.3×10^{-1} to 3.3×10^{-7} W/cm ²
	S 5	200 - 775	1.5×10^{-8} to 4.5×10^{-1} W/cm ² at 400 nm	1.5×10^{-8} to 4.5×10^{-1} W/cm ² at 400 nm
580 20A	S 20	350 - 800	7.1×10^{-9} to 2.1×10^{-1} W/cm ² at 420 nm	7.1×10^{-9} to 2.1×10^{-1} W/cm ² at 420 nm
580 20A	S 1	350 - 1100	2.3×10^{-7} to 6.9×10^{-1} W/cm ² at 800 nm	2.3×10^{-7} to 6.9×10^{-1} W/cm ² at 800 nm

Requirements

Size: 12" x 5-1/4" x 7"
Weight: 8 lbs

Features

Has provisions for scope display

Cost

\$2100

Remarks

Reads out to either 580-11A indicator unit or to the 580-12 digital indicator unit.

Address

E.G. & G. Inc.
Electro-Optics Division
35 Congress Street
Salem, Mass. 01970
(617) 745-3200

Portable Broadband Laser Hazard Monitor
International Laser Systems

Class Portable

Stage of Development Prototype being evaluated by U. S. Bureau of Radiological Health

Principle of Operation Optical input has four fiber-optics bundles which feed four detectors. Detectors are silicon PIN photosensitive diodes, each connected to a separate integrator. Each of the four channels has a different response characteristic with a flat response over a separate temporal range (10^{-8} to 10^{-6} sec; 10^{-6} to 10^{-4} sec; 10^{-4} to 10^{-2} sec; 10^{-2} to 10^{-1} sec). The read-out is in terms of a biologically-weighted "hazard factor", (defined as the ratio of level being measured to damage threshold). The biological weighting is accomplished within the electronics.

The biologically-weighted function used by the instrument for determining "hazard factor" is shown in Figure 1, which plots "damage threshold for the human eye" as a function of Power Density (watts/cm^2) and Pulse Train Duration (sec).

Sensitivity and Range

1. Basic spectral sensitivity is from 400 to 1100 nm, spectrally corrected from the human eye's spectral sensitivity.
2. Aperture has diameter of 7.5 mm which simulates a fully dilated human eye.
3. Unit is capable of integrating laser pulses or pulse trains of duration from 10^{-8} seconds to CW.
4. Normal angular field of view is about 60 degrees; narrow field adapter and sun-shield is provided as an accessory.
5. Sensitivity is over range of "hazard factor" from 0.1 to 1000 (where 1000, for example, would be the very hazardous condition in which the measured level exceeds the threshold-damage level of Figure 1 by a factor of 1000).

Sampling	Unit is activated by a push button. Response of panel meter is instantaneous.
Performance	In NORMAL position (front panel button) the system simultaneously monitors all four temporal channels, identifies the channel which has the largest hazard factor, and displays the hazard factor associated with the channel. Individual display is also available for each of the four temporal channels separately.
Features	<ol style="list-style-type: none">1. Input from remote optical head uses a 4-foot armored fiber-optics bundle.2. Filters are available for the most common laser wave-lengths: 488, 515, 530, 568, 633, 647, 694, 840, 910, and 1060 nm. Filters are not needed in most applications. They would be required for a frequency-doubled laser, however.3. Narrow-field adapter at optical input allows device to isolate a highly-directional source.4. Calibration done at factory. Field calibration technique not yet developed.
Requirements	<ol style="list-style-type: none">1. System operates on 115 V ac line, or on rechargeable Ni-Cd batteries (8-hour lifetime). Panel light indicates charge condition of batteries.2. Size: 9" W x 50" x 14" H3. Weight: 20 lbs.
References	<ol style="list-style-type: none">a.) Developed by the company under contract with the U.S. Bureau of Radiological Health. (Contract: Mr. Roger Schneider, BRH, 12720 Twinbrook Pkwy., Rockville, Md. 20852, Tel. (301) 443-2824.)b.) W.T. Ham, Jr., R.C. Williams, H.A. Mueller, D. Guerry, A.M. Clarke and W.J. Geeraets, "Effects of Laser Radiation on the Mammalian Eye", Trans. New York Academy of Sciences, Vol. 28, p. 517, 1966.
Remarks	This instrument is the first which attempts to incorporate the biological-response characteristics of the eye when exposed to different pulse-rates from laser sources. It is also the first designed for measurement in the hazard area involved with diffuse reflections. Its large dynamic range allows a survey of hazard levels in a large variety of situations within this single instrument.
Cost	\$10,800 plus \$250 per calibrated wavelength channel
Address	International Laser Systems, Inc. 3404 N. Orange Blossom Trail Orlando, Florida 32804 (305) 295-4010 Attn: Mr. Graham Flint

Figure 1.
Damage Threshold for Human Eye (Ref. 2)

Laser Hazard Monitor
Laser Energy Evaluator

Class	Portable field survey meter
Principle of Operation	7 mm diameter aperture (simulates fully dilated human eye) backed by a photo-diode circuit.
Sensitivity and Range	Spectral Response: 350-100 nm (flat from 420 to 940 nm) Minimum Sensitivity: About 0.1 μ J pulsed Ranges: Pulsed 0-1 μ J, 0-3 μ J, 0-10 μ J, 0-30 μ J Continuous Wave 0-3 mW, 0-10 mW
Sampling	For pulsed laser unit samples up to 100 pulses/sec For continuous wave lasers, unit gives watts into 7 mm diameter aperture
Performance	Accuracy: $\pm 10\%$ Stability: 2%
Requirements	Power: Two replaceable mercury batteries 50 hr operation (minimum) Size: 3" x 4" x 7" (approximate) Weight: 3 lbs
References	a.) Developed under contract with U.S.A.F. School of Aerospace Medicine, San Antonio, Texas, Attention: Capt. Donald Marston b.) Meets U.S.A.F. regulation AFM-161-8, April 1, 1969
Cost	Meter about \$1000 LED calibration source about \$200
Remarks	Light emitting diode calibration source provides fail-safe circuit
Address	United Detector Technology 1732 - 21st Street Santa Monica, California Attn: Mr. George Richardson (213) 829-3517

Laser Hazard Monitor
Laser Energy Evaluator