SECTION 9 - INTERNAL DOSIMETRY CONSIDERATIONS

A. Main Factors Affecting Dose Calculations

Much effort has been directed toward the problem of calculating the dose which a person receives as a result of internally deposited radionuclides. Among the many factors of main concern that enter into such a calculation are: the shape of the organ, the type of radiation, the amount of the deposit, and the distribution of the deposit.

When the organ is in the shape of some standard geometric form, such as a sphere, the problem is somewhat lessened. If one can assume a uniform distribution of the deposit, the problem is further reduced. Then, if the range of the radiation is small compared with the size of the organ, the calculation of the absorbed dose becomes rather simple. One can then say that the energy emitted per unit time is equal to the energy absorbed per unit time. A knowledge of the concentration (Bq/kg), decay scheme of the radionuclide, and the energy of the emitted radiation is all that is needed to complete the calculation.

It is rare when one is able to apply this ideal model to an actual case. For this reason, one cannot make a truly precise calculation of the organ dose. The most vital factor which affects the accuracy of the result is the nonuniform distribution of the deposit. This is clearly seen in the case of alpha particles, whose range in tissue is so very short. In this respect, local "hot" spots may be the item of chief concern, rather than the average dose over the tissue mass.

Of further interest is the problem of stating the dose in terms of a common scale. One can make a reasonable estimate of the absorbed dose in an organ. However, this merely specifies the energy absorbed and does not supply information about the effect of this dose on the organ. The absorbed dose which will give the same biologically equivalent dose is not the same for all types of radiation. Also, the lack of knowledge about the effects from low dose rates over long exposure times adds to the complex nature of the problem.
For purposes of protection work, the present method of dose determination, proposed by the ICRP in 1979,\(^1\) provides an adequate estimate. One must keep in mind that as knowledge is gained, more changes may be warranted. But the changes which may occur will only serve to reflect an attempt to obtain more precise results.

Given an amount of a radionuclide in an organ of the body, a reasonable estimate of the organ dose can be made. A much greater problem is to estimate the amount of a radionuclide in the organ which results from intake into the body. This involves a number of biological factors, many of which are not quantitatively known. Much work is needed in this area to improve the body of knowledge which now exists. At present, many assumptions must be made in order to relate an environmental concentration to which a person is exposed to the amount of the radionuclide which he receives as a result. In view of this fact, the ICRP has always taken the stand that review and modification are needed on a continuing basis. The present recommendations\(^1\) define a quantity called the Annual Limit on Intake (ALI). The ALI is the activity of a radionuclide which taken alone would irradiate a person to the limit set by the ICRP for each year of occupational exposure.

The method of calculation suggested in ICRP 30 is still not very precise. It can only yield a good estimate in cases where there are sufficient data to back up the assumptions made.\(^3,4\) But the value of these calculations lies in the fact that they provide a guide for evaluation of a given environmental concentration. For the purposes of protection work, there is a very special need for these guidelines. For this reason, the methods of estimating the derived air concentration (DAC) values are useful. The DAC is found by dividing the ALI for a given radionuclide by the volume of air inhaled in a working year (2400 m\(^3\)).\(^1\) On the other hand, one must be careful to note the conditions under which these values apply. In many cases, the use of these values without due regard to the assumptions which underlie them can lead to many pitfalls.
Let us focus on the nature of the problem. Assume an intake of a radionuclide which deposits in one organ and we wish to determine the dose. A knowledge of the mass, size, and shape of that organ is needed. For a certain radionuclide of known decay scheme and energy of emissions, one can calculate an initial organ burden. This will be the amount of the radionuclide in the organ which delivers the stated dose equivalent limit (see Section 7.C.3.a) for that organ. This limit may be based on the risk of stochastic effects or nonstochastic effects.  

There are many ways in which the radionuclide may reach this organ: inhalation, ingestion, diffusion through skin, and absorption through wounds or punctures. For most cases, only one mechanism is of concern for occupational exposure: that is, inhalation. Ingestion may be important in the considerations of the internal dose to the general public. The problem then is to relate a concentration in air (Bq/m³) to the amount of the radionuclide which ends up in the organ. Many factors affect the amount of the radionuclide which is deposited in a given organ.

From the standpoint of absorption into the body, the transportability (solubility) of the radioactive substance in body fluids plays a major role. Whether a given substance will be readily transported in body fluids or not is not easy to tell. However, as a mode of entry, ingestion is of concern only for transportable (soluble) radionuclides. Also, in the case of elements not required by the body, absorption by ingestion is poor, leading to rapid elimination of most of the material.

In those cases where absorption does occur, one is then concerned with the transfer of the substance to the organ, that is, only a fraction of the ingested material will go into the body fluids (blood stream). Of this, only a fraction will go into the organ of concern. These fractions are often poorly known. To add to the problem, a portion of the substance may end up in a large number of organs. In this case, more than just a single organ will have to be considered. Then, an organ will be irradiated not only by the radioactivity in that organ, but may also be irradiated by the radioactive emissions in nearby organs.
In regard to inhalation, both transportable and non-transportable matter must be treated. Other factors to consider are density and particle size. The movement and retention of matter in the respiratory tract are functions of the particle size and dissolution rate. Because of the complex relationships which exist, estimates of the transfer of radioactive matter by this mode of entry are poor. Of that released from the lung, a fraction will get into the body fluids (transfer compartment) and then be transported to a given organ.

Until recently, the ICRP used a general model for use in the calculations. In that model, deposition in the lung was based upon classification of materials as soluble or insoluble; with little regard for particle size and chemical form effects on clearance rates. A more sophisticated model utilizing dust deposition data and clearance fractions for portions of the respiratory tract has been used in the ICRP 30 methodology (see 9.D.3).

One can see that many factors are already present in an attempt to state intake parameters. But one must also treat the matter of elimination; although a substance is deposited in an organ, the time of stay in that organ is not indefinite. A fraction of this substance will be eliminated from the organ as time passes. Because the substance is radioactive, a certain fraction will decay per unit time, but also some of the material will be lost from the organ because of biological processes. Therefore, one must have knowledge of the net retention of a material in a given organ in order to proceed with the calculation. This means that one must have an elimination model for substances which are taken up by a given organ. The ICRP assumes that each organ can be treated as either a separate compartment or a number of compartments in which the elimination of a substance from each compartment proceeds at a constant rate. This exponential model for elimination means that each organ compartment has a half life for biological elimination of a given substance. This is only very roughly true and in some cases a power series or some other mathematical representation may be a better model. However, it provides a
simple means of obtaining a rough estimate of the retention in a certain organ.

Given the yearly intake rate for air (2400 m³ occupational), one can then solve for the DAC value. The calculation consists of finding the daily intake, over the working year, which would not result in an accumulation greater than the ALI. Then, the ICRP dose limit will not be exceeded. For the purposes of this methodology, it is immaterial whether the ALI is spread out over the year or acquired in a single incident within the year,\(^4\) except in the case of occupational exposure of women of reproductive capacity and pregnant women.\(^6\)

B. Reference Man

Throughout the above outline of the calculation of the DAC value, and its basis, many factors which concern biological processes in man were mentioned. Because one individual varies from the next, these factors differ from one person to the next. In order to obtain agreement throughout the world in regard to the calculations of internal dose, certain values have been agreed upon for these factors. The ICRP achieves this goal by the use of the "Reference Man" concept.\(^1\) The Reference Man represents a set of agreed-upon values for the many characteristics of man which are needed for internal dose calculations. These values can be found in ICRP Report 23.\(^9\) Since the values are intended to represent an average adult, no account is taken of the differences which occur among individuals. The published DAC values\(^1, 10-12\) have been calculated using the Reference Man data, dosimetric data for the relevant radionuclides contained in References 13-16, and the metabolic data for these radionuclides found in References 1, 10 and 11.

C. Internal Exposure - ICRP 2 Model

Previous estimates of internal dose equivalent were based upon the ICRP Publication 2 model,\(^8, 17\) in which it was assumed that the organ
retention could be represented by a single exponential term, that a specific organ could be considered to be the critical organ, that physical characteristics of the model (e.g., intake parameters, transfer fractions, tissue size and weight) could be represented by the "Standard Man" data, that spherical geometry could be assumed for organ shape, and that scattered radiation could be ignored. This basic model was used to generate inhalation dose factors that are contained in USNRC Regulatory Guide 1.109. For the basic dosimetry model, it was assumed that uniform deposition of the radionuclide in the organ occurs and that the energy emitted is equal to the energy absorbed, modified by a correction factor for the escape of photon energy from an organ of small dimensions. Integration of the dose-rate equation over a suitable time interval yielded the dose equivalent, H, delivered by the radionuclide deposited in the organ for the stated period. The system of dose limitation required that the specified annual dose equivalent to the critical organ not be exceeded.

The intake of radionuclides was then limited by establishing "Maximum Permissible Concentration" (MPC) values in air and water that would ensure that the dose-equivalent rate in the critical organ would not exceed that allowed by the dose-rate limitation value over a 50-year intake period. Values of the parameters needed in these calculations for the various radionuclides are contained in ICRP Publication 2, along with the MPC values. For purposes of dose limitations, the ratio of a given concentration to the MPC value for a specific radionuclide was considered as approximately equal to the ratio of the respective dose rates. This was generally true for radionuclides with relatively short retention times in the body and increasingly conservative for those radionuclides with longer retention times.

The dosimetric model incorporated the assumption that the radionuclide is uniformly deposited in an organ of spherical shape. For a large enough radius of the organ, one could assume that the energy emitted is equal to the energy absorbed. For an organ of small dimensions, the
previous assumption must be modified by a correction which accounted for
the fraction of photon energy that escapes. For this model, the dose
equivalent rate \( H(t) \) was given by:

\[
H(t) = \frac{51.2 \, \Sigma EF(RBE)n}{m} \frac{q(t)}{d} \quad \text{(rem)}
\]

where \( q(t) \) is the activity (\( \mu \text{Ci} \)) of the radionuclide in the organ at
some time \( t \), as determined by a single exponential retention function,
\( \Sigma EF(RBE)n \) is the effective absorbed energy per transformation in the
organ (MeV/dis), including the correction for escaping radiation, and \( m \) is
the organ weight (g). Values of the parameters needed for the various
radionuclides are contained in ICRP Publication 2.

Equation 9.1 indicates that the dose equivalent rate, \( H(t) \), is
directly proportional to the activity of the radionuclide present at any
time in the organ. To limit the organ dose rate, the buildup of activity
must be limited. The supply rate was limited such that the uptake rate of
the radionuclide by the organ balances the elimination rate at the end of
the 50-year continuous intake period. At this point, the activity in the
critical organ delivers the allowable organ dose equivalent rate.

For radionuclides with short retention times (rapid elimination
rates), the activity builds up quickly in the organ and rapidly approaches
an equilibrium value during continuous exposure. For these cases, the
activity in the organ will be approximately proportional to the MPC for
the radionuclide, and thus the MPC will be approximately proportional to
the permissible organ dose equivalent rate. The ratio of the dose
equivalent rate for some other concentration of the given radionuclide to
the permissible organ dose equivalent rate will be the same as the ratio
of that concentration to the MPC. This was the basis for using the
published MPC values as an index of the organ dose equivalent rate de-
levered by the radionuclide.

Some present standards are based upon the ICRP-2 model and appli-
cation of this methodology will be required until the regulatory agencies
officially adopt the new ICRP-30 models. However, the ICRP-30 model represents a revision and updating of the older ICRP-2 model, particularly with respect to the metabolic data, and its use as a model for internal exposure can be expected to be adopted. The NCRP has recommended adoption of this new methodology, and proposed revisions of USDOE, USEPA, and USNRC regulations have included the basic methodology of ICRP 26 and 30.

D. Internal Exposure - ICRP 30 Model

The ICRP has changed both its basic recommendations and revised the system of dose limitation. These revisions, as contained in ICRP Publication 26, reflect the availability of sufficient data about the effects of radiation for the Commission to estimate the risk per unit dose equivalent (H) with respect to fatal cancer in exposed people and to serious disease in the offspring of exposed people. Cancer and hereditary effects are referred to as stochastic effects (see Section 5.41) and the risks are assumed to be directly related to the dose equivalent, without threshold. So the probability of the effect occurring, rather than its severity, is a function of H. Other effects, called nonstochastic, are those in which the severity of the effect varies with H.

For the revised dosimetry model used in ICRP Publication 30, Part 1, it is assumed that organ retention can be represented by one or more exponentials, that the critical organ concept no longer applies and that one must account for contributions to the dose in one organ due to photons which are emitted from other body organs containing the given radionuclide, and that the physical characteristics and other parameters of the model can be represented by the "Reference Man" data. Spherical and other geometrical shapes are assumed for appropriate representations of body organs, and the scattered radiation contribution is accounted for by Monte Carlo calculations of the fraction of absorbed photon energy. More recent radiation transformation data for radionuclides and metabolic
data for elements and their compounds have been utilized. Revisions to the ICRP 2 models for the dosimetry of the respiratory tract, digestive system and bone are also used.

In this model, it is assumed that deposition in an organ is uniform. The total dose equivalent averaged throughout the tissue mass over 50 years after an intake is computed; then, an annual limit of intake (ALI) is determined for the particular radionuclide, whether for inhalation or ingestion. Values of the ALI for inhalation and ingestion can be found in the ICRP 30 Reports, Parts 1, 2 and 3, as well as the Supplements to these reports.1,10-15

If intake is only by inhalation or only by ingestion, the ICRP recommendations for dose limitation will be satisfied if the intake by either mode is less than the ALI for that mode. If both modes are involved, as well as several radionuclides contributing to each mode, the dose limitation will be met if:

\[ \sum \left( \frac{I_i}{\text{ALI}_i} \right)_\text{ING} + \sum \left( \frac{I_j}{\text{ALI}_j} \right)_\text{INH} \leq 1 \] 9.2

where \( I_i \) is the total annual intake by ingestion of radionuclide \( i \) and \( I_j \) is the total annual intake by inhalation of radionuclide \( j \). The terms \( (\text{ALI}_i)_\text{ING} \) and \( (\text{ALI}_j)_\text{INH} \) represent the respective ALI values for ingestion and inhalation.

In the use of the above inequality, one should be cautioned that the values of ALI for ingestion in the ICRP reports are for the total contribution from all ingestion pathways, such as food, drinking water, etc. To satisfy the Commission's limits, one must determine the contribution from each pathway and sum the results, which then can be compared to the ALI for ingestion of that radionuclide. Reference Man data9 can be used to estimate the consumption rates for the various ingestion pathways, and the intake \( (I_n,i) \) for a given pathway \( n \) is the product of the radionuclide concentration (Bq/kg) for that pathway and the yearly consumption (kg), and
\[ I_i = \sum_{n} I_{n,i} \]

For example, assume an individual has an annual intake of \(10^5\) Bq of a radionuclide whose ALI for inhalation is \(10^6\) Bq. In addition, the individual drinks water (730 L/y) containing an average concentration of the same radionuclide of 5 kBq/L, and consumes foodstuffs (520 kg/y) with a concentration of 100 Bq/kg. The ALI for ingestion is \(7 \times 10^6\) Bq. Will the ICRP dose limit be satisfied?

Using equation 9.3,

\[
I_i = \frac{2}{n=1} \left( 5 \times 10^3 \frac{\text{Bq}}{\text{L}} \right) \frac{1}{y} \left( 730 \frac{1}{y} \right) + 100 \left( \frac{\text{Bq}}{\text{kg}} \right) \left( 520 \frac{\text{kg}}{y} \right)
\]

\[= 3.702 \times 10^6\] Bq.

This is the amount ingested. Then, using equation 9.2,

\[
\frac{3.702 \times 10^6}{7 \times 10^6} + \frac{10^5}{10^6} = 0.63 \leq 1,
\]

so the dose limit is satisfied.

An obvious extension of the method used above can be made for the case involving external exposure as well as inhalation and ingestion. The inequality would then be stated

\[
\sum_{H_{\text{WB, L EXT}}} I_{i,k} + \sum_{I_{\text{ALI}}} I_{n,i} + \sum_{I_{\text{ALI}}} I_{j, i} \leq 1 \tag{9.4}
\]

\(H\)

\(I_{i,k}\) is the deep dose-equivalent index (defined as the maximum value of \(H\) that would occur in a 30 cm diameter tissue sphere [see Section 12.A.3.d]) for the kth contributor to the external exposure, and \(H_{\text{WB, L}}\) is the annual dose equivalent limit for uniform whole body irradiation, 0.05 Sv (5 rem) - see Section 7.C.3.a.
1. **Dose Limits**

As discussed in Section 7.C.3, the dose equivalent limits recommended by the ICRP are intended to prevent nonstochastic effects and limit the occurrence of stochastic effects to an acceptable level. To this end, an annual limit for occupational exposure of 0.5 Sv (50 rem) to all tissues [except 0.15 Sv (15 rem) to the lenses of the eyes]\(^{10}\) is deemed sufficient to prevent nonstochastic effects. This limit applies irrespective of whether the tissues are exposed singly or together with other organs. For stochastic effects, the limit on risk should be equal whether uniform whole body irradiation or non-uniform irradiation of several organs occurs. For non-uniform irradiation, the relationship expressed by equation 7.1 should be met:

\[
\sum_{T} w_T H_T \leq H_{WB,L},
\]

where \( w_T \) is a weighting factor given by the ratio of the stochastic risk in tissue \((T)\) to the total risk for uniform whole body irradiation (see Table 7.1), and \( H_T \) is the annual dose equivalent received by tissue \((T)\). In ICRP Publication 30, Part 1, the limits for intake of radionuclides are set by satisfying both of the conditions:

\[
w_T H_{50,T} \leq 0.05 \text{ Sv} \quad 9.5
\]

and

\[
H_{50,T} \leq 0.5 \text{ Sv} \quad 9.6
\]

where \( w_T H_{50,T} \) is called the **weighted committed dose equivalent** (committed effective dose equivalent) and \( H_{50,T} \) is the **committed dose equivalent**, which is the total dose equivalent averaged through a tissue \((T)\) in the 50 years following the intake of material in the given year.\(^1\)
EXAMPLE  Assume that the lung, bone surfaces, red marrow and kidney all receive committed dose equivalents of 0.15 Sv in a year. Are the basic limits met?

By inspection of equation 9.6, we see that the committed dose equivalent is not exceeded. Utilizing information from Table 7.1, in equation 9.6, we have

\[ \sum w_T H_{50,T} = 0.12(0.15) + 0.03(0.15) + 0.12(0.15) + 0.06(0.15) \]

\[ = 0.0495 \]

which is < 0.05 Sv and the limit is met.

It should be noted that in DOE Order 5480.11, the DOE proposes a modification of the ICRP system to arrive at an annual dose equivalent. The dose equivalent received in a year by a given organ from internal exposure is multiplied by the organ weighting factor \( w_T \), and this product is added to the external effective dose equivalent for that year to arrive at an annual effective dose equivalent.

This procedure follows the NCRP recommendation that the committed effective dose equivalent not be used for the evaluation of the consequences of radiation exposure in individuals.\(^4\) One may use the anticipated or prospective committed effective dose equivalent estimated for some planned practice for the purpose of design and control of the workplace. However, to demonstrate compliance with the exposure limits, the actual or retrospective assessed organ dose equivalent received by the individual during the year is used.

2. Dosimetric Model

The basic dosimetric model assumes uniform deposition in an organ. The total dose equivalent averaged throughout the tissue mass over 50 years after the intake is computed from:\(^1\)
\[ H_{50} = \sum_{i} \int_{M}^{D_{50,i}} \frac{Q_{i}}{dmdm} = \sum_{i} Q_{i} \frac{D_{50,i}}{M} \]

in which \( D_{50,i} \) is the total absorbed dose during 50 years following intake averaged throughout the tissue mass, \( M \), for each radiation type \( i \). The quality factor, \( Q_{i} \), for radiation of type \( i \) has one of three values in the ICRP model.\(^1\) These are:

- \( Q=1 \) for \( \beta \) particles, electrons and all electromagnetic radiation including \( \gamma \), \( x \) rays and bremsstrahlung.

- \( Q=10 \) for fission neutrons emitted in spontaneous fission and protons (Note that the ICRP has recommended \( Q=20 \) for fast neutrons).\(^19\)

- \( Q=20 \) for \( \alpha \) particles from nuclear transformations, for heavy recoil particles and for fission fragments.

When contributions for each radiation of type \( i \) from radionuclide \( j \) are summed, and if a number \( j \) of radionuclides, such as in a mixture, are contributing and need to be summed, and if several other source organs (\( S \)) are irradiating the given organ (\( T \)), the general expression becomes:

\[ H_{50,T} = 1.6 \times 10^{-10} \sum_{S} \sum_{i} \left[ U_{S} \text{SEE} (T+S)_{i} \right]_{j} \text{Sv}, \]

in which \( U_{S} \) is the number of transformations of \( j \) in a source organ \( S \) over 50 years following unit intake as obtained by integration of the retention function and SEE (MeV/g-trans) is the specific effective energy term for radiation of type \( i \), absorbed in \( T \) for each transformation in \( S \), modified by the quality factor. Values of \( w_{T}H_{50,T} \), \( H_{50,T} \), \( U_{S} \) and SEE (\( T+S \)) have been tabulated for radionuclides of 94 elements in the
Supplements to the various parts of ICRP Publication 30.\textsuperscript{13-15} These values are given for both inhalation and ingestion.

The limit on intake of a particular radionuclide is established by solving the following inequalities:

\[ I \sum_T w_T (H_{50,T} \text{ per unit intake}) \leq 0.05 \text{ Sv} \quad 9.9 \]

and

\[ I (H_{50,T} \text{ per unit intake}) \leq 0.5 \text{ Sv}, \quad 9.10 \]

whether for inhalation or ingestion. The annual limit on intake (ALI) is the greatest value of \( I \) which satisfies both of the above inequalities. The ALI\textsuperscript{1} is defined as the activity (Bq) of a radionuclide which taken alone would irradiate Reference Man to the limit set by the ICRP for each year of occupational exposure. Values for the ALI for inhalation and ingestion can also be found in the Supplements to ICRP 30.\textsuperscript{13-15}

The values of ALI developed in ICRP Publication 30 are based upon an occupationally exposed adult. The data and models described in the report are not recommended by the ICRP for estimating the dose equivalent (H) to members of a population based solely on the differences in mass of organs or magnitude of intake. At the end of Chapter 9 of ICRP-30, Part-1 (Reference 1), a bibliography has been included concerning the methodology of estimating H for different age groups.

With respect to members of the public, the ICRP recommends the application of the appropriate dose equivalent limit to the weighted mean whole-body dose equivalent of the critical group (that group expected to receive the highest dose equivalent), see paragraph 85 of ICRP Publication 26.\textsuperscript{6} In the calculation of the dose equivalent from intake of radionuclides, the metabolic differences between children and adults need to be taken into account.

In the data given in the ICRP 30 report for \( H_{50,T} \), ALI and DAC, allowance is made for the committed dose equivalent contributed by daughter buildup in the body from parent decay. The metabolic behavior of
all the radioactive daughter products is assumed to be the same as that of the parent. In the dosimetric data for any radionuclide, values are given for $U_s$ for the parent and for the daughters which have built up in the body for the 50 years following intake.

3. Respiratory Model

For inhalation, ALI values are listed for three classes of material based upon their relative retention in the pulmonary or deep section of the lung-D, W or Y. The classification is based upon a range of half times: $D < 10 \text{ d}, 10 \text{ d} < W < 100 \text{ d} \text{ and } Y > 100 \text{ d}$. These categories represent increasing retention with respect to half times, and the metabolic data, found in ICRP Publication 30, Parts 1, 2 and 3,\textsuperscript{1,10,11} classifies certain chemical forms of the material in terms of these categories. When the chemical form of the material is known, the proper choice of category--D, W or Y--can be made; otherwise, a conservative choice should be made.

The mathematical model used in the methodology is shown in Figure 9.1, and is taken from ICRP 30, Part 1.\textsuperscript{1} The model originally was developed by the Task Group on Lung Dynamics\textsuperscript{7} but has undergone changes in the values of some of the parameters over the years. Three major deposition regions are defined: the nasal-pharyngeal, tracheo-bronchial and the pulmonary region. The fractions initially deposited in these sections are $D_{N-P}$, $D_{T-B}$ and $D_{P}$, respectively, based upon an aerosol particle size of 1 $\mu$m. Deposition estimates for other sizes can be made using information in Reference 1.

As seen from the drawing in Figure 9.1, each of the three sections is divided into compartments. These compartments are associated with certain clearance pathways and have a defined half time $T$ for clearance with an associated fraction F of the material removed by that clearance pathway. Either $T$, or $F$, or both may have a different value as the retention category changes from D to Y. Also, for some retention categories, some of the compartments are not applicable. Compartments a, c, e
Figure 9.1 Mathematical model used to describe clearance from the respiratory system. The values for the removal half times $T_{a-i}$ and compartmental fractions, $F_{a-i}$, are given in the tabular portion of the figure for each of the three classes of returned material. (Reprinted with permission from Annals of the ICRP, Vol. 2, No. 314, ICRP Publication 30, Part I, Limits for Intakes of Radionuclides by Workers, Copyright 1979, Pergamon Press, Ltd.)
represent direct transfer to the body fluids (transfer compartment) by which some material may be transported to systemic organs and some excreted. Compartment h represents an indirect transfer to the body fluids through the lymph nodes. Also, for a Class Y aerosol, only a portion of the material (namely that from compartment i) is transferred by this pathway. The remainder is retained indefinitely in compartment j. The remaining compartments: b, d, f and g, transfer material to the gastrointestinal tract (GI tract). The metabolism of a radionuclide once it reaches either body fluids or the gastrointestinal tract is then governed by the metabolic model. Some material which goes through the GI tract may be routed back to the transfer compartment and then on to the systemic organs.

4. Gastrointestinal Tract Model

For ingestion, the ALI is based upon the fraction of material which is transferred from the GI tract to the systemic system ($f_1$). These values are also listed in the metabolic data and associated with the above categories. Again, in the absence of specific knowledge, one should choose a conservative value.

The model for the GI tract is based upon a biological model originally put forth by Eve. The GI tract is assumed to consist of 4 sections, each section being a single compartment with a defined half time for transfer of material. This model is shown in Figure 9.2, which is adapted from Reference 1. Included in the figure are the data from ICRP for the mass of the walls and contents of the various components of the system.

During ingestion, material passes through the four sections of the GI tract: stomach, small intestine, upper large intestine and lower large intestine. Each section is considered as a single compartment, so the entire system may be treated as a linear chain of first order differential equations. A portion of the material is assumed to be absorbed from the small intestine to the body fluids, shown by the transfer rate constant $\lambda_B$ in Figure 9.2. The particular values of
Table 9-18

<table>
<thead>
<tr>
<th>Section of GI tract</th>
<th>Mass of walls (kg)</th>
<th>Mass of contents (kg)</th>
<th>Mean residence time (day)</th>
<th>day^{-1}</th>
</tr>
</thead>
<tbody>
<tr>
<td>Stomach (ST)</td>
<td>0.15</td>
<td>0.25</td>
<td>1/24</td>
<td>24</td>
</tr>
<tr>
<td>Small Intestine (SI)</td>
<td>0.64</td>
<td>0.40</td>
<td>4/24</td>
<td>6</td>
</tr>
<tr>
<td>Upper Large Intestine (ULI)</td>
<td>0.21</td>
<td>0.22</td>
<td>13/24</td>
<td>1.8</td>
</tr>
<tr>
<td>Lower Large Intestine (LLI)</td>
<td>0.16</td>
<td>0.135</td>
<td>24/24</td>
<td>1</td>
</tr>
</tbody>
</table>

Figure 9.2 Mathematical model used to describe the kinetics of radionuclides in the gastrointestinal tract. (Reprinted with permission from Annals of the ICRP, Vol. 2, No. 3/4, ICRP Publication 30, Part I, Limits for Intakes of Radionuclides by Workers, Copyright 1979, Pergamon Press Ltd.)

\[ \lambda_B, \text{ are estimated from the metabolic parameter } f_1, \text{ which is the fraction of the stable element which reaches the body fluids following ingestion. The relationship used is} \]

\[ \lambda_B = \frac{f_1 \lambda_{SI}}{1-f_1}, \tag{9.11} \]

in which \( \lambda_{SI} = 6d^{-1} \) from the tabular data in Figure 9.2, and \( f_1 \)
is obtained from the metabolic data for the particular compound of the radionuclide. Of the material in the transfer compartment (body fluids), some is translocated to organs in the systemic system, as designated by the metabolic model, and the remainder is excreted. Material which is not cleared from the GI tract is excreted in the feces.

5. **Systemic Model**

When an inhaled or ingested radionuclide has been transferred to the body fluids (transfer compartment), its subsequent transfer to the compartments of the various tissues and organs of the systemic system is indicated schematically in Figure 9.3, taken from Reference 1. Whatever time is taken to transfer from a deposition site (lungs, GI tract) to a tissue by means of the body fluids, is represented by transfer compartment a. Unless otherwise stated in the metabolic model, the clearance half time for this compartment is 0.25 day. Each tissue which receives the radionuclide will consist of at least one, and maybe more, compartments.

**FROM GI TRACT AND RESPIRATORY SYSTEM**

![Diagram of systemic model](image)

**Figure 9.3** Mathematical model usually used to describe the kinetics of radionuclides in the body. (Reprinted with permission from Annals of the ICRP, Vol. 2, No. 3/4, ICRP Publication 30, Part I, Limits for Intakes of Radionuclides by Workers, Copyright 1979, Pergamon Press Ltd.)
Each compartment, in turn, has an associated elimination rate for excretion of the radionuclide. In this model, it is usually assumed that there is no feedback, or recycling, to the transfer compartment, either from tissue compartments or excretion routes. So, the metabolic models, with the exception of the alkaline earths, may be expressed as systems of first order differential equations with constant coefficients.

The final compartment model for a given radionuclide may be quite complex from the standpoint of the total number of compartments involved. Figure 9.4 shows the ICRP-30 Pu model for a Class Y aerosol, schematically. In this model, material entering the transfer compartment is distributed mainly to liver and bone \((F_L = F_B = .45)\), with the fraction \(.10\) going to all other tissues and early excreta. The fraction going to gonads is taken as \(3 \times 10^{-4}\) and \(1 \times 10^{-4}\) for males and females, respectively.

\[T_b = 0.25 \text{ d}\]

Figure 9.4  ICRP-30 Pu metabolic model, Class Y aerosol for inhalation or Class Y for ingestion. (Reprinted with permission from Annals of the ICRP, Vol. 2, No. 3/4, ICRP Publication 30, Part 1, Limits for Intakes of Radionuclides by Workers, Copyright 1979, Pergamon Press Ltd.)
With respect to the model in Figure 9.4, let us look at the case of inhalation. For a single intake of unit activity (1 Bq), the initial amount of material in lung compartment \( x \) is

\[
q_x(0) = D_x F_x, \tag{9.12}
\]

in which \( D_x \) corresponds to \( D_{N-P} \), \( D_{T-B} \) or \( D_P \), depending upon the location of compartment \( x \). \( F_x \) is the fraction of the inhaled material of the appropriate aerosol (Class D, W or Y) which deposits in compartment \( x \). The differential amount retained \( q_x(t) \) is given by

\[
\frac{dq_x(t)}{dt} = - \left( \lambda_x + \lambda_r \right) q_x(t), \tag{9.13}
\]

in which \( \lambda_x \) is the biological removal rate constant for compartment \( x \) and \( \lambda_r \) is the radioactive decay constant for the particular radionuclide. Solution of the above, for \( q_x(0) = D_x F_x \) gives

\[
q_x(t) = D_x F_x e^{-\left(\lambda_x + \lambda_r\right)t}. \tag{9.14}
\]

The contribution from the N-P region is not included in the ICRP respiratory dosimetry model, but the model does include the lymph compartment contributions. The retention function for the lung then, is the sum of the retention functions for compartments c through j (see Figure 9.1), for dosimetry purposes. This leads to a retention function containing 6 exponential terms for a Class Y material.

The material in the lung serves as a continuous supply source for the body fluids, lymph nodes and GI tract. The contribution from a given lung compartment \( x \), to the body fluids (BF) or transfer compartment may be expressed as

\[
\frac{dq_{BF}(t)}{dt} = \lambda_x q_x(t) - (\lambda_{BF} + \lambda_r)q_{BF}(t), \tag{9.15}
\]
which yields the solution
\[
q_{BF}(t) = \frac{\lambda D}{X X X} \cdot e^{-(\lambda + \lambda r)t} - e^{-(\lambda BF + \lambda r)t}
\]
when \( q_{BF}(0) = 0 \). Referring to Figure 9.4, additional sources of supply are from the lymph and GI tract. Each of these sources yields a contribution similar to that expressed in equation 9.16. Following combination of similar terms, the retention function contains 9 exponential terms for a Class Y material.

In turn, the body fluids act as a continuous supply source for the organs of the systemic system. The contribution from the body fluids to a particular organ will yield a differential equation similar in form to 9.15. Due to the chain nature of the model, a recursive relationship may be applied to successive differential equations in the chain to aid in generating the solutions.\(^{21}\)

The retention function for an individual organ; say, bone in the case of \(^{239}\)Pu and Class Y material, consists of the sum of ten exponential terms. Although expressions of this size become somewhat cumbersome for hand calculations, computer programs can perform the necessary computations swiftly and easily.

The final form of the retention function for a given organ relates the time course of the material in the organ back to a unit intake of the material. Integration of the retention function, over a 50-year time period following the intake, yields the total disintegrations, \( U_5 \), of the radionuclide in that organ (see 9.D.2).

6. **Bone Dosimetry Model**

The ICRP 30 model for bone deposition, distinguishes two types of bone depositors-surface and volume. For those radionuclides considered to be uniformly distributed over bone surfaces, the deposition is assumed
to be equally distributed in cortical and trabecular bone. For bone volume depositors, 80% of the deposition is assumed to be in cortical bone, 20% in trabecular bone. In the bone dosimetry model, cortical and trabecular bone are considered to be the source organs. The target sites are the bone surfaces (BS) and red marrow (RM), which are the radiosensitive tissues of concern in bone.

Bone dosimetry utilizes estimates of \( U_S \) in the source organs, cortical and trabecular bone, and computed values of the fraction of the emitted energy from these source organs which is absorbed in the target organs, bone surfaces and red marrow. Fractions are estimated for \( \alpha \), \( \beta \), and photon radiations emitted by the source. This dosimetric information is contained in the quantity \( \text{SEE}(T\text{-}S) \) of equation 9.8, and is given in the dosimetric information.\(^{13-15}\) The specific effective energy \( \text{SEE} \) is discussed in Section 9.F.

7. **Subersion Exposure**

The problem of setting DAC values for inert gases involves primarily external exposure rather than internal exposure. These gases are poorly absorbed by the body, so that only a small portion of the gas in a cloud of large volume is held in the body. The ICRP has considered three sources of exposure: the external dose due to submersion, the internal exposure due to absorbed gas, and the lung dose due to contained gas. They conclude\(^1\) that the external exposure received in the cloud will be of most importance, and one may ignore the other contributions.

One is then concerned with the exposure of an individual essentially surrounded by a semispherical cloud of radioactive gas. For radioactive noble gases, other than radon and thoron which are treated in Reference 12, the exposure is limited by the DAC (submersion) which is based upon 2000 exposure hours in a working year. Details of the methods used to derive the DAC values are discussed in Chapter 8 of Reference 1.
E. Absorbed Dose Computations

1. MIRD Method

The method of calculation used to determine absorbed doses to tissues in the ICRP methodology has been adapted from the MIRD (Medical Internal Radiation Dosimetry Committee) method. This model was developed for application to medical dosimetry, in which it is often desirable to know the dose to one organ from radioactivity source in another nearby organ. Because of the quantities of radionuclides used in some clinical studies, one needs to be concerned with irradiation of nearby organs from the radionuclide being used in the study of a particular organ. A review of this methodology can be found in ICRU Report 32, along with a discussion of clinical applications.

In the MIRD methodology, one distinguishes between target regions and source regions. Then, the concept of the absorbed fraction may be introduced. The absorbed fraction \( \Phi(T-S) \) is the fraction of the energy emitted in a source region \( S \) that is absorbed in a target region \( T \). Target and source regions may be completely separated; they may overlap, or they may coincide. One may define the specific absorbed fraction \( \Phi(T-S) \) as the absorbed fraction divided by the mass of the target region.

The absorbed dose in a target region depends upon a number of physical and biological parameters: activity, radioactive half life, decay scheme data, the location, mass and shape of the target region, the absorbed fraction and the temporal and spatial distribution of activity in the source region. The specific absorbed fraction depends upon the activity distribution in \( S \). The activity is assumed to be uniformly distributed in the source organ. The specific absorbed fractions for Reference Man were computed by the Monte Carlo method for a representative phantom. Values can be found in Appendix I of Reference 9.

Assume a radionuclide that decays with the emission of one type of radiation only. The activity of this radionuclide is uniformly
distributed in some source region, and the emissions from the source irradiate the target region. The mean absorbed dose, \( \bar{D} \), in the target region, is given by the product of five terms:

1. the number of transformations (disintegrations) in the source during the time of interest, \( \bar{A} \);
2. the mean number of ionizing particles per nuclear decay, \( n \);
3. the mean energy per particle, \( E \);
4. the absorbed fraction, \( \phi(T-S) \); and
5. the reciprocal of the target mass, \( m^{-1} \).

This may be written

\[
\bar{D} = \bar{A} n E \phi(T-S)m^{-1}
\]  

9.17a

when all quantities are expressed in SI units. The product \( nE \) is usually written as \( \Delta \), the mean energy emitted per disintegration and \( \phi \) divided by \( m \) is the specific absorbed fraction, so

\[
\bar{D} = \bar{A} \Delta \phi (T-S)
\]  

9.17b

Now, if we extend the concept to a source with a number of different emissions, each of which has an associated specific absorbed fraction, the expression becomes

\[
\bar{D} = \bar{A} \sum_i \bar{\Phi}_i (T-S).
\]  

9.17c

In the equation, \( \bar{A} \), called the cumulated activity, is a function of time and is obtained by the integration of the retention function over the time period of interest. The other quantities under the summation are generally
grouped together and tabulated as the "S" value. These are given in units of rad/μCih in Reference 26 and can be converted to nGy/dis by dividing S by 13.32.

If one substitutes the activity value A into the above equation, the dose rate, \( \dot{D} \), is obtained

\[
\dot{D} = A \sum_i \Delta_i \Phi_i(T+S).
\]

9.18

**Example** A sample of \( 4 \times 10^4 \) Bq (dis/s) of \(^{131}\)I is uniformly distributed in the thyroid. Compute the initial absorbed dose rate, \( \dot{D} \), to the thyroid. From Reference 25, p. 185, \( S \) for \(^{131}\)I=2.2×10^{-2} rad/μCih with thyroid as target and source. So,

\[
\frac{\text{nGy}}{\text{dis}} = \frac{S}{13.32} = \frac{2.2 \times 10^{-2}}{13.32} = 1.65 \times 10^{-3}
\]

and

\[
\dot{D} = 4 \times 10^4 \frac{\text{dis}}{\text{s}} \left(1.65 \times 10^{-3} \frac{\text{nGy}}{\text{dis}}\right) = 66 \frac{\text{nGy}}{\text{s}} = 6.6 \times 10^{-8} \frac{\text{Gy}}{\text{s}} \approx 23.8 \text{ mrad/h}.
\]

If A is expressed in Bq, \( \Delta_i \) in MeV/dis and \( \Phi_i(T+S) \) in kg^{-1}, the absorbed dose rate in Gy/h will be

\[
\dot{D} = 5.76 \times 10^{-10} A \sum_i \Delta_i \Phi_i(T+S) \text{ Gy/h},
\]

9.18a

in which the constant includes the factors to convert MeV to J and s to h. Again, the S value from Reference 25, when multiplied by 469, can be substituted for \( \sum_i \Delta_i \Phi_i(T+S) \) in equation 9.18a. That is,
\[ D = 5.76 \times 10^{-10} \times 4 \times 10^4 \times 469 (2.2 \times 10^{-2}) = 2.377 \times 10^{-4} \, \text{Gy/h} \]
\[ = 6.6 \times 10^{-8} \, \text{Gy/s} \] (23.8 mrad/h).

Since there may be a number of source organs involved because a radionuclide will distribute in several tissues, the total dose, or dose rate, is obtained by a summation of the individual contributions.

For the case of a uniformly distributed source in a large volume, one can assume that the energy emitted per unit mass will be equal to the energy absorbed per unit mass. Then, the absorbed dose rate is

\[ D = C \Delta_i, \]

in which \( C \) is the activity per unit mass (Bq/kg) and \( \Delta_i \) is the total mean energy emitted per disintegration. Since all energy is absorbed, the absorbed fraction will equal 1. Values of \( \Delta_i \) for a number of radionuclides, many of which are of medical interest, can be found in Appendix A of Reference 2.

a. **Penetrating and Nonpenetrating Radiations**

In some cases, the fraction of energy lost outside of a given tissue volume by the emitted radiation will be small enough to ignore. This is generally the case for \( \alpha \) and \( \beta \) radiation. For photons or neutrons though, a large fraction of the emitted energy may be lost outside of a given tissue volume. The term penetrating radiation is used for those radiations which may lose a significant fraction of their energy outside of an organ. The term nonpenetrating is used for \( \alpha \), most \( \beta \) and those photons of E<15 keV. In the case of nonpenetrating radiation, the absorbed fraction is taken as 1 if the source and target regions are the same. The value of \( \phi (T-S) \) is 0 if the source and target regions are separate.\(^2\)
b. *Cumulated Activity*

The activity as a function of time in a tissue is expressed by the retention function for that radionuclide in the particular organ. For the general case, the activity as a function of time can be written

\[ A(t) = e^{-\lambda t} \sum_{i} k_i e^{-\lambda_i t}, \]  \hspace{1cm} 9.20

in which \( e^{-\lambda t} \) is the term expressing radioactive decay and is common to all the compartments, \( k_i \) is the initial value of the activity for each compartment \( i \), and \( e^{-\lambda_i t} \) expresses the biological elimination from the organ. Note that \( k_i \) may also be expressed as a fraction of unit activity which is deposited in the \( i \)th compartment. The cumulated activity, \( \tilde{A} \), is obtained by integration of the above expression, and gives

\[ \tilde{A}(t) = \sum_{i} \frac{k_i}{\lambda + \lambda_i} (1 - e^{-(\lambda + \lambda_i)t}), \]  \hspace{1cm} 9.21a

which for \((\lambda + \lambda_i)t\) large enough, becomes

\[ \tilde{A}(t) = \sum_{i} \frac{k_i}{\lambda + \lambda_i} \]  \hspace{1cm} 9.21b

For the case in which the retention function is expressed by only a single exponential,

\[ A(t) = A_0 e^{-(\lambda + \lambda_1)t} = A_0 e^{-\lambda_{\text{eff}1} t} \]

\[ = A_0 e^{-\frac{\ln 2}{T_{\text{eff}1}} t} \]

and

\[ \tilde{A}(t) = \frac{A_0}{\lambda + \lambda_1} = \frac{A_0}{\lambda_{\text{eff}1}} = \frac{A_0 T_{\text{eff}1}}{\ln 2} = 1.443 \frac{A_0 T_{\text{eff}1}}{} \]  \hspace{1cm} 9.23
In the above expression, $T_{\text{eff}}$, is the effective half life in the particular compartment 1 and is found from

$$T_{\text{eff}} = \frac{\ln 2}{\lambda + \lambda_1} = \frac{\ln 2}{\lambda_{\text{eff}}_1} = \frac{T_1}{T + T_1}$$

9.24

**EXAMPLE** The activity deposited in a certain organ, with a single exponential retention function of biological half life 6d, is 10 MBq ($10^7$ dis/s) of half life 3d. Find the cumulated activity to complete decay.

$$A(t) = e^{-\lambda t} \sum k_i e^{-\lambda_i t} = e^{-\ln 2 t} \left( 10^7 e^{-\frac{\ln 2}{3} t} \right)$$

$$= 10^7 e^{-\left( \frac{\ln 2}{3} + \frac{\ln 2}{6} \right) t}$$

is the retention equation and

$$\tilde{A}(t-\infty) = \frac{10^7}{\frac{\ln 2}{3} + \frac{\ln 2}{6}} = \frac{10^7}{0.5 \ln 2} = 2.885 \times 10^7 \text{ dis.}$$

Note from equation 9.24, $T_{\text{eff}} = \frac{3.6}{3+6} = \frac{2d}{3+6}$ and $\tilde{A}(t-\infty) = 1.443 \times 10^7$ gives the same answer.

Examples of absorbed dose calculations using the MIRD method can be found in Reference 2, Appendix C; Reference 24, Chapter 10 and Reference 26, Chapter 6.

The modification of the MIRD system to incorporate the quality factor and to obtain the specific absorbed fraction by transport methods using Monte Carlo calculations is attributed to Snyder and his collaborators. In addition to the tabulation of "S" in MIRD Pamphlet 11, Snyder, et al. also published a tabulation of dose equivalent per $\mu$Ci-day for various radionuclides. This formed the basis of the internal dosimetry model which is used in the new ICRP methodology.
F. Specific Effective Energy (SEE)

The ICRP has utilized the Snyder modifications to the model for absorbed dose calculations based upon the MIRD method in its new internal dosimetry methodology. In the supplements\textsuperscript{13-15} to ICRP 30 Parts 1, 2 and 3, data is given for the Specific Effective Energy, defined by

\[ \text{SEE}(T,S)_j = \sum_i \frac{Y_i E_i \text{AF}(T,S)_{ji} Q_i}{M_T} \text{ MeV} \] \hspace{1cm} 9.25

in which \( \text{SEE}(T,S)_j \) is the specific effective energy absorbed in target \( T \) from radionuclide \( j \) in source organ \( S \). The summation on the right side of the equation is taken over all radiation types \( i \) emitted in each disintegration of radionuclide \( j \). The individual terms are:\textsuperscript{1}

- \( Y_i \) is the yield of radiations of type \( i \) per disintegrations,
- \( E_i \text{(MeV)} \) is the average or unique energy of radiation type \( i \) as appropriate;
- \( \text{AF}(T,S)_{ji} M_T \) is the specific absorbed fraction as defined in the MIRD method, and can be found in ICRP 23\textsuperscript{9} for photons, but is expressed in units of \( 	ext{g}^{-1}(10^{-3} \text{ kg})^{-1} \);
- \( Q_i \) is the quality factor and has the values discussed in 9.D.2

Values for \( Y_i \) and \( E_i \) can be found in ICRP 38.\textsuperscript{16}

Since there may be a number of target and source organs for a particular radionuclide, the SEE values in the Supplements are arranged in tables with the columns representing source organs and the rows representing target organs. For example, Table 9.1, is an excerpt from Reference 13, showing the general arrangement of the SEE information. As might be expected, the SEE value is largest when the target and source
organs are the same. Note that the values shown refer only to the designated radionuclide, and do not include any daughter contribution. The values for any daughter radionuclides are given in a separate table.

Table 9.1 - SEE (MeV per gram per transformation) of Sr-89

<table>
<thead>
<tr>
<th>TARGETS</th>
<th>SOURCES</th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>LUNGS</td>
<td>ULI CONTENT</td>
<td>LLI CONTENT</td>
<td>CORT. BONE</td>
<td>TRAB. BONE</td>
</tr>
<tr>
<td>Gonads</td>
<td>4.8E-11</td>
<td>2.3E-09</td>
<td>4.6E-09</td>
<td>2.8E-10</td>
<td>2.8E-10</td>
</tr>
<tr>
<td>R.Marrow</td>
<td>3.4E-10</td>
<td>5.6E-10</td>
<td>8.3E-10</td>
<td>8.6E-10</td>
<td>1.4E-04</td>
</tr>
<tr>
<td>Lungs</td>
<td>5.8E-04</td>
<td>1.1E-10</td>
<td>4.3E-11</td>
<td>2.8E-10</td>
<td>2.8E-10</td>
</tr>
<tr>
<td>Bone Surf.</td>
<td>2.7E-10</td>
<td>1.9E-10</td>
<td>2.8E-10</td>
<td>7.3E-05</td>
<td>1.2E-04</td>
</tr>
<tr>
<td>ULI Wall</td>
<td>1.2E-10</td>
<td>1.3E-03</td>
<td>1.1E-09</td>
<td>2.2E-10</td>
<td>2.2E-10</td>
</tr>
<tr>
<td>LLI Wall</td>
<td>2.6E-11</td>
<td>9.1E-10</td>
<td>2.2E-03</td>
<td>3.0E-01</td>
<td>3.0E-10</td>
</tr>
</tbody>
</table>

To illustrate the computation of SEE(T-S)j, let us assume that the source of $^{89}$Sr is located in the lungs and the target is the lungs. From page 195 of ICRP 38, the dosimetric data for $^{89}$Sr gives a nonpenetrating component $\beta^-$ of $Y_i E_i = 5.83 \times 10^{-1}$ and a photon component ($E_i = 0.9091$ MeV) of $Y_i E_i = 8.45 \times 10^{-5}$. The daughter of $^{89}$Sr is stable. Utilizing ICRP 23, Appendix I, we estimate the specific absorbed fraction for a 0.9 MeV photon for lung as both the source and the target (page 453 of ICRP 23) as about $4.6 \times 10^{-5}$. The specific absorbed fraction for the beta is $1/M_T = 10^{-3}$ and $Q_i$ is 1 for both $\beta$ and photons. Utilizing equation 9.25,

$$\text{SEE(Lung-Lung)} = \frac{\sum_{i=1}^{2} Y_i E_i AF \text{(Lung-Lung)}_i Q_i}{M_T}$$

$$= 5.83 \times 10^{-1} \times (10^{-3})(1) + 8.45 \times 10^{-5} \times (4.65 \times 10^{-5})(1)$$

$$= 5.83 \times 10^{-4} = 5.8 \times 10^{-4} \text{ MeV g trans}$$
This result gives us one entry in Table 9.1, for lungs as the source and lungs as the target. For the other source organs, specific absorbed fractions for the $\beta$ component are 0, so the computation only involves the photon component. Using ICRP 23, the specific absorbed fractions for the other source organs can be found and SEE computed. This will fill out the row with lungs as the target. Then, one can choose a different target organ and begin all over again to fill out that row! As the decay scheme becomes more complicated, one has to deal with more radiation types than in this simple example for $^{89}$Sr. Again, although hand computations become tedious, a computer can handle this task easily. Moreover, the ICRP Supplements 13-15 already contain this information for most radionuclides of interest to health physicists.

Committed Dose Equivalent $H_{50,T}$

The committed dose equivalent, as expressed by equation 9.8, is the product of two factors: $U_s$, the total number of transformations of the radionuclide in a source organ $S$ over a period of 50 years following intake, and $\text{SEE}(T+S)$, the energy absorbed in the target, modified by the quality factor, for each type of radiation emitted per transformation in $S$. For a radiation type $i$, the committed dose equivalent is

\[
H_{50}(T+S)_i = U_s \text{(trans.)} \times 1.6 \times 10^{-13} \left( \frac{J}{\text{MeV}} \right) \text{SEE}(T+S)_i \left( \frac{\text{MeV}}{\text{kg trans}} \right) \\
\times 10^3 (\text{Sv}) \]

\[
- 1.6 \times 10^{-10} \ U_s \ \text{SEE}(T+S)_i \ \text{Sv} \]

Multiply by 100 in equation 9.26 to obtain $H_{50}(T+S)_i$ in rem. Equation 9.26 is then applied to all types of radiations emitted by the radionuclide. This requires the summation of $\text{SEE}(T+S)_i$ over all $i$ radiations emitted by the radionuclide. If a mixture of radionuclides is taken in, then a summation over the $j$ radionuclides is required. In
addition, there may be several source organs involved so that the contributions will need to be summed over all S, which leads to equation 9.8,

$$H_{50,T} = 1.6 \times 10^{-10} \sum_{S} \left[ \sum_{i} U_{S} \sum_{j} \text{SEE}(T+S)_i \right]_{j} \text{ Sv}$$

From our example for $^{89}\text{Sr}$ above, i=2 radiations ($\beta$ and $\gamma$) and j=1 radionuclide ($^{89}\text{Sr}$), but S=6 (lungs, ULI content, LLI content, cort. bone, trab. bone and total body) from Table 9.1. In order to find $H_{50,T}$, equation 9.8 reduces to

$$H_{50,T} = 1.6 \times 10^{-10} \sum_{S} U_{S} \sum_{i} \text{SEE}(T+S)_i$$.

The summation over $i$ has already been done and these are the tabulated values in Table 9.1. What remains is to determine the $U_{S}$ values which are found on page 82 of Reference 13. Assuming a Class Y aerosol was involved in a unit intake by inhalation, our expression becomes

$$H_{50,\text{Lungs}} = 1.6 \times 10^{-10} \left[ 8.9 \times 10^5 (5.8 \times 10^{-4}) + 2.3 \times 10^4 (1.1 \times 10^{-10}) 
+ 4 \times 10^4 (4.3 \times 10^{-11}) + 5.7 \times 10^3 (2.8 \times 10^{-10}) 
+ 4.6 \times 10^3 (2.8 \times 10^{-10}) + 5.5 \times 10^3 (8.3 \times 10^{-6}) \right]$$

$$= 8.3 \times 10^{-8} \text{ Sv}$$

So, for each Bq (dis/s) of intake, the $H_{50,\text{Lung}} = 8.3 \times 10^{-8} \text{ Sv}$ (8.3x10^{-6} rem). The weighted committed dose equivalent is then found by multiplying $H_{50,\text{Lungs}}$ by $w_T$ for lungs, which is 0.12. This gives 1x10^{-8} Sv. It turns out that for this particular situation (Class Y intake by inhalation), only the lung dose is important. Solving for the ALI, using equations 9.9 and 9.10, gives

$$I \leq \frac{0.05}{1 \times 10^{-8}} = 5 \times 10^6 \text{ Bq}.$$ 

and

$$I \leq \frac{0.5}{8.3 \times 10^{-8}} = 6 \times 10^6 \text{ Bq}.$$
In this case, the choice for the ALI would be $5 \times 10^6$ Bq since this value will satisfy both inequalities. The DAC is then given by the

$$\frac{6}{2400} = \frac{5 \times 10^6}{2400} = 2.0 \times 10^3 \text{ Bq/m}^3,$$

rounded to two places.

More examples of the application of the ICRP 30 methodology can be found in Reference 28. References 29 and 30 discuss the use of the methodology for problems of assessing compliance with standards and arriving at cleanup criteria.

H. Internal Intake Assessments

As mentioned earlier in this section, the determination of the amount of a radionuclide which deposits in a specific organ is not an easy task. A multitude of factors affect the deposition of a given radionuclide in a specific organ. Many biological transfer fractions are still poorly known. In terms of the measurement of the amount of a radionuclide which deposits in an organ, two general measurement approaches are used. One utilizes in vitro bioassay samples. In the in vitro bioassay method, the amount of the radionuclide in an organ is estimated by identifying, and measuring, if possible, the quantity of a specific radionuclide in samples that are excreted, secreted or removed from the body and which may include urine, blood, breath, sputum, sweat, saliva, hair, nasal discharges and feces. For some radionuclides, this indirect method allows the identification of the amount of a radionuclide in a specific organ, but to compute the committed dose equivalent requires reliance on a metabolic model. That is, the determination of the amount of a radionuclide in an organ at any particular time allows one to compute only the initial dose rate in that organ, relative to the time of measurement. The dose commitment and/or the annual dose equivalent may only be calculated if one knows the necessary metabolic data (retention function). The generalized metabolic data which is available in ICRP 30 and the dosimetric data in
the Supplements, allows one to compute the dose rate in an organ, and the
dose commitment, but then only to the model, which is represented by
Reference Man. For any individual, the actual dose commitment will differ
from this value depending upon the individual's characteristics. Individual data obtained from excretion analysis should be used to obtain
a more realistic estimate of the organ burden and the retention function.\(^{31}\)

By integration of the retention function for a given organ over a
suitable time duration (i.e., one year), the total number of transforma-
tions per unit activity during the year can be determined. Multiplying
this result by the initial activity in the organ and utilizing the ICRP
tables for SEE values, allows one to estimate the annual dose equivalent
in that organ by use of an expression similar to equation 9.26.

One further difficulty in bioassay analysis, is that the sample
which is collected may represent contributions from a number of organs in
which the radionuclide has deposited. One may sometimes relate fractions
of the total activity in the in vitro biological sample to a few organs
which are excreting the radionuclide. However, this becomes increasingly
difficult when the radionuclide is prone to deposit in many organs of the
body.

Although bioassay has disadvantages as a quantitative method,\(^{31,35}\)
it is of value as a qualitative indicator of the effectiveness of the
control methods. However, to be effective, the bioassay program needs the
cooperation of the sampled individual. That is, if one is unwilling to
submit the samples, the method will be ineffectual. For samples which are
collected, care must be taken to avoid contamination of the sample. More
discussion on the aspects and features of a suitable bioassay program can
be found in References 33-35, and 37.

Another of the bioassay methods referred to above is in vivo
measurements by external whole body counting (WBC). In this direct method,
the gamma rays emanating from a radionuclide in a given organ can be
counted and their energies analyzed by pulse height analysis. The
identification of the nuclide can be accomplished rather easily, but the
actual location of the radionuclide material may not be so definite. Since radionuclides may deposit in a number of different organs, the response of the WBC may be a composite of the contributions from a number of organs. This makes the assignment of the amount in a given organ somewhat arbitrary. In addition, the quantification of the amount of a radionuclide in a given organ, even if only one is involved, is not an easy task. Because of the differences in chest wall thickness, and organ shape and weight, the absorption and scattering properties may vary from individual to individual.

Even on the presumption that the radionuclide is located in a single organ, so that identification and quantification are simplified, the computation of the dose commitment will still require the use of the retention functions. Moreover, calibrations must be performed, using phantoms, in order to adequately assess the quantity of a radionuclide which is deposited in a given organ. This is because of the absorption and scattering of the photon radiation as it passes through the body.

A disadvantage of the WBC methodology is that the radionuclide to be identified must emit sufficiently high energy photons so that radiations will reach the detector. In reactor applications, many of the radionuclides of interest emit photons which are energetic enough to meet this criterion. However, radionuclides such as $^{239}$Pu, emit only low energy photons (approximately 17 keV) in quantities large enough to be detected. These low energy photons are almost completely absorbed by the chest wall so that the detection of $^{239}$Pu in lungs by this method is poor unless the concentration is very high.

The advantage of WBC over in vitro bioassay is the ability to obtain results much more quickly. Bioassay samples require a certain time for collection, followed by an even longer time for analysis, since sample preparation may require a significant amount of time. So, in a suspected incident, one generally cannot get an accurate estimate of the intake within a short time following the incident, if excreta samples are used. On the other hand, whole body counting, following a suspected incident, should be performed as soon as possible after the incident. This
will ensure that the majority of material taken in by inhalation will still be residing in the lungs, rather than distributed in many organs. Moreover, for radionuclides which distribute in only a few discrete organs, this method will give a better estimate of the quantity of a radionuclide in a particular organ, even if a count is taken much later.

Care needs to be taken so that personnel who are involved in a suspected incident, and are going to be counted, do not have external contamination, such as on clothing or skin. Not only does this give erroneous results with respect to the whole body count, but it also may result in contaminating the WBC, which is very sensitive to any increased background activity.

Individual organ counts can also be accomplished by this methodology. One of the more notable applications in this respect is thyroid counting for radiiodine. Solid state detectors, notably Cd-Te, have been successfully used for monitoring of cuts and wounds.

One other method used to estimate intake of radionuclides is to use concentration measurements as determined by air samples. The advantage of this method is its simplicity. No special samples or analyses are required. The air sample is typically obtained on a filter paper and counted on relatively simple equipment. If the identity of the radionuclide is not known, the analysis then needs to be performed on more sophisticated equipment to identify the radionuclide in addition to the activity determination. Once the concentration in air has been determined, one can estimate the intake of radionuclides by use of the presumed time that the individual is exposed to the radionuclide concentration. This method of estimating intakes is not recommended by DOE except for unusual circumstances in which bioassay data is unavailable or inadequate.

Although simpler than either of the first two methods, this method has several disadvantages relative to the others. As is discussed in Section 14.D, the placement of the sampler is important. The concentration that the worker breathes can be significantly different from the concentration measured by the sampler. In addition, the worker may be
moving around in the relevant area so that he may be exposed to a gradient of concentrations. Moreover, the concentration in the given region may not be constant, at all, during the time of exposure. These factors lead to a significant uncertainty in the estimation of the intake. So, any intake estimates need to be supported by bioassay determinations.

Once the material enters the body, there is more uncertainty with respect to the distribution of the material in the body. That is, material entering by inhalation goes to the lungs, is distributed to the blood and then deposited in the various organs. The fractions involved are also poorly known as are the metabolic parameters.

If the material is not analyzed for particle size, solubility, and chemical properties, more uncertainties are introduced. There is no one set of biological factors which can be universally applied to convert air concentration to the amount deposited in a given organ. 7

As can be seen from the above discussion, the determination of the organ burden is somewhat difficult. When applicable, whole body counting is the desirable method since it will often yield a better estimate of the quantity of the radionuclide in a particular organ. However, even given the amount in the organ, the initial dose equivalent estimate must generally be determined using the metabolic data from ICRP 30 or some other specific metabolic model. The final estimate can be obtained by using the actual excretion data from the individual bioassay samples.

REFERENCES


27. Snyder, W.S., et. al., Tabulations of Dose Equivalent Per Microcurie-Day for Source and Target Organs of an Adult for Various Radionuclides, ORNL-5000, Parts 1 and 2, NTIS, Springfield, VA (1975).


BIBLIOGRAPHY


QUESTIONS

9.1 Name four factors of main concern that enter into internal radiation dose calculations.

9.2 How do the distributions, uniform and nonuniform, affect the accuracy of the calculation of internal dose. Give an example.

9.3 Explain the difference between the absorbed dose and biologically equivalent dose.

9.4 Define Annual Limit of Intake (ALI).

9.5 What is the Derived Air Concentration (DAC)? How is it useful?

9.6 Name some of the ways in which a radionuclide may reach an organ. What are two important ways of concern?

9.7 What characteristic of a radionuclide determines its absorption in the body?

9.8 What factors determine the movement and retention of matter in the respiratory tract?

9.9 What are the several processes of concern in estimating the net retention of a radioactive material in a given organ of the body?

9.10 Why is it necessary to invoke the concept of "Reference Man"?

9.11 What are some of the major assumptions of ICRP-2 model?

9.12 Explain the concept of "critical organ."

9.13 What is the "MPC"?

9.14 In what circumstances will the activity in the organ be approximately proportional to the MPC?

9.15 In the ICRP-30 model, how is the concept of risk related to dose?

9.16 Explain how dose limitation from inhalation and ingestion by several pathways is to be accounted for in the ICRP-30 model.
9.17 What is deep dose equivalent index? How is it used in the ICRP-30 model dose limitation inequality?

9.18 In the respiratory model, how many classes of materials are listed and what is the basis for this classification?

9.19 What is the basis for the ALI in the case of ingestion?

9.20 What are two types of bone depositors in the ICRP-30 bone dosimetry model?

9.21 What are three sources of exposure considered for inert gases? Which type of exposure is considered most important?

9.22 From which source are ICRP-30 absorbed dose computation methods adapted?

9.23 What is the specific absorbed faction?

9.24 Explain the term "mean absorbed dose" in the target region.

9.25 How are penetrating and non-penetrating radiations classified with respect to the tissue volume?

9.26 What is "effective half-life"? How is it related to the cumulated activity?

9.27 What are the general measurement approaches in assessing the internal intake?

9.28 Mention some of the disadvantages of bioassay method as a tool for internal intake assessment?

9.29 What are some of the advantages and disadvantages of whole body counting?

9.30 What uncertainties are involved in the measurement of concentrations by the air sampling method as a tool to assess internal intake?

**PROBLEMS**

9.1 An individual has an annual intake of 1000 Bq of Po-210, whose ALI for inhalation is $2 \times 10^4$ Bq. In addition, the individual drinks water, 700 l/y, containing an average concentration of the same nuclide of 50 Bq/l and consumes food stuffs, 500 kg/y, with a concentration of 100 Bq/kg. The ALI for ingestion is $1 \times 10^5$ Bq, will the ICRP dose limits be satisfied?

Answer: $0.9 < 1$, YES
9.2 A radiochemist suffers an accidental exposure to $^{131}$I. It was found that 740 kBq were deposited in his/her body, of which 220 kBq are in the thyroid gland and the rest distributed in the remainder of the body. Using bioassay and body scanning data, the thyroid dose was estimated to be 370 mGy and the whole body dose, 0.5 mGy. Calculate the chemists' effective dose equivalent. Was this exposure within the ICRP criteria? (Hint: see Table 7.1 for relevant data.)

Answer: 11.6 mSv. Yes

9.3 Calculate the cumulated activity $\tilde{A}$ in organ "L" of the body for a radionuclide, if the initial administered activity is $3.7 \times 10^7$ Bq. Assume 85% of the activity is retained in the organ and the effective half life is equal to the physical half life (6 h).

Answer: $9.8 \times 10^{11}$ Bq

9.4 In problem 9.3, if 50% of the activity in the organ "L" is eliminated with a half life of 2 h and 50% with a half life of 3 h, what is the cumulated activity in the organ for the same initial activity?

Answer: $2.86 \times 10^{11}$ Bq

9.5 $^{24}$Na decays by $\beta$ emission to $^{25}$Mg, followed by two $\gamma$ rays of energies 2.75 MeV (100%) and 1.37 MeV (100%). The mean $\beta$ energy is 0.549 MeV (100%). The physical half life of $^{24}$Na is 15.03 h. When ingested, $^{24}$Na is found to be uniformly distributed throughout the body, with a biological half life of 11 days, long compared to the physical half life.

Prepare a table of relevant parameters and calculate the mean absorbed dose for 1 MBq of $^{24}$Na activity. (The absorbed fractions are 0.314 and 0.274 for the 1.37 MeV $\gamma$ and 2.75 MeV $\gamma$; respectively, and 1 for the $\beta$. Assume 70 kg for the mass of the body.)

Answer: $A: 7.39 \times 10^{10}$ Bq, $D = 2.93 \times 10^{-4}$ Gy

9.6 Cesium-137 is rapidly and almost completely absorbed from the GI tract. It is also distributed uniformly in the body. The retention of cesium is described by a two-compartment equation.

$$R(t) = 0.1 e^{-0.693 \frac{t}{T_1}} + 0.9 e^{-0.693 \frac{t}{T_2}}$$
Where $T_1 = 2$ days and $T_2 = 110$ days. (Reference ICRP 30, Part 1, Page 91). Calculate the effective clearance rates of the two compartments. $T_{1/2}$ of $^{137}$Cs is 30 y.

Answer: $0.347 \text{ d}^{-1}$ and $0.00636 \text{ d}^{-1}$

**9.7** Calculate the specific effective energy (SEE) for whole body internal exposure to $^{137}$Cs using the following data.

<table>
<thead>
<tr>
<th>Radiation*</th>
<th>$n/\text{trans.}$</th>
<th>Mean Energy</th>
<th>Absorbed</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>$n_1$</td>
<td>$E_1, \text{ MeV}$</td>
<td>Fraction $\phi_1$</td>
</tr>
<tr>
<td>$\beta_1$</td>
<td>0.935</td>
<td>0.1749</td>
<td>1</td>
</tr>
<tr>
<td>$\beta_2$</td>
<td>0.065</td>
<td>0.4272</td>
<td>1</td>
</tr>
<tr>
<td>$\gamma$</td>
<td>0.840</td>
<td>0.6616</td>
<td>0.34</td>
</tr>
<tr>
<td>$K$ (ice)</td>
<td>0.0781</td>
<td>0.6242</td>
<td>1</td>
</tr>
<tr>
<td>$L$ (ice)</td>
<td>0.0140</td>
<td>0.6560</td>
<td>1</td>
</tr>
<tr>
<td>$M$ (ice)</td>
<td>0.0031</td>
<td>0.6605</td>
<td>1</td>
</tr>
<tr>
<td>$K_{\text{\beta}_1}$ x ray</td>
<td>0.0374</td>
<td>0.0322</td>
<td>0.76</td>
</tr>
<tr>
<td>$K_{\text{\beta}_2}$ x ray</td>
<td>0.0194</td>
<td>0.0318</td>
<td>0.76</td>
</tr>
<tr>
<td>$K_{\text{\beta}_1}$ x ray</td>
<td>0.0105</td>
<td>0.0364</td>
<td>0.72</td>
</tr>
<tr>
<td>$K_{\text{\beta}_2}$ x ray</td>
<td>0.0022</td>
<td>0.0374</td>
<td>0.71</td>
</tr>
</tbody>
</table>

*Neglected K,L,M auger electrons and L-x rays.

Assume the mass of the body to be 70 kg.

Answer: $6.313 \times 10^{-3} \text{ MeV/transkg}$

**9.8** Use the SEE value from problem 9.7 and find the initial dose rate for 1 Bq of $^{137}$Cs.

Answer: $3.64 \times 10^{-12} \text{ Gy/h}$

**9.9**

a) Using the information from problems 9.6 and 9.7 and the ICRP stochastic dose limit criterion (0.05 Sv) estimate the allowable limit of intake (ALI) for ingestion. *(Hint: Remember that there are two compartments in the tissue with two clearance rates.)*

Answer: $4.0 \times 10^6 \text{ Bq}$

b) What is the committed dose equivalent for intestine of 1 ALI?

Answer: $1.25 \times 10^{-8} \text{ Sv/Bq}$
9.10 Make use of Figure 9.1 of the text and calculate the initial number of transformations per second, and clearance rates ($\lambda_E$), after an inhalation of 1 Bq of $^{137}$Cs (cesium is assigned to clearance category D). Also remember that the N-P region does not directly contribute to the lung dose.

Answer:

<table>
<thead>
<tr>
<th>Compartment</th>
<th>A(t), t/s</th>
<th>$\lambda_E$, s$^{-1}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>(c→BF)</td>
<td>0.076</td>
<td>8.02x10^{-4}</td>
</tr>
<tr>
<td>(e→BF)</td>
<td>0.2</td>
<td>1.60x10^{-5}</td>
</tr>
<tr>
<td>(h→lymph nodes)</td>
<td>0.05</td>
<td>1.60x10^{-5}</td>
</tr>
<tr>
<td>lymph nodes</td>
<td>0.0</td>
<td>1.60x10^{-5}</td>
</tr>
<tr>
<td>(d→GI tract)</td>
<td>0.004</td>
<td>4.01x10^{-5}</td>
</tr>
</tbody>
</table>

9.11 From the results of problem 9.10, calculate the total number of $^{137}$Cs and $^{137m}$Ba transformations in the lung for a time of 50 years (Hint: Since 50 years is long compared to the effective half times in each compartment, use equation 9.23 of the text).

Answer: 1.89x10$^6$ transformations from $^{137}$Cs, 1.79x10$^4$ from $^{137m}$Ba.

9.12 The specific effective energy $\text{SEE}(L-L)$ for $^{137}$Cs is 1.9x10$^{-1}$ MeV/kg·trans and for the short-lived $^{137m}$Ba it is 9.5x10$^{-2}$ MeV/kg·trans. (ICRP 30, Supplement to Part 1, Page 235). Use the $\text{SEE}(L-L)$ and the result from problem 9.11 to calculate the committed dose equivalent to the lung from ($^{137}$Cs-$^{137m}$Ba) deposited in the lung.

Answer: 8.47x10$^{-10}$ Sv

9.13 If 30% of $^{137}$Cs is deposited in the n-p region (see Figure 9.1), 8% is deposited in the T-B region and 25% in the P-region, what is absorbed into the body for an initial inhalation of 1 Bq? (Note: $f_1 = 1$).

Answer: ~ 0.63 Bq

9.14 Use the information and results in problem 9.6 and the result of problem 9.13, and calculate the total number of $^{137}$Cs and $^{137m}$Ba transformations in the body from what is deposited in the respiratory tract and transferred to the body.

Answer: 7.7x10$^6$ transformations for $^{137}$Cs, 7.3x10$^6$ for $^{137m}$Ba.
9.15 a) The specific effective energy, SEE(L=Total Body) for $^{137}\text{Cs}$ is $2.7\times10^{-3}$, and for $^{137}\text{mBa}$, $4.0\times10^{-3}$ MeV/kgtrans. (ICRP 30, Supplement to Part 1, Page 235). Use the SEE (L=Total Body), and the results of problem 9.14, to calculate the committed dose equivalent to the lung from $^{137}\text{Cs}$ in the body.

Answer: $8.0\times10^{-9}$ Sv

b) What is the total committed dose equivalent to the lungs from the inhalation of 1 Bq of $^{137}\text{Cs}$?

Answer: $8.8\times10^{-9}$ Sv/Bq.

9.16 Using the result of problem 9.15b and a weighting factor of 0.12 for lung, calculate the weighted committed dose equivalent in lung from 1 Bq of $^{137}\text{Cs}$.

Answer: $1.0\times10^{-9}$ Sv/Bq.