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**PROCEDURES FOR ASSESSING  
OCCUPATIONAL RADIATION  
MONITORING DATA FOR USE  
IN EPIDEMIOLOGIC STUDIES**

**Douglas J. Crawford-Brown  
James E. Watson, Jr.  
Daniel J. Strom  
William G. Tankersley**

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by

Douglas J. Crawford-Brown, Ph.D.  
James E. Watson, Jr., Ph.D.  
Department of Environmental Sciences and Engineering  
University of North Carolina at Chapel Hill

Daniel J. Strom, Ph.D.  
Department of Radiation Health  
University of Pittsburgh

and

William G. Tankersley, M.S.  
Center for Epidemiologic Research  
Oak Ridge Associated Universities

January 1989

This report concerns work undertaken as part of the Health and Mortality Study of U.S. Department of Energy workers being conducted by Oak Ridge Associated Universities, Oak Ridge, Tennessee, with the collaboration of the University of North Carolina at Chapel Hill, North Carolina, under contract number DE-AC05-76OR00033 between the U.S. Department of Energy and Oak Ridge Associated Universities.

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## Abstract

This report details a consistent and complete methodology for making use of health physics monitoring data in an epidemiologic study. The intent of the methodology is to calculate, for individuals in a study, indices of dose delivered during the course of employment where the index corresponds as closely as possible to the ideas of absorbed dose or dose equivalent. Separate discussions of the use of in vivo monitoring data, bioassay data, or air monitoring data as bases for dose assignments are provided. Mathematical formulas and sampling considerations underlying the use of such data in a large study are developed. Additional discussions focus on the collection, processing, editing, and characterization of the primary data, as well as methods of quality assurance. The methods described in this report are applicable to epidemiologic studies of large populations exposed to radiation by providing a set of consistent procedures to reduce monitoring data to estimates of dose or dose equivalent.

## ACKNOWLEDGMENTS

Portions of this report were abstracted from the dissertation titled "A Strategy for Assessing Occupational Radiation Monitoring Data from Many Facilities for Use in Epidemiologic Studies" prepared by Daniel J. Strom under the direction of James E. Watson, Jr. and Douglas J. Crawford-Brown (1983).

This project would not have been possible without the participation of many persons. T. Terrill Hudgins made significant contributions to this project in ideas and the development of computer programs to test the assessment procedures. He and Susanne Wolf assisted in the preparation of Chapter IV on Data Processing. Other persons at the University of North Carolina at Chapel Hill (UNC) who helped with this project include Joanna Smith, Martha McClanahan, Sharon Pickard, and Bethsaida Seagroves.

Staff members of Oak Ridge Associated Universities (ORAU) who contributed to this effort include Peter Groer, Martha Wray, and Jolene Jones, and former staff members William L. Beck and C. Lynn Sowder.

Ideas, support, and oversight were provided by Carl M. Shy at UNC and Shirley A. Fry and C.C. Lushbaugh at ORAU. C.M. West and L.M. Scott provided valuable assistance in testing these procedures at the Y-12 Plant.

This work is part of the Health and Mortality Study of U.S. Department of Energy (DOE) Workers being conducted by Oak Ridge Associated Universities (ORAU) with the collaboration of the Department of Epidemiology and the Department of Environmental Sciences and Engineering of the School of Public Health of the University of North Carolina (UNC). This work is supported in part by Contract No. DE-AC05-76OR00033 between DOE and ORAU. The data on which the Health and Mortality Study are based are the property of DOE. They are included in DOE's System of Records No. 30 and are protected according to the requirements of the Privacy Act (1974).

Daniel J. Strom worked under appointment to the Nuclear Science and Engineering and Health Physics Fellowship program administered by Oak Ridge Associated Universities for the U.S. Department of Energy.

## PREFACE

The goal of dose assessment is to enable the use of data recorded by occupational radiation monitoring programs to infer the value of the exposure variable for individuals in an epidemiologic study. These data must be transformed into machine-readable records containing some measure of annual doses (or dose equivalents) received by the total body or by one or more target organs for each worker for whom such doses can be inferred. The intent of this report is to provide a set of procedures that will facilitate dose assessment. These procedures can serve as a check list to ensure that necessary tasks are performed and provide recommended methods for converting monitoring data into measures of annual dose.

Our goal is to develop a set of general procedures that will be applicable to the population of workers at any site. Such standard procedures should increase the efficiency and effectiveness of assessing monitoring data for epidemiologic studies. Time and effort spent determining what needs to be done and how to do it should be reduced. Also standard procedures should facilitate combining of data from multiple facilities for a common epidemiologic study.

We recognize that this effort to develop standard procedures applicable to all sites will be incomplete. There will be situations at specific sites not covered by these procedures. Also it is likely that there are other procedures and methods that will complement these. We request comments and suggestions for additions and revisions to these procedures. They should be sent to either Douglas J. Crawford-Brown or James E. Watson, Jr. at the University of North Carolina at Chapel Hill, N.C. Based upon the comments received, and lessons learned from further tests of these procedures, we anticipated that a revised procedures manual will be prepared in the future. We believe that with the cooperation of others involved in similar work, standard assessment procedures can be developed which will be valuable for use in radiation epidemiologic studies.

## CHAPTER 1: INTRODUCTION

A common problem in epidemiological studies is the reduction of a large body of potentially useful data on exposures to some index of that exposure. In radiation studies, the preferred index of exposure to radionuclides is absorbed dose or dose equivalent. By generating study results in the form of a relationship between dose and risk, it is possible to interrelate studies at facilities where exposures may differ in the radionuclide of concern or in its chemical form. These doses, however, are not measured directly by available monitoring data and must, instead, be calculated from the data through use of mathematical formulas. The large number of individuals in a study requires that these formulas be amenable to a computer-based analysis.

The present report details one particular methodology for using radiation monitoring data as a basis for estimating doses to groups or individuals at facilities that form the base of epidemiologic studies. It is assumed that the goal of dose assessment (which itself is only part of an epidemiologic study) is to provide the epidemiologist with a measure of the doses to various body organs as delivered by exposures at a facility or activity under study. In addition, it is assumed that the epidemiologist will require some assurance that the dose estimates represent the best available under the conditions prescribed by the monitoring data. At times, it may not prove possible to calculate doses due to a lack of available models for relating doses and exposures. In such cases, the epidemiologist must receive a measure of exposure which is most closely related to the absorbed dose or dose equivalent. The goals of dose assessment, therefore, are:

1. to collect all data pertinent to the calculation of exposures to radionuclides,
2. to examine the data in as much detail as possible in order to assure its validity as a base for estimates of exposure,
3. to remove from the primary monitoring data those data that are suspected of being unrepresentative of exposure (due to flaws in the data or to irrelevance),

4. to convert the monitoring data into estimates of exposure, dose, or dose equivalent (in reverse order of preference) for each individual or group in the study,
5. to assure the epidemiologist that the generated estimates of exposure, dose, or dose equivalent are the best possible under the circumstances, to provide some understanding of the degree to which such estimates are accurate and precise, and
6. to place the estimates of exposure, dose, or dose equivalent into a format that will be useful to the epidemiologist in conducting a large study that typically requires manipulation of data in a computerized format.

Figure 1-1 displays the general steps that will be developed in detail within the body of this report. For conceptual purposes, the process of dose assessment has been broken into a series of discrete steps that are connected in the figure by solid arrows. These steps consist of:

1. a preliminary determination of the nature of available exposure data and their suitability for generation of estimates of exposure (Feasibility Evaluation),
2. the actual retrieval of data from various types of files in which the data are found (Data Retrieval Criteria and Methods),
3. the retrieval of any and all information pertaining to the quality of the monitoring program (Retrieval of Program Evaluation Data),
4. the processing of the monitoring data to verify its soundness and to place it into a format that is readily used in mathematical algorithms (Data Processing),
5. the editing of the data to remove any data that are suspected of having errors, followed by a summary description of the remaining data (Editing and Characterization),
6. the calculation of the index of exposure, dose, or dose equivalent using the remaining data (Synthesis),
7. the generation of computer files containing the exposure, dose, or dose equivalent estimates in a form that is utilized easily in epidemiologic studies on large populations (Creation of Standard Analysis Files), and

8. the creation of a report describing the quality of the monitoring program and its generated data (Dosimetry Program Evaluation).

While these eight steps are depicted in Figure 1-1 as being distinct and flowing in a linear fashion (see the solid arrows), in practice they are interrelated since findings of any one step may cause the dose assessment process to revert to earlier steps. These links among the various steps are shown by the dashed arrows in the figure. It is the experience of the authors that the actual process of dose assessment rarely flows smoothly between the steps, but instead requires several iterations through the entire sequence (Strom, 1983).

Finally, there is the issue of units used in the report to quantify exposure, dose, and dose equivalent. It is assumed here that exposure refers to some product of either exposure rate (for external exposures) or concentration (for internal exposures) and time of exposure. The radiological quantities of interest then will be (1) the quantity of flux rate of radiation in the environment or (2) the concentration of a radionuclide in air, water, etc. The units then are those of (1) ionizations per unit mass of air and (2) activity per unit mass or volume. For external exposures, the primary units will be Roentgens (R), while the units of internal exposure will be Becquerels (Bq) or curies (Ci) or elemental mass per unit of environmental medium. Dose always will refer to the absorbed dose, which is the ratio of the radiation energy (E) absorbed by an organ to the radiation mass (m) of that organ. Dose equivalent is the product of the absorbed dose for an organ and the quality factor for the primary radiation to which the worker was exposed. Typically, the unit of absorbed dose is the rad or the grey (Gy), while that for the dose equivalent is the rem or sievert (Sv). The report presumes a familiarity with these basic radiological units, and with the general principles of metabolic modeling and dosimetry. If the reader is not already familiar with these areas, it is recommended that some preliminary reading in the subject matter of health physics and radiation protection precede the use of this report.

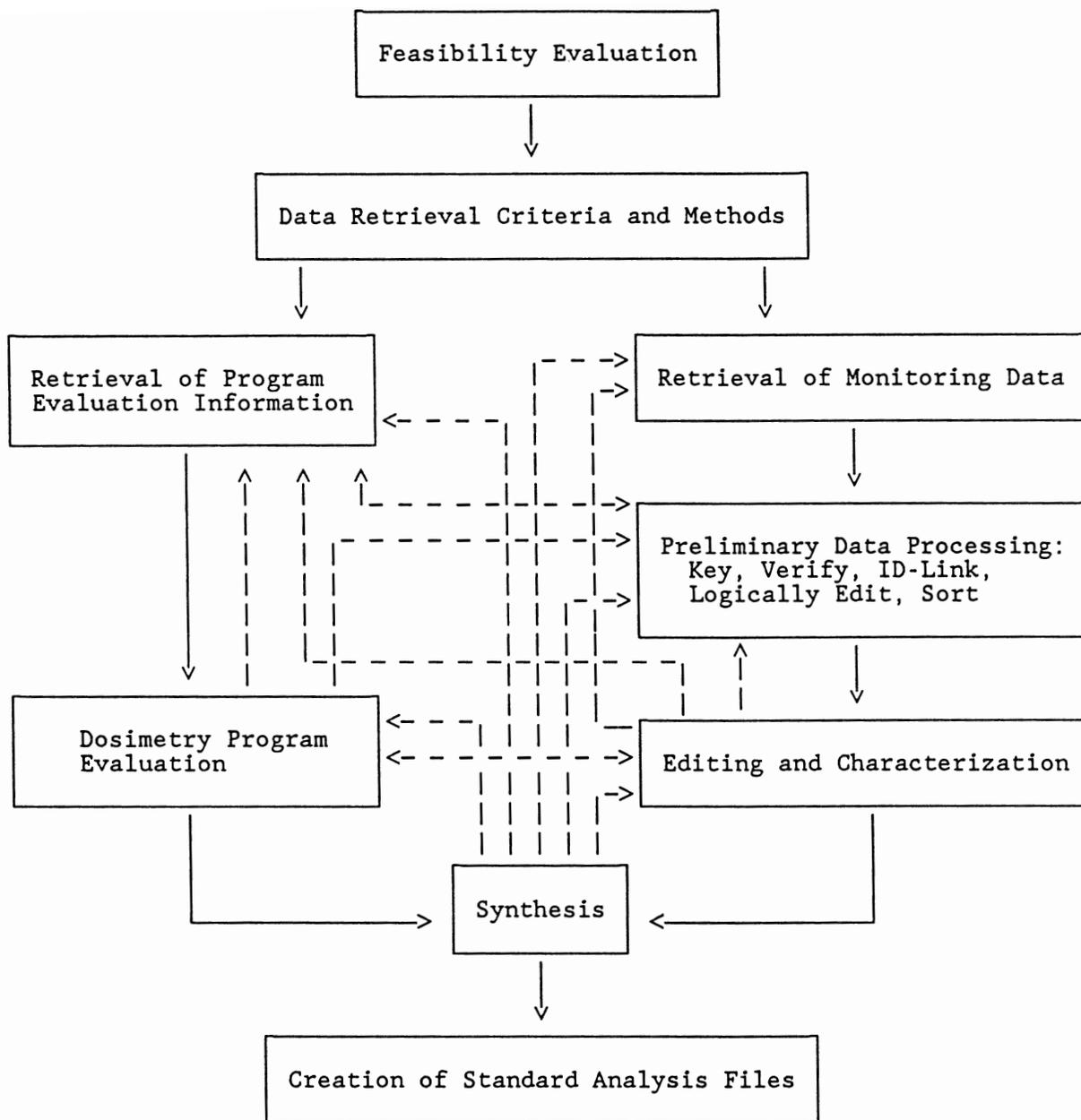


Figure 1-1. Standard Assessment Procedures in Simplified Form. Solid lines represent the ideal course of dose assessment; dashed plus solid lines represent actual experience.

Despite the real world departures from the ideal dose assessment process diagrammed by the solid arrows in Figure 1-1, the ideal process is a useful organizational framework for the standard assessment procedures. The use of standard data handling procedures will result, in the long run, in the most efficient use of time and in greater uniformity and reliability of results.

The following sections describe in detail the steps of dose assessment as outlined above. Many of these steps have been applied and tested on internal and external health physics monitoring data at one large DOE facility concerned primarily with uranium processing (Strom 1983).

## CHAPTER II: FEASIBILITY EVALUATION

The first step in dose assessment is to evaluate the feasibility of obtaining and using monitoring data and program evaluation information from a facility to be studied. Investigators should be aware that the extent to which data usage is deemed feasible will depend on the particular epidemiological question being asked. For example, computation of individual-specific doses requires more complete data than does the computation of doses to a population such as might constitute a job title. In any event, use of the monitoring data is feasible only if there is some assurance that the data reflect actual exposures at the facility.

To determine what data are available, a "Dosimetry Records and Radiation Hazards Questionnaire" (DRRHQ) has been developed (Appendix A). Some ideas for this questionnaire were derived from the work of Dreyer et al. (1980) and the work of Fix, Selby and Vallario (1981). However, most of the questions in the DRRHQ are adapted from the "Dosimetry Assessment Fact Sheet" (DAFS) developed by Beck, Stansbury and Watson (Appendix B). The DAFS is a checklist of information needed to evaluate an occupational dosimetry program in the context of an epidemiologic study. The DRRHQ compiles the following information:

1. identification of knowledgeable health physics contacts currently at the facility or retired or transferred,
2. a list of dosimetry program documentation and the location of such documentation,
3. a categorization of site operations by year,
4. a determination of which years employees were exposed, externally to each of four types of radiation and during which years they were monitored,
5. a description of possible internal exposures and monitoring,
6. a list of units and quality factors applicable for each high-LET radiation type and year, and
7. information on several miscellaneous topics such as instrument calibrations, and measures of precision.

The completeness and appropriateness of dosimetry programs are assessed by comparing the hazards at a facility to the monitoring programs in place.

Completeness is the degree to which there were no gaps in the program (e.g., an exposure existed since 1945, but was not monitored before 1950).

Appropriateness is the degree to which the monitoring program addressed the hazards (e.g., whether a beta-gamma badge was used to monitor tritium exposures). A monitoring program that, given the technology of the day, adequately addressed all potential (significant) hazards at a facility is considered appropriate. For example, film badges were and are an appropriate monitoring program for external beta and gamma exposures but would not be appropriate for measuring alpha exposure. At the outset of the study, therefore, it should be determined whether all significant exposures were appropriately monitored throughout the history of these exposures.

Since dose assessment includes an examination of the methods by which monitoring data were generated, documentation detailing the monitoring methods and their accuracy, precision, and calibration procedures should be collected. While the judgment as to whether such documentation is adequate revolves about a subjective evaluation of that documentation, the goal of this step should be to ensure that all monitoring procedures have been scrutinized. If the data reported by a facility have been generated by application of an algorithm, this algorithm should be documented and the values of all parameters determined. This step is particularly important when the records at a facility have been generated by application of metabolic and dosimetric models, since significant changes have occurred with time. Where possible, steps should also be taken to obtain documentation on the choice of sampling schedules at a facility, since the results of a monitoring program can be invalidated by a sampling schedule that does not reflect the proper temporal pattern of exposures.

No clear rules exist for determining whether documentation is adequate. In general, however, documents involving peer review (such as journal articles) should receive greater attention than documents such as internal reports or personal memoranda. The intent is to ensure that measurement procedures and algorithms are acceptable in light of current knowledge and practice. Some facilities will have employed commercial services for the monitoring program, necessitating a search for documentation of the procedures employed by that particular commercial service.

### CHAPTER III: DATA RETRIEVAL CRITERIA AND METHODS

Analyses of DRRHQ information lead to the development of criteria and methods for data retrieval from each facility. These analyses prompt decisions regarding additional dosimetry program information that should be requested (e.g., specific documents relating to the dosimetry programs); individual monitoring data that should be requested, and for which individuals; forms and formats to specify for monitoring data; identifiers needed for monitoring data; and documentation required to make monitoring data useful (e.g., units).

Prior to data retrieval, it should be determined whether the facility has plans to convert any data from non-machine-readable form to machine-readable form. If such plans exist, it might be most cost-effective to postpone dose assessment until the data are in machine-readable form.

For facilities that are still in operation, telephone and mail contact is made with facility health physicists and management. A letter describing the desired health physics information is sent to those who will be involved with data retrieval. Following this, a visit to the site to talk with the health physicists responsible for personnel dosimetry and records is the best way to retrieve most of the needed information. During such a visit, those parts of the DAFS which have not already been answered can be completed, and ambiguities or omissions in responses to the DRRHQ can be clarified. Monitoring data and program evaluation information can be photocopied, microfilmed, or obtained in machine-readable form. The decisions about how to retrieve data can be made during the facility visit.

It is important that a health physicist or other qualified researcher be actively involved with decisions of what data to retrieve and how to retrieve data in order to avoid ambiguities and misunderstandings. The health physicist must have a knowledge of the relationships between measurement practices, measurement quantities, and doses. In addition, he or she should have a firm understanding of how such data relate to necessary metabolic and dosimetric models. The dose assessment staff must be able to determine if a metabolic model applies on any time scale or if it only applies on a specific time scale. For example, many of the ICRP metabolic models are intended only for the computation of 50-year committed doses and

may not be valid for computation of annual doses. Where alternative models are not available, however, the researcher has no recourse other than to use the ICRP models and note the uncertainties. The health physicist ensures that sufficient documentation is obtained for understanding and interpreting data and also avoids retrieving data that are of no use in assessing exposure or doses. Such topics as chemical forms and solubility classes of airborne radioactivity, notional doses, possibility of hardcopy validation, machine codes, and personal identifiers accompanying the data should be discussed. It is important to be certain that all relevant dose data are retrieved, including notional doses created following accidents and nonroutine monitoring that may have been done for only a few individuals. Particular emphasis should be placed on the distinction between current and past health physics policies and procedures.

## CHAPTER IV: DATA PROCESSING

Before data processing begins, investigators should have engaged in sustained dialogue with health physics and data processing personnel at the facility to ensure common understandings with respect to data acquisition.

Minimum preparation should include:

1. identifying classes of information of interest to investigators: A well-conceived study is characterized by asking a set of questions that can be answered with specifiable data rather than by asking what data are available and then centering requests around the response;
2. revising study plans and data requests in light of unexpected, restricted, or unobtainable information;
3. establishing a claim to information held by the provider by receiving access authorization from facility officials: Once formal authorization is received, four-way contact should be maintained, between investigators and facility officials at one level and research staff and facility data processing personnel at another level, to ensure that understandings have been and continue to be communicated adequately. Problems frequently occur when individuals at the "control" level assume that adequate preparation for information transfer between persons at technical levels has been accomplished by a series of congenial telephone conversations; interlevel communication must be open and frequent; and
4. reaching agreement on types and volume of data required and mechanisms of transfer.

While consensus on the value of these practices is common, the failure to implement or maintain them over the course of data transfer is equally common. The success and efficiency with which the data processing operations outlined below are carried out is dependent on preparation of the kind described above.

## Preliminary Data Processing

Preliminary data processing includes keying of non-machine-readable data, verifying data received in machine-readable form, and linking each monitoring record with an individual known to have worked at the facility.

If monitoring records are retrieved from a facility in hardcopy form, the data must be keyed into machine-readable form. To ensure that all appropriate information is extracted from hardcopy dosimetry records, a health physicist should enlist the aid of data processing personnel. To ensure accuracy, data should be keyed twice, and the results compared.

If data derived from paper forms, but delivered in machine-readable form, were entered by personnel not under project control, it is important to evaluate the accuracy of transcription. Hardcopy validation is done to reveal (1) errors in transcription, (2) omissions of specific data items or records, and (3) more systematic omission of groups of records, for example those covering a particular time period. The criteria for hardcopy validation require that at least some records from each year be checked and that special attention be paid to data items derived from paper records whose format has changed over time. It is not uncommon for definitions and units of measurement to change although the same underlying property is being recorded.

To check accuracy of transcription a random sample of records is selected from machine-readable data. These records are printed and manually compared with the original records at the facility. From this comparison, a judgment is made regarding the overall accuracy of transcription, and an estimate is made of the magnitude and direction of any bias that inaccuracies in transcription may have caused. Poor agreement may require reentry of some or all dosimetry records.

It is also useful to generate a frequency distribution of record counts by year (i.e., a histogram of the number of monitoring results within preselected bounds versus the mean of each bounded interval, or a histogram of the number of monitoring results versus year). An uneven distribution usually indicates a change in the monitoring program or a change in the number in the work force. However, this finding could be the result of

missing records. Any significant maldistribution should be discussed with the health physics personnel at the facility to ascertain the reason.

With the exception of area air sampling and area dose measurements, personnel monitoring records are associated with individuals. Data from facilities may have a variety of personal identifiers associated with them such as name, social security number, employee number, badge number, date of birth, race, and sex. To link dosimetry records to the appropriate individual, as many of these attributes as needed should be used. Records that cannot be linked to an individual known to have worked at the facility should be set aside in a separate data set and described by year of monitoring, dose, and any other variable that might identify the records as control or calibration records, or records for consultants, visitors or loaned personnel.

#### Error Detection and Correction

Preliminary error detection efforts check for exact duplicate records, that is, records that are alike in every aspect. Procedures for handling duplicate records depend on the number and nature of the duplicates. For example, records measuring dose rate or organ concentration, such as concentration of radioactivity in urine or radioactivity in the lung, may be left in duplicate without affecting calculated doses, if the temporal-integration approach to dose estimation is employed. However, records measuring a dose, such as film badge or TLD readings, cannot be left in duplicate without upwardly biasing the exposure variable. The same is true if dose rate or bioassay measurements are simply averaged over an interval of time to yield a measure of mean exposure or organ burden. A good general rule is to remove all duplicate records, leaving only a single record from each set of duplicates. In any case, duplicate records are dealt with at this stage of the dose assessment process. They are counted and characterized by year of occurrence, employee ID, type of assay, and other information contained on the record, whether they occur at random, or if there is a pattern that may signify an underlying problem. If no underlying pattern is apparent, duplicate records can be removed from the file. If a pattern emerges for some or all duplicate records of the dose type, then an

explanation should be sought from the facility that supplied the dose records.

The second step in the editing process consists of univariate edit checks. Interval or continuous data such as film badge readings, bioassay or lung counting results, sample volumes, and sample times should have the following calculated: mean, median, range, quantiles. Bar graphs of doses grouped in predetermined categories and normal probability plots are also useful as long as one keeps in mind the effects in these distributions caused by large outliers. Records containing large outliers are output for examination and verification at this stage. It may be possible to eliminate or adjust large negative outliers as self-evident mistakes. A large negative outlier is a data value that is more than two standard deviations below zero, where the standard deviation is that for the counting method. For example, if the standard deviation of a urinalysis measurement is known in advance to be 10 dpm/24 hours, then a value of -21 or less is a "large negative outlier." Such a large negative outlier is unlikely (< 5 percent) to be a valid data point and may represent an error. Such errors inject a systematic negative bias into the data set. Large positive outliers could be valid, so it must be ascertained, by hardcopy validation if possible or by interviewing health physics contacts at the facility, if these are results of real measurements. Since large doses usually are accompanied by an accident report, large positive doses often can be checked by documentation such as journal articles reporting the incident.

Nominal or ordinal data are checked by frequency distributions for missing or out-of-range values. A value test is simply a comparison of the values found on the data file with values found in documentation supplied by the facility. Values found in the documentation are assumed to be valid; values found on the data file but not in the documentation indicate that either the data file values or the documentation are in error. If the occurrence of invalid values affects a significant number of records, it must be resolved by checking one's own work and if no errors are found, by contacting the facility which supplied the data.

Multivariate edit tests are performed where possible to detect logical inconsistencies. For external dose records, such edits include the comparison of total penetrating dose with the sum of gamma and neutron doses

(they should be equal) or the comparison of skin dose with the sum of total penetrating dose and dose from betas, taken as the entrance dose to the body. Multivariate checks may also be possible for internal monitoring results. For example, if raw counting data are available, the sum of net count rate,  $N_n'$ , and background count rate,  $N_b'$ , should be equal to the gross count rate. Records failing these tests are counted and characterized by employee ID and year to identify any patterns. Attempts are made to correct errors by contacting the facility or consulting documentation.

Different types of records can be compared for inconsistent dates. For example, an individual may have dose records outside of his employment periods at the facility. Employment dates are suspect if a number of dose records in chronological order lie outside of recorded employment dates. However, one or two records outside of a person's employment periods could indicate errors in monitoring record to employee linkage.

Other problem records should also be considered. They include records that are not relevant or are unusable, such as control, calibration, and background measurements; records made for administrative purposes but containing no dosimetric information; and records that have flags such as "do not use" associated with them. Evaluation of problem records is necessary to determine the exact nature of the problems and then the final disposition of the records.

The characterization of control, calibration, or background measurements provides insight into minimum detectable levels, precision, accuracy, and trends in these values over time. (A more complete discussion of the need for these data is deferred until the discussion on uncertainties.) Histograms of the number of monitoring records whose values fall in various categories reveal the distribution of monitoring results and can be used to evaluate these data, particularly when the histograms are constructed by year or other logical time unit. Reasonable time periods are judged from documentation and other sources indicating when record formats or measurement practices changed.

## Documentation

These remarks are based on experience with several large, complex facilities but generally are applicable to production sites where dosimetry records are sought. We once more emphasize that the data acquisition and characterization phases of dosimetry will be facilitated by clear communication among all parties. Also, the importance of paper documentation of all data, both newly received and transformed, cannot be overstressed. Researchers should request detailed descriptions of all data items for data processing purposes including storage layouts and formats, and for dosimetry purposes including definitions of data items, the values they may assume, and their units of measurement.

Data processing personnel must consistently and intelligibly document all work in detail, beginning with received data, through the production of usable dose assessment files, and concluding with the archiving of programs and files.

## CHAPTER V: DOSIMETRY PROGRAM EVALUATION

Information concerning radiation hazards monitoring, calibration, and recordkeeping procedures, each as a function of time, is needed for the evaluation of dosimetry programs and for the calculation of doses from monitoring records. This is a continuation of the tasks discussed in Chapter II, "Feasibility Evaluation." Such information is referred to as "program evaluation information" and is described in detail in the Dosimetry Assessment Fact Sheet (Appendix B). Dosimetry program documentation listed in the DRRHQ is retrieved if it is judged to be relevant to the dose assessment process. Such documentation is used for many of the steps in the dosimetry program evaluation and is necessary for the interpretation and use of monitoring results in the creation of annual dose equivalents to target organs or some measure of exposure.

### Completeness and Appropriateness of Dosimetry Program

A profile of activities which produced or used radiation is constructed for the facility in question, based on responses to the DRRHQ and other available historical information. From a knowledge of what went on at the site, deductions are made about the types of radiation hazards which would have been present. For a detailed listing of radiations associated with a wide range of radionuclides see Kocher (1981).

Responses to the DRRHQ are also used to construct a profile of the radiation monitoring program as a function of time. Additional information is gained from the dose data themselves as a check of DRRHQ responses. For example, if no monitoring data actually exist for a period during which DRRHQ responses indicate they should, then it is safe to conclude that monitoring was inadequate or records were lost.

The completeness and appropriateness of the monitoring program are evaluated by comparing the temporal radiation hazards profile with the temporal radiation monitoring profile. This is similar to the assessment described in the feasibility evaluation stage but uses additional information as well as DRRHQ responses. Where possible, conclusions are drawn regarding the importance of any gaps. In the review of program

documentation and other relevant literature, attention is paid to the mention of small dosimetry programs or non-routine dosimetry that may have been done for only a few individuals. Prototype programs may have existed in early days with results stored apart from main record files. If such programs come to light, it is important to retrieve data from these programs.

### Quality Assurance

Quality assurance is used here to mean calibration and standardization procedures which were employed to ensure that monitoring was accurate, reproducible, and as unbiased as possible. Quality assurance is evaluated by studying program documentation insofar as it is available. In general, three levels of quality assurance should be examined. The first level ensures that appropriate measurement techniques were employed for the hazards at a site. The second level ensures that these measurement techniques were properly calibrated against known levels of radiation at the appropriate energies and geometries. The final level ensures that the measurement techniques were employed properly in practice. Quality of a monitoring program can be seriously affected at any of these levels and each must be examined in detail for all significant hazards. An additional consideration, apart from the measurement schemes themselves, is the question of the extent to which the radiation hazards have been identified with acceptable precision and completeness, as discussed above.

### Dosimetry Initiation Criteria and Minimum Detectable Quantities

A determination of criteria for the issuance of personnel dosimeters, the performance of bioassays, or area monitoring is made from existing program documentation. According to Energy Research and development Administration (ERDA), now Department of Energy, regulations (ERDA 1977), "Monitoring is required where the potential exists for the individual to receive a dose or dose commitment in any calendar quarter in excess of 10 percent of the quarterly standards." Earlier regulations provided for monitoring for potential exposures in excess of 25 percent of standards.

Thus, persons exposed to levels below the limits may not have been monitored, although they might have received exposures.

It is important to assess the criteria used for initiating monitoring, since it is not generally valid to assume that unmonitored workers were unexposed. If a large fraction of the population was excluded from routine monitoring, it may still be possible to place some bounds on the likely exposures to these unmonitored workers. This may be done by searching for area monitoring results in buildings housing the unmonitored workers. These buildings are usually physically distinct from those housing operations requiring monitoring. In addition, some facilities will sample randomly a small number of workers in unmonitored job titles as a check of the efficacy with which exposed workers are being identified. These random samples could be taken as indicative of exposures in the unmonitored workers if it can be assured that the random sample truly was random and not chosen on the basis of expected exposures. This topic is discussed in more detail in Chapter VI in the section on uncertainties.

Monitoring instruments and methods typically have a lower limit for the measured quantity (activity, exposure rate, etc.) that can be measured with confidence. This minimum detectable level (MDL) generally corresponds to the lowest value of the measured quantity that can be distinguished from zero at some level of confidence. As a result, values at or below the MDL are considered by health physics staffs as being poor indicators of exposure levels. Such doses may have been recorded as (1) "equal to the MDL"; (2) "less than the MDL"; (3) "minimal"; (4) zero; or (5) some fraction of the MDL such as "one half MDL." Any of these methods has the effect of biasing the collective dose equivalent or mean exposure variable in one direction or another. Potential bias increases with increasing numbers of dosimetry results which are below the MDL and with increasing values of the MDL. This problem is discussed in more detail in a section of Chapter VI concerning the handling of minimum detectable quantities. At the present stage, it is necessary only to identify the existence of MDL records and the nature of reporting for such records.

## Procedures for Lost or Damaged Dosimeters and Notional Doses

Where possible, an evaluation is performed of procedures used for dealing with lost or damaged dosimeters at a facility. Loss or damage of some dosimeters is virtually a certainty in an occupational monitoring program. A variety of actions may have been taken by the facility when this loss or damage occurred. Sometimes a new dosimeter is issued, and no other action is taken. This results in a record of exposure or dose which is a lower limit for the monitoring period, since it does not include exposures during the "lost" period of time. Sometimes, after interviewing the worker in question or the health physics staff, a monitoring result is assigned based on area monitoring and time-in-area. Such an assigned dose is an example of a notional dose. A third alternative is the assignment of a monitoring result based on some averaging scheme using results recorded before and after the loss or damage. Yet another possibility is the assignment of the maximum permissible dose for the monitoring period. This last possibility is valid from the radiation protection point of view if it can be assumed that the maximum permissible dose had not been exceeded, because it prevents the individual from being overexposed in the future due to an underestimate of the "lost" dose. However, this solution biases the individual's records towards a higher dose than was actually received and is thus not valid for epidemiology. In all cases except the latter, the recorded results represent the best estimate that can be made. In the latter case, a procedure such as NEARBY, described in Appendix C, should be used to generate a more likely monitoring value. This procedure could also be used in cases where a badge was lost and a new badge issued without accounting for the "lost" dose. The NEARBY procedure then can be used to determine the exposure rate during the missing period of time which can be multiplied by the appropriate length of "missed" time.

Other reasons for the generation of notional doses by the facility should be examined, if such doses appear in the records. The occurrence of accidents or other known high exposures is investigated. Notional doses created for administrative reasons are evaluated.

## Analysis of Problem Codes Accompanying Monitoring Data

In some cases, monitoring data may be accompanied by items that contain information about the monitoring data such as an indication of "investigated dose." Patterns in codes uncovered during editing and characterization of data must be interpreted in light of the dosimetry program that produced them. Investigation of the meanings and implications of these codes is done where possible. Such data items are of use in assigning judgment flags in the Synthesis Step of these procedures (see Chapter VI).

## Quality Factors

A quality factor is a number which is used to multiply absorbed doses in rad (or grays) to yield dose equivalents in rem (or sieverts). Quality factors are 1 for x, gamma, and beta radiation and are higher (up to 10 or 20) for high linear energy transfer (LET) radiation such as neutrons or alpha particles. Table 5-1 shows the evolution of quality factors (Q) and their antecedents, the Relative Biological Effectiveness (RBE), as well as the values used by the United States Nuclear Regulatory Commission (rem/rad). For a multiple-facility epidemiologic study, doses must be recorded in the same units.

Dose equivalent, H (rem or sieverts), is equal to the product of absorbed dose, D (rad or grays), the quality factor, Q, and the product of other modifying factors, N (ICRU 1980):  $H = D Q N$ . The rem (or sieverts) calculated using one quality factor for alphas (e.g., 10) are not the same units as rem (or sieverts) calculated from the same absorbed dose values using a different quality factor (e.g., 20). To ensure comparability across facilities in cases where doses or dose equivalents are employed, quality factors must be known and a consistent set of quality factors employed. Likewise, if epidemiologic studies are to be conducted using absorbed dose, D (rad or grays), and dosimetry data have been reported in dose equivalent units, H (rem or sieverts), quality factors underlying the calculation of the dose equivalent must be known to reconstruct the absorbed dose. Responses to the DRRHQ should be sufficient to determine the quality factors assumed by a facility in calculating dose equivalent. Otherwise, the values

in Table 5-1, in the row labeled NRC 1978, should be assumed to have been used.

On a final note, exposure to several radionuclides may result in organ-specific doses from each radionuclide. Due to differences in the "quality" of differing radiations (Alper 1979), it is not generally possible to determine the potential risk to an organ by a simple summation of doses from the underlying radionuclides. In radiation protection, however, it is common practice to sum the contributions from all radiations to obtain the dose equivalent to a specific organ, resulting in a single measure of the potential for biological damage.

It may, at times, be desirable to separate the various dose components comprising a total dose equivalent to an organ. This could arise, for example, in instances where the doses from several radionuclides are delivered to very different cellular subpopulations (usually of interest only for high-LET radiations such as alphas). If this is deemed likely, it might prove best to compute only organ doses for each separate radionuclide, followed by some form of stratified analysis, and not employ the concept of dose equivalent. In addition, the use of doses as the exposure variable is necessitated by any attempt to develop the quality factors themselves from an epidemiologic study. It should be noted, however, that the low doses received by study populations as a result of occupational exposures makes it unlikely that values of  $Q$  can be determined directly in any epidemiologic studies other than studies of highly exposed individuals.

Table 5-1  
Quality Factors (Q), Relative Biological Effectiveness  
(RBE), and Regulatory Factors (rem/rad) from  
1945 to the Present

Ref.	X ray gamma, beta	Electron <.03MeV	Thermal neutron	Fast neutron	Proton	Alpha	Recoil	Factor type
Stone 1951 (Manhat. Project, 1945)	1	-	4	5	-	10	-	r per rep
Taylor 1971	1	-	5	10	-	10	-	RBE
ICRP 1955	1	-	-	10, spect**	10, spect	spect	20, spect	RBE
ICRP 1960	1	1.7	-	-	-	10	20	RBE
NCRP, No. 38, 1971	-	-	2	10, spect	-	-	-	Q
NCRP, No. 39, 1971	1	1	2, or 3 if E<10keV	10*, spect	1-10, spect	1-20, spect	20	Q
ICRP 1973, ICRP 1977	1		2.3, spect	10*, spect	spect	20, spect	20, spect	Q
NRC 1978	1	-	2, spect	10, spect	10	20	20	rem per rad

\* "Acceptable, but may be unduly conservative".

\*\* "Spect" indicates quality factors between 1 and 20 are used depending on the energy of neutrons or other particles, as determined by spectroscopy. Tables and graphs of quality factors as functions of energy or linear energy transfer (LET) are found on pages 19, 91, and 92 of ICRP 1955; in Table 4, page 16 of Publication No. 38, NCRP 1971; on pages 81-83 of Publication No. 39, NCRP 1971; in Figures 1, 2, 3, and 15 of ICRP 1973; in paragraphs 19 and 20 and Figure 1 of ICRP 1977; and in Section 20.4 of of NRC 1978. The ICRP has not changed its RBE (now Q) versus LET values since 1955; however, the Q for alphas and fast neutrons changed to 20 in 1977 and 1985, respectively.

## CHAPTER VI: SYNTHESIS OF PROGRAM EVALUATION INFORMATION AND MONITORING DATA

In the synthesis phase of dose assessment, occupational radiation monitoring data are processed and modified as necessary, based on results from the dosimetry program evaluation and from the editing and characterization of the actual data. For external monitoring data, the endpoint of the synthesis phase is annual cumulations of exposure or dose for each worker for each type of radiation (gamma, beta, neutron) and each calculated quantity (penetrating and skin dose). Internal monitoring data are put in the form of annual organ doses or some exposure measure for each worker. Uncertainties are computed and assigned to external and internal calculated values. Judgment flags that relate to the suitability of data for use in epidemiologic analysis are also assigned. These data are then ready for transfer to analysis files for epidemiologic studies. In much of the following discussion, it is assumed that the end result of dose assessment is the generation of doses or dose equivalents for members of the exposed population. It is recognized that, at times, only the development of exposure estimates will be feasible, and these cases are noted accordingly.

### External Doses

At this phase of assessing the monitoring data, data processing, including editing, validation, and initial characterization, will have been completed. Also, the facility's dosimetry program will have been evaluated with regard to completeness and appropriateness, treatment of minimum detectable quantities and notional doses, use of problem codes, and quality factors. Using information gained from the program evaluation, minimum detectable quantities and notional doses are dealt with. Annual cumulations of dose (or exposure variable) data for each worker are done and final analyses and characterizations of the data are performed.

## Minimum Detectable Doses

Before dose data can be cumulated to annual totals for individuals, the question of minimum detectable doses (MDDs, the dosimetric analogue of MDLs) must be addressed. Doses at or below the MDD may have been recorded (1) as zero; (2) as equal to the MDD; (3) as "minimal"; (4) as some other value such as one half the MDD; or (5) as the actual measured value.

If dosimetry results below the MDD are recorded as equal to the numerical value of the MDD, a positive bias can be introduced into the data (Waite et al. 1980). Changing dose records equal to the MDD to values of zero typically reduces the magnitude of bias in the dataset (since most workers assigned the MDD actually experience less than 50 percent of the MDD) and changes the direction of the bias from an overestimation of the collective dose to an underestimation. The ICRP has recommended that doses less than or equal to a small "recording level" such as an MDD be treated as zero "for the purposes of radiation protection" (ICRP 1977). Many facilities appear to have followed policies similar to this ICRP recommendation over the years, so monitoring results of zero may not be indicative of real zero exposures.

An alternative to setting MDD values to zero is to set them equal to a small positive number. A defensible method of arriving at this small positive number is to fit doses, obtained over a given interval, above the MDD to a lognormal, or other appropriate, curve. Then, assuming that doses below the MDD, as integrated over the same time interval, would have had a similar distribution had they been measured accurately, the average dose from zero to the MDD is computed from the distributional fit. This value is used for all records originally having values equal to the MDD or below. This method is described in detail by Strom (Strom 1986). We believe that this computation is generally not necessary because of the small values of doses less than the minimum detectable dose. However, it may be important in early years, when the MDD was larger. In any event, care must be taken to ensure that all pertinent data are the result of single measurements, that they do not represent a summation over several measurements, and that all doses are computed for equal intervals of time.

## Notional Doses

Notional doses include any doses that were created or assigned without having been directly measured for individuals (Reissland 1982; UNSCEAR 1983). Such doses arise when dosimeters are lost, damaged, not worn, or when an exposure occurred to an individual who was not routinely monitored. Notional doses may be generated by the facility in question and added by them to their data set, or they may be added to the data set by the persons doing dose assessment.

When health physics records have flags indicating "do not use," "damaged film," "do not average," "flawed," or other notations, it may be possible to replace the unusable record with a notional dose. A computer program has been developed to generate a notional dose each time a flawed record is encountered (Hudgins and Strom 1983). The program, called NEARBY, interpolates between chronologically nearby records to generate a notional dose for each flawed record. The NEARBY program is described in Appendix C. This program may also be used to replace a notional dose assigned by a facility based on the maximum permissible dose for the monitoring period. This possibility is discussed earlier in "Dosimetry Program Evaluation."

It may be necessary for the health physicist doing dose assessment to create notional doses for workers during times when there was no personnel monitoring. Inferences about appropriate notional doses may be made from later time periods when there was a monitoring program, combining job titles with monitoring records. In this extrapolation-to-earlier-time method, a person with the same job title, both before monitoring and afterward, would be assigned radiation doses at the same annual rate as he had received after monitoring began. Alternatively, population average dose rates could be compiled for that job title and used in the same way as the individual doses. Clearly, such an approach assumes that exposures did not change appreciably with time.

## Calculation of Annual External Doses

Three general cases arise in computing annual doses due to external irradiation. Prior dose assessment should have revealed which case holds true for any given worker or job title.

Case 1. If a worker was monitored by a personal dosimeter at all times of significant exposure during the year, and if the intervals of time represented by the dosimeters do not overlap, then the sum of all such monitoring results during the year is set equal to the annual exposure. This would apply for all forms of radiation. Let  $x = \{x_1, x_2, x_3 \dots x_n\}$  be the set of  $n$  personal dosimeter readings during the year for a given worker. This worker is assigned an annual exposure,  $E$ , of  $E = \sum_i x_i$ . Dates of measurements are not important in this case, other than to insure that duplicate records have been removed from the set  $x$ . Investigators should remember that some of the  $x_i$  values may have resulted from the generation of notional doses such as might arise in the use of the NEARBY program.

Case 2. A worker may have been monitored by a personal dosimeter only on a random basis (in time) to ensure that exposures were low. If this is the case, exposure rates during periods when the worker was not monitored may be significant. The exposure rates during the periods of monitoring in a year should then be averaged and assumed to hold throughout the year. Again, let  $x$  be the set of personal dosimeter readings for an individual during a year of interest;  $x = \{x_1, x_2, x_3 \dots x_m\}$ . Also, let  $t_{i,s}$  and  $t_{i,r}$  be the times (in days) associated with the start (s) of exposure and removal from exposure (r) for the monitoring result  $x_i$ . The mean exposure rate,  $\hat{E}$ , during the year of interest then is given by

$$\hat{E} = \left( \sum_{i=1}^m x_i \right) / \sum_{i=1}^m (t_{i,r} - t_{i,s}) \quad (6-1)$$

The annual exposure for the worker then is

$$\begin{aligned} E &= \hat{E} (N) \\ &= N \left( \sum_{i=1}^m x_i \right) / \sum_{i=1}^m (t_{i,r} - t_{i,s}) \end{aligned} \quad (6-2)$$

where N equals the number of days of exposure and assuming no change in job status during periods of non-monitoring.

Case 3. A more general case of 2 is one in which a job title has been determined by the health physics staff of a facility as having exposures that do not require continuous monitoring. In this case, persons may have been selected at random from the job title population and monitored for short periods of time. Again, let  $x = \{x_1, x_2, x_3 \dots x_k\}$  be the set of k measurements on the job title population. The time values are as described in case 2. The mean exposure rate for the job title population is given again by

$$\hat{E} = \left( \sum_{i=1}^k x_i \right) / \sum_{i=1}^k (t_{i,r} - t_{i,s}) \quad (6-3)$$

The mean annual exposure for the persons in the job title then is given by

$$E = \hat{E} (N) \\ = N \left( \sum_{i=1}^k x_i \right) / \sum_{i=1}^k (t_{i,r} - t_{i,s}) \quad (6-4)$$

Note that this annual exposure applies only to workers employed in the job title throughout the year of interest. Otherwise, this annual exposure is multiplied by the fraction of the working year during which the worker was employed in the job title.

In some cases, personal monitoring results are not available and E must be estimated on the basis of area monitoring results at locations typical of work stations for a job title, department, or building. At such times, workers or job classifications must be matched with measurements made at the appropriate work locations. Let  $Y = \{y_1, y_2, y_3 \dots y_n\}$  (in units of exposure per day) be the set of n area exposure rate measurements matched to a worker or job title. They are assumed to be a representative sample of the exposure rates at the locations of interest (care should be taken to ensure this fact). The mean exposure rate for the job title then is

$$\hat{E} = \sum_{i=1}^n y_i / n ,$$

and the mean annual exposure is

$$E = \hat{E} (N)$$
$$= \frac{N}{n} \sum_{i=1}^n y_i . \quad (6-5)$$

If annual cumulations of dose or exposure from all radiation sources for workers have not already been done, these dose-type data must be added together for each worker for each year for the total body and skin. External monitoring results of gamma (and X ray) and neutron exposures combine to give penetrating total body dose equivalent (D.E.):

$$\text{Penetrating D.E.} = \text{Gamma D.E.} + \text{Neutron D.E.}$$

Beta dose equivalent is added to gamma and neutron dose equivalents to yield the skin dose equivalent:

$$\begin{aligned} \text{Skin D.E.} &= \text{Gamma D.E.} + \text{Neutron D.E.} + \text{Beta D.E.} \\ &= \text{Penetrating D.E.} + \text{Beta D.E.} \end{aligned}$$

Reported measurements of external exposure strictly apply to the exposure at the skin surface (or, more specifically, at the location of the instrument or personal dosimeter). Doses to specific organs within the body may differ from this external exposure reading. Conversion factors from external exposure readings to organ-specific doses for penetrating radiations emitted by common radionuclides may be found in a number of references (Jones 1964; Clifford and Facey 1970; Kocher 1980; O'Brien and Sanna 1976). Use of such conversion factors usually is not important for higher energy quantum emissions (>200 keV gamma rays or X rays), but can be significant for lower energy emissions and for high density organs such as bone.

## Judgment Flags

Judgment flags are assigned to individual records, to annual dose cumulations for individuals, and even to individuals per se. Judgment flags relate to the suitability of data for use in epidemiologic analysis and are assigned by the dose assessor as a way of indicating problems that arose in the process of dose assignment. Judgment flags include those marking notional doses, questionable results, or gaps in monitoring such as an accident for which no dose estimate is available. Judgment flags can be used to describe flaws in data. Such flags are "no flaw," "minor flaw," "severe flaw," "fatal flaw" (do not use), and "pending" (unresolved potential flaw).

Another important kind of judgment flag denotes the possible, probable, or certain existence of a large unrecorded radiation exposure. The flag alerts the epidemiologist to the fact that the dose assessment procedure indicated that the worker may have been exposed significantly during an unmonitored accident. This flag should be placed onto the master roster of personnel in the study.

## Analysis of Dose Distributions

Characterization of annual dose distributions includes computations of the annual collective dose equivalent (or absorbed dose) for each type of radiation monitoring (gamma, beta, neutron, penetrating, and skin); the annual number of workers monitored; the mean and median annual doses; the standard deviation, variance, and the geometric standard deviation of the distribution; and the fractions of workers and the fractions of collective doses, or UNSCEAR fractions, (UNSCEAR 1977) received in dose ranges 0-MDD, 0-0.5, 0-1.5, and 0-5.0 rad. For years prior to 1961, it may be desirable also to compute the above fractions for the range 0-15 rad, since that value approximates the annual dose limit (12 rem) at that time. The numbers of flawed or adjusted records and any other judgment flags are plotted versus year. These should be printed to provide the reader a feel for the quality of the data as well as the problems associated with them. This information will prove important to the epidemiologist in determining suitable dose or

exposure categories. In instances where only job-title specific exposures or doses are computed, a more reasonable and useful summary will be average annual dose or exposure by year for each job title in the study.

### Internal Doses

The goal of dose assessment is to enable the use of data recorded by occupational radiation monitoring programs to infer a measure of the exposure for individuals in an epidemiologic study. Ideally, these data should be transformed to machine-readable records of annual doses (or dose equivalents) received by the total body or by one or more target organs for each worker for whom such doses can be inferred. At times, however, monitoring data may be sufficient only to develop a semiquantitative measure of organ burden or an estimate of radionuclide concentration in environmental media. Table 6-1 provides a listing of target organs currently considered to be of dosimetric significance by the International Commission on Radiological Protection (ICRP 1979) and the associated masses for those organs in normal adult.

In addition to the assignment of organ doses, dose assessment entails an examination of the validity of all procedures, algorithms, and instrumentation underlying the dose assignment. We will turn first to the problem of dose assignment, with the question of dose validation being deferred to a later section.

While the ICRP metabolic models (ICRP 1979) allow, in principle, the computation of doses to all target organs listed in Table 6-1, in many cases it is neither feasible nor desirable logistically to infer doses to all target organs. This is because the inferences usually will be based on measured activity in a single organ, thereby introducing very large uncertainties into doses computed for organs farther down the catenary chain or in other mammillary compartments. This uncertainty can be especially large in instances where large, isolated intakes occur for a worker, with no indication of the time of intake. In addition to such considerations, usually only a few organs are deemed dosimetrically significant, owing either to the large doses to them or to the organs' high radiosensitivity. For a facility with external exposures only, total body (and possibly skin)

are generally the target organs of greatest interest. For a facility with limited operations, such as a facility processing only uranium, total body and skin (for external exposures) and lung, bone, and perhaps kidney (for inhalation of uranium) are the target organs receiving the greatest doses. For a given exposure pathway to a given radiochemical, the target organs receiving the greatest doses generally are the "critical organs" specified in ICRP Publication 2 (ICRP 1960). Since the early work of the ICRP has been modified and improved over the years, the organs having the greatest "dose per unit intake" factors (Dunning et al. 1979) for the radiochemical and pathway in question should be the principal focus of attention, although in many cases they are the same as the ICRP Publication 2 critical organs. It also may be desirable to infer doses to organs that are particularly radiosensitive (ICRP Publication 26 lists "tissues at risk"), or to infer doses to organs that are involved in epidemiologic hypothesis generation (e.g., brain, for a brain cancer case-control study). Existing metabolic and dosimetric models have been developed to the greatest degree for organs and radionuclides of importance in health physics practice. In general, such models are strongest for organs of high radiosensitivity and those that receive the largest doses following occupational exposures.

When radioactive materials are introduced into the human body, some or all tissues may receive radiation doses. Unlike external exposures, which for the purposes of epidemiology are considered to result in fairly uniform doses over all parts of the body, internal doses generally are not uniform (ICRP 1979). For example, some elements such as calcium, radium, and plutonium, are bone-seekers, delivering considerably greater doses to parts of bone than to total body or other specific organs. Thus it is important to realize that "internal exposures" is a term that includes various levels of dose or dose equivalent to various organs, extending to the complicated considerations of microdosimetry (Roesch 1977; Goodhead 1982).

As mentioned previously, it is possible, in principle, to calculate or infer doses to each target organ on a list such as that in Table 6-1 for a well-characterized intake of radioactivity. Metabolic and dosimetric models have been published for this purpose (ICRP 1979; ICRP 1960), and compilations of dose commitments have been reported (Dunning et al. 1979). At present, doses cannot be inferred for organs for which there are no

Table 6-1

## Masses of Organs and Tissues of Reference Man\*

<u>Source Organs</u>	<u>Mass (g)</u>	<u>Target Organs</u>	<u>Mass (g)</u>
Ovaries	11	Ovaries	11
Testes	35	Testes	35
Muscle	28000	Muscle	28000
Red marrow	1500	Red marrow	1500
Lungs	1000	Lungs	1000
Thyroid	20	Thyroid	20
Stomach content	250	Bone surface	120
Small intestine content	400	Stomach wall	150
Upper lge. intestine content	220	Small intestine	640
Lower lge. intestine content	135	Upper large intestine	210
Kidneys	310	Lower large intestine	160
Liver	1800	Kidneys	310
Pancreas	100	Liver	1800
Cortical bone	4000	Pancreas	100
Trabecular bone	1000	Skin	2600
Skin	2600	Spleen	180
Spleen	180	Thymus	20
Adrenals	14	Uterus	80
Bladder content	200	Adrenals	14
Total body	70000	Bladder wall	45

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\* Table adapted from ICRP Publication No. 30 (1979, p. 16). Dunning et al. (1979) do not include muscle or skin target organs, use bone and endosteal cells instead of bone surface, and include total body, respiratory lymph nodes, and yellow marrow as target organs.

metabolic models for the radionuclide in questions (for example, uranium in brain). Dose assessment personnel could, in principle, develop their own models from the available data. However, a good general rule is that if the ICRP has not developed these metabolic models, either the data are insufficient or there is no indication that the organ is at significantly increased risk due to exposure. In the case of insufficient data, organ-specific doses must be replaced by exposure estimates, given in units of environmental concentration, or by integral organ burden in the first organ of the catenary chain (usually lung or G.I. tract).

Many criteria have been developed to judge whether a statistical association between a health effect and an exposure may be of a causal nature (Hill 1965; Rothman 1976; Reissland 1982). Among these is the criterion of biological plausibility, which states that "it must be biologically plausible that a given exposure could cause a given disease." While this criterion should not be construed as negating, a priori, any attempts at epidemiologic studies for organs with no past history of radiation-associated disorders, it does place some clear limits on the use of exposure variables. For example, if it is assumed that a disorder in organ x arises from damage induced by radiation to that organ, care must be taken to ensure that the exposure variable employed either is identical to, or in constant proportion to, the organ-specific dose. Such a consideration discourages any attempts to base epidemiologic studies on broad categories of internal exposures, that might result from the use of ordinal exposure indices at facilities dealing with a large assortment of radionuclides. In this case, there is no reason to assume that placement into a category of exposure is associated with a specific range of doses to a given organ.

A possible exception to the preceding discussion arises in the case of significant cross-irradiation. Cross-irradiation refers to the ability of radionuclides residing in one organ to irradiate (primarily through gamma emissions) other body organs. An example is the cross-irradiation that occurs from radionuclides in the lung, resulting in significant doses to other organs such as the brain (Eckerman, Ford, and Watson 1981). In general, this is important only for radionuclides with substantial gamma or high energy beta emissions (not including, for example, any naturally occurring nuclides of uranium). When cross-irradiation occurs, doses to one organ may prove to be proportional to doses in another, although this should be proven before accepted.

#### Calculation of Internal Organ Burdens from One or More In Vivo Radioactivity Measurements or Bioassay Results

Three types of monitoring data typically are collected by a health physics program. These consist of air monitoring results, in vivo measurements, and urinalyses. In addition, fecal samples may be collected

for some radionuclides. Through the use of metabolic models, it is possible to relate each of these monitoring results to estimates of organ-specific internal doses if one is willing to make some assumptions concerning the pattern of intake of the radionuclide. In the following section, the method for calculating internal doses from in vivo counting and urinalysis results is developed. No attempt is made to review the use of fecal samples, since those are not available in most instances. A later section in this chapter addresses the use of air monitoring measurements.

### General Considerations in Calculating Internal Organ Burdens

Unlike external exposures, which usually are reported as integrated exposures over time, results of measurements on internal organ burdens arrive in the epidemiology study as measures that may be related to dose rate. As will be described below, the results of such measurements can be converted to absorbed dose rates in various tissues such as lung, kidney, or whole body by the use of metabolic and dosimetric models. To calculate absorbed doses, these dose rates must be multiplied by the time period over which they were in effect. This multiplication is actually an integration of dose rate data over time, so that a general equation is

$$D(t_1, t_2) = \int_{t_1}^{t_2} D'(t) dt ,$$

where  $D(t_1, t_2)$  is the dose received between times  $t_1$  and  $t_2$ , and  $D'(t)$  is the dose rate as a function of time.

Algorithms for doing integrations must be developed for each type of monitoring, for each radionuclide, and for each exposure type (for example, solubility class for inhaled radionuclides), using appropriate models and related assumptions. This process is described below for in vivo counting data and urinalysis results. A summary of notation employed in this report appears in Table 6-2.

Table 6-2

## Summary of Notation for Internal Dosimetry

A	Approximate integral organ burden
AMAD	Activity median aerodynamic diameter
$A_{s,j}$	Coefficient associated with exponential retention function for $j^{\text{th}}$ radionuclide in organ S
Bq	Becquerel, a unit of activity equal to one disintegration per second
C	Concentration of radionuclide in air
d	Fraction of inhaled or ingested material deposited in the lung or G.I. tract
D	Absorbed dose
$D'$	Dose rate
$D^*$	Dose per unit intake
$D(t_1, t_2)$	Dose delivered between times $t_1$ and $t_2$
e	Expiration, breath pathway
E	Exposure level in an environmental compartment
$E(t)$	Activity eliminated via all pathways per unit time
$\Sigma$	Exact integral organ burden
f	Fraction of an interval of time during which a worker is exposed to an atmosphere
$f_e$	Feces pathway
$f_s$	Fraction of $q_T$ contained in organ S
$f'_{2,s}$	Fraction of radionuclide transferred from blood to organ S
FP	Fraction eliminated via pathway P
Fe	Fraction of radionuclide eliminated by exhalation
Ff	Fraction of radionuclide eliminated through feces pathway
Fp	Fraction of radionuclide eliminated by perspiration
Fu	Fraction of radionuclide eliminated through urine pathway
$F(s)$	Fraction of total elimination rate of radionuclide from the body which results from organ S
H	Dose equivalent
i	Index indicating the $i^{\text{th}}$ measurement or the $i^{\text{th}}$ individual in a population or the $i^{\text{th}}$ year of exposure
I	Intake rate of a radionuclide into the body

$I'$	Intake rate of radionuclide into the bloodstream
$j$	Index indicating the $j^{\text{th}}$ radionuclide in an organ or body
$\lambda_b$	Lambda b, biological removal rate constant
$\lambda_e$	Lambda e, effective removal rate constant
$\lambda_r$	Lambda r, radioactive removal rate constant
$\lambda_{s,j}$	Fractional rate of removal of the $j^{\text{th}}$ radionuclide from organ S
$m$	Mass of aliquot
$M_k$	The $k^{\text{th}}$ measurement of an organ burden
$M$	Mass of sample
$n$	Index referring to the $n^{\text{th}}$ year of exposure
$N$	Product of all other modifying factors
$N_a$	Number of days per year represented by routine monitoring for a population
$N_b$	Number of background counts
$N_b'$	Background count rate
$N_g$	Number of gross counts
$N_g'$	Gross count rate
$N_n'$	Net count rate
$N_{na}$	Number of bioassay results obtained during times of routine monitoring for a population
$P$	An elimination pathway
$p$	Perspiration pathway
$q$	The activity or organ burden
$q'$	Activity elimination rate
$q_0$	Initial activity in the body
$q(t)$	Total activity in the body at time $t$
$q_s(j, t_i)$	The activity in organ S from radionuclide $j$ at time $t_i$
$q_T$	Total activity in the body
$q'(j, P, t_i)$	Activity elimination rate for the $j^{\text{th}}$ radionuclide by pathway $P$ at time $t_i$
$Q$	Quality factor
$R(t)$	Retention fraction, fraction of initial intake remaining in organ at time $t$
$R_b(t)$	Biological retention function, the fraction of an initial intake remaining in the body at time $t$ after intake, accounting only for biological removal

$R'_b(t)$	Biological fractional removal rate
$R_e(t)$	Effective retention function, the fraction of an initial intake remaining in the body at time $t$ after intake, accounting for radioactive decay and biological removal
$R'_e(t)$	Effective fractional removal rate, the fraction of radionuclide eliminated from the body at time $t$ after intake, accounting for radioactive decay and biological removal
$R_r(t)$	Radiological retention function, the fraction of an initial intake remaining in the body at time $t$ after intake, accounting only for radioactive decay
$R'_r(t)$	Radiological fractional removal rate
$s$	Index indicating an organ $S$
$S$	Source organ, associated with the $S$ -factor
$S$ -factor	Conversion factor from cumulated activity to dose
$SD$	Standard deviation associated with a distributed variable
$t$	Time
$t_b$	Background counting time
$t_g$	Gross counting time
$t_i$	Time associated with the $i^{\text{th}}$ measurement of an individual
$t_{i,r}$	Time associated with the removal of the $i^{\text{th}}$ worker from exposure
$t_{i,s}$	Time associated with the start of exposure for the $i^{\text{th}}$ worker
$t_{na}$	Number of days per year represented by routine monitoring for a population
$t_{boy}$	Time associated with the beginning of a year (at times simplified to BOY)
$t_{eoy}$	Time associated with the end of a year (at times simplified to EOY)
$t(i),$ $t(i+1)$	A pair of time values, $t(i+1)$ later than $t(i)$
$t_1, t_2$	A pair of time values, $t_2$ later than $t_1$
$T$	Target organ, associated with the $S$ -factor
$T_a$	Number of person-days in a population for which bioassay results were associated with known intakes
$T_b$	Biologic half-life
$T_c$	Time over which a bioassay sample is collected
$T_e$	Effective half-life
$T_r$	Radioactive half-life

u	Urine pathway
U	Cumulated activity
U(j,s, annual)	Cumulated activity from radionuclide j in organ S over the course of an entire year
U(j,s, t <sub>2</sub> -t <sub>1</sub> )	Cumulated activity from radionuclide j in organ S between times t <sub>1</sub> and t <sub>2</sub>
v	Volume of aliquot
v'	Breathing rate, volume/time
V	Total daily volume
X <sub>a</sub>	Organ burden result obtained as a result of an accident
X <sub>na</sub>	Organ burden result obtained during times of routine (non-accident) monitoring
X <sub>i</sub>	The i <sup>th</sup> parameter X
Y <sub>c</sub>	Counting yield
Y <sub>t</sub>	Total yield, including both radiochemical and counting
Y <sub>r</sub>	Radiochemical yield
Y(t)	Fraction of initial burden eliminated per day
Y <sub>c</sub> (i,j,s)	Counting yield for radionuclide j and organ S in individual i
z	Fraction of workday
μ	The mean value associated with a distributed variable, not to be confused with the designation for "micro" when used with Ci, that is, μCi

The first step in calculating internal doses lies in determining the organ burden  $q_s(j, t_i)$  of radionuclide j in organ s at time  $t_i$ . The organ burden should be calculated in units of activity (e.g. Bq, dpm, μCi). Since S-factors (Dunning, Pleasant, and Killough 1977) are reported in units of rem per μCi-day of residence in an organ, a good general rule is to calculate the organ burdens in units of μCi (these S-factors will be utilized in a later step). This is true for all forms of internal monitoring data. The method by which this calculation is accomplished is first described for in vivo radioactivity measurements and then for bioassay results.

## Use of In Vivo Measurement Results for Estimating Organ Burden

The following method assumes that the  $i^{\text{th}}$  radiation emission of the  $j^{\text{th}}$  radionuclide has been counted by gamma spectroscopy. If more than one radionuclide has been measured, the steps must be repeated for each radionuclide, or some functional relationship may be assumed between the various radionuclide activities. These functional relationships are typically assumed in the case of commonly occurring mixtures of uranium nuclides, for which typically encountered ratios of activities are found.

The paper by Cofield (1960) presents a review of the in vivo counting method for uranium. In general, the in vivo measurement method consists of counting the number of gamma rays emerging from an organ, subtracting a background count, and multiplying the result by a calibration factor ( $\mu\text{Ci}$  per count per minute) obtained from measurements on a phantom containing known activity.

From measurements of gross counts ( $N_g$ ) and gross counting time ( $t_g$ ), one calculates the gross count rate:

$$N_g' = N_g/t_g \text{ (counts per unit time) .} \quad (6-6)$$

(Throughout the remainder of this report, a primed quantity indicates a temporal rate, or the rate of change of that quantity with time.) This result is associated with an individual worker and an "instant" in time ( $t_i$ ; e.g.,  $t_i = 8:55 \text{ a.m., August 12, 1948}$ . This value should not be confused with counting times,  $t_g$  or  $t_b$ ). From measurements or estimates of background counts ( $N_b$ ) and background counting time ( $t_b$ ), one then calculates the background count rate:

$$N_b' = N_b/t_b \text{ (counts per unit time) .} \quad (6-7)$$

For in vivo counting, the selection of an appropriate background is often difficult. For the example of uranium in lung, problems arise due to choice of person to use for background because of individual differences such as chest wall thickness, height, weight, potassium-40 and cesium-137 contents (Scott and West 1967; Scott et al. 1969; Scott and West 1975). In later

years of monitoring, however, background in a particular energy region often was determined from other energy regions.

Using the above results, one calculates the net count rate in the specific energy region of interest:

$$Nn' = Ng' - Nb' \text{ (counts per unit time) .} \quad (6-8)$$

Specifically,  $Nn'$  is the net count rate associated with the  $i^{\text{th}}$  emission of radionuclide  $j$  in source organ  $S$ . If the  $j^{\text{th}}$  radionuclide is found in more than one source organ (for example, U-235 in lung and U-235 on skin of chest) (Scott and West 1975), then  $Nn'$  is not due entirely to radionuclide  $j$  in source organ  $S$ . This is discussed in more detail below.

Counting yield (or calibration factor) is a parameter pertaining to a particular machine, detector, assumed geometry, and configuration of components such as dials, knobs, switches, voltages. It must be sought from the literature or from the purveyor of in vivo counting service, or it can be postulated to be a given value if enough is known about the system. Defining counting yield  $Yc(i,j,S)$  as

$$Yc(i,j,S) = \frac{\text{(net counts per unit time)}}{\text{(activity in organ S)}}, \quad (6-9)$$

for the  $i^{\text{th}}$  emission of radionuclide  $j$  in source organ  $S$ , the activity at time  $t_i$  in organ  $S$  is

$$q_s(j,t_i) = \frac{Nn'(i,j,S)}{Yc(i,j,s)}. \quad (6-10)$$

#### Separating Contributions from Organs

If the net count rate  $Nn'$  is due to activity in source organs  $S_1$  through  $S_m$ , then the activity in  $S_1$  is less than the total measured activity by a factor that is a function of the relative amounts of activity in the various source locations. If the relative amounts of activity are different from one worker to another, as they would be if workers had significant and

different amounts of surface contamination on their skin, then  $Yc(i,j,S)$  will differ from one worker to another, and uncertainty increases in the organ burdens inferred for such workers. In the case of lung counting for uranium, one should assume that if no alpha surface contamination is detected, all the net counts are due to uranium in lung. This, of course, ignores the contributions from uranium in bone or pleural membrane, which may be significant for workers exposed for many years (Crawford-Brown and Wilson 1984).

Consider, for example, the case of an in vivo measurement that results in the detection of gammas from  $n$  source organs, each organ,  $S$ , being characterized by a counting yield  $Yc_s$ . The net count rate then is equal to

$$\sum_{S=1}^n Yc_s q_s = Nn' \quad (6-11)$$

where  $q_s$  is the activity in the  $S^{\text{th}}$  organ at the time of the measurement. Clearly the contribution from the organ  $S$  to the net counts is  $Yc_s q_s$ . Let  $q_T$  be the total burden of the radionuclide in the body and let  $f_s$  be the fraction of this burden represented by the contents of organ  $S$ . Then  $q_s = f_s q_T$ . Denoting the net count rate by  $N'n$ , it may be noted that

$$Nn' = \sum_{S=1}^n Yc_s f_s q_T$$

or

$$q_T = Nn' / \sum_{S=1}^n f_s Yc_s \quad (6-12)$$

From a knowledge of the  $f_s$  values, the activity in organ  $S$  may be calculated as

$$q_s = f_s q_T \quad (6-13)$$

The remaining problem consists of a determination of the values for  $f_s$  to be employed. An estimate of these values may be obtained if either the exposures were constant for a worker throughout his or her time in the work force or if the change in exposure was on a time scale large compared to the

longest effective half-life of the radionuclide in the body organs. In the first case, let  $I'$  be the constant rate of intake of a radionuclide into the bloodstream and let  $R_s(t)$  be the retention function for this radionuclide in organ  $S$  as a function of time. Similarly, let  $f'_{2,s}$  be the deposited fraction of the radionuclide in organ  $S$  (all such values obtained from, for example, ICRP Publication 30 [ICRP 1979]). At time  $t$  after the onset of exposure, the activity in organ  $S$  is given by

$$q_s = \int_0^t f'_{2,s} I' R_s(t) dt . \quad (6-14)$$

As a result, it may be seen that

$$f_s = q_s / q_T$$

$$= \int_0^t f'_{2,s} I' R_s(t) dt / \sum_s \int_0^t f'_{2,s} I' R_s(t) dt . \quad (6-15)$$

If the changes in exposure status are determined to be on a scale large compared to the time needed for  $R_s(t)$  to return to negligible values, then it may be assumed that the values of  $f_i$  represent values typical of a steady state. The steady state solutions to equation 6-15 will depend on the particular functional forms of the individual functions,  $R_s(t)$ . If a general multiple exponential function is employed for the retention functions, then

$$R_s(t) = \sum_{j=1}^n a_{s,j} e^{-\lambda_{s,j} t} \quad (6-16)$$

where the coefficients  $a_{s,j}$  and  $\lambda_{s,j}$  represent the fractional deposition and removal rate constants, respectively, for the  $j^{\text{th}}$  compartment of organ  $S$ . The steady state content of organ  $S$ , then is proportional to

$$f'_{2,s} \sum_{j=1}^n \frac{(a_{s,j})}{(\lambda_{s,j})} .$$

In this case, it may be shown that

$$f_s = \frac{f'_{2,s} \sum_{j=1}^n \frac{(a_{s,j})}{(\lambda_{s,j})}}{\sum_s f'_{2,s} \sum_{j=1}^n \frac{(a_{s,j})}{(\lambda_{s,j})}} \quad (6-17)$$

where the summation on the index  $s$  extends over all organs with significant burdens. In general,

$$\lambda_{s,j} = \frac{0.693}{T_b} + \frac{0.693}{T_r} ,$$

where  $T_b$  and  $T_r$  are the biological and radiological removal half-lives, respectively, for radionuclide  $j$  in organ  $S$ .

At this point, the organ burden of radionuclides  $j$  in organ  $S$  at time  $t_i$ ,  $q_s(j,t_i)$ , may be used in the next stage of dose calculations; the calculation of cumulated activity from organ burden values. Before discussing the cumulated activity calculations, it is important to outline the steps in the inference of an organ burden from bioassay results.

#### Use of Bioassay Results for Estimating Organ Burden

An organ burden may be inferred from one or more bioassay measurements. A bioassay measurement is defined as a measurement of the amount of radioactivity in excreta or other output from the body. This output concentration is related functionally to the burdens of the radionuclides in the body organs. The primary pathways,  $P$ , of elimination of radioactivity from the body that are of use in bioassay measurements are urine ( $P=u$ ), feces ( $P=fe$ ), sweat (perspiration;  $P=p$ ), or breath (exhalation;  $P=e$ ) (ICRP 1968). In many cases, considerable sample preparation, such as drying, ashing, distilling, or radiochemistry, may have been required prior to counting. The radiochemical yield,  $Y_r(j)$ , of any sample preparation procedures must be determined or assumed. Radiochemical yield is the fraction of element  $j$  in the sample (or aliquot) that is recovered by

radiochemical procedures. In the case of uranium urinalysis, a fluorometric or other nonradiologic technique may be most sensitive or most practical for quantitating the amount of radionuclide present (McRee, West, and McLendon 1965; Lippman 1959).

Often only a small aliquot from a bioassay sample is analyzed. If a radiometric technique is used, the steps in arriving at a net count rate for the  $i^{\text{th}}$  emission of radionuclide  $j$  in an aliquot from elimination pathway  $P$ ,  $Nn'(i,j, \text{aliquot}, P)$  are identical to those outlined above for calculating  $Nn'(i,j,S)$  from in vivo measurements. The selection of an appropriate background for bioassay may be less difficult for bioassay of alpha or gamma emitters than for in vivo measurements, because counting geometry differences can be minimized. On the other hand, urine bioassay for pure beta emitters can be frustrated by greatly varying amounts of naturally occurring radioactive potassium-40 in individual samples from one day to the next; thus the selection of background becomes less certain (Sedlet 1982).

Counting yield,  $Yc(i,j)$ , for the  $i^{\text{th}}$  emission of radionuclide  $j$  must be determined by counting a known standard with the same geometry and self-absorption as the unknown samples. For epidemiology, counting yield is ordinarily determined (if needed) in the "Program Evaluation" stage of dose assessment. Counting yield is not needed if monitoring results from the facility are already expressed in units of activity (i.e., the bioassay result is reported in units of activity per unit volume of urine).

The total yield for the  $i^{\text{th}}$  emission from radionuclide  $j$  is the product of the radiochemical yield and the counting yield:

$$Yt(i,j) = Yr(j)Yc(i,j) . \quad (6-18)$$

The activity of the aliquot that was counted is the quotient of the net count rate of the nuclide or nuclide mixture and the total yield:

$$q(j, \text{aliquot}, P) = \frac{Nn'(i,j, \text{aliquot}, P)}{Yt(i,j)} . \quad (6-19)$$

Two steps are needed to infer an organ burden from  $q(i,j, \text{aliquot}, P)$ . First, one must decide how the activity in the aliquot is related to the total elimination of radioactivity by pathway  $P$  per unit time. Note that

$q'$ , the rate of elimination, is a daily rate of elimination of radioactivity (activity eliminated per day), expressed, for example, in dpm/24hr, or  $\mu\text{Ci}/\text{day}$ .

Second, one must decide how  $q'(j,P,t_i)$  is related to the organ burden  $q_S(j,t_i)$ . If a model is adopted that assumes the  $j^{\text{th}}$  radionuclide is eliminated through pathway P in proportion to volume (or mass) of excreta eliminated per day from the organ, then the daily elimination of radionuclide j is related to the activity in an aliquot by the ratio of the total volume (or mass) eliminated daily from the organ to the volume (or mass) of the aliquot:

$$q'(j,P,t_i) = \frac{\text{volume}(\text{total daily}) q(j,\text{aliquot},P)}{\text{volume}(\text{aliquot})} \quad (6-20)$$

Mass can be substituted for volume in both the numerator and denominator of the above equation if it is more appropriate. If the daily volume or mass eliminated from an organ via pathway P is not measured or otherwise known for individuals, then values for reference man can be used (ICRP 1975, pp. 343-365). Of course, since reference man values are simply central tendencies for populations, and since there are large departures from these values for individuals, both systematically among individuals and daily within an individual, the use of reference man values increases the variance of  $q'(j,P,t_i)$  considerably when compared with use of measured values for individuals (such as 24-hour urine or feces collection). Reference man values may be different by a factor of 2 or more for individuals (ICRP 1975).

Tritium (as oxide; that is, tritiated water) is an example of a radionuclide for which rate of elimination via urine is directly proportional to daily urine volume (ICRP 1968). If a tritium urinalysis result is reported in units of  $\mu\text{Ci}/\text{ml}$ , and reference man produces 1400 ml of urine per day (ICRP 1975), then the rate of elimination of  $^3\text{H}$  from the body is

$$q'(^3\text{H}, \text{daily, urine}) = V(1400 \text{ ml/day}) \cdot q(^3\text{H}, 1 \text{ ml, urine}) \quad (6-21)$$

If one adopts a model assuming that the  $j^{\text{th}}$  radionuclide is eliminated at a relatively constant rate per day, independent of the total mass or volume of matter eliminated per day and being only a function of the organ burden, then the activity in the aliquot is time-weighted to infer a daily excretion rate via this pathway,  $q'(j,P,t_i)$ . To make this inference,  $q(j,\text{aliquot},P)$  is first multiplied by the ratio of the mass (or volume) of the sample to the mass (or volume) of the aliquot to obtain

$$q(j,\text{sample},P) = \frac{M(\text{mass of sample})}{m(\text{mass of aliquot})} \cdot q(j,\text{aliquot},P) \quad (6-22)$$

This result then is multiplied by the ratio of 24 hours divided by the number of hours of elimination that the sample represented, thereby yielding the activity of radionuclide  $j$  eliminated each day via pathway  $P$ :

$$q'(j,P,t_i) = \frac{(24 \text{ hours})}{(\text{time between eliminations})} \cdot q(j,\text{sample},P) \quad (6-23)$$

Uranium is an example of a radionuclide that is eliminated in urine at a rate that is largely independent of daily urine volume (Quastel et al. 1970). For urinalysis, an alternative to the time-weighting calculation described above is to weight a sample by the fraction of standard daily creatinine excretion (Jackson 1966; Quastel et al. 1970; ICRP 1975), if creatinine has been measured for that sample. This assumes that radionuclide elimination is proportional to creatinine elimination and, in any event, such data rarely will be available. Note that equations 6-20 and 6-23 are identical when the rate of elimination of biological medium is constant.

Once  $q'(j,P,t_i)$  has been calculated, it is necessary to infer the activity of nuclide  $j$  eliminated by all pathways,  $q'(j,\text{all } P,t_i)$ , from the organ (or organs). Letting  $FP$  denote the fraction of total daily excretion that occurs via the pathway that forms the basis for the bioassay measurement  $P$  ( $FP = F_u, F_f, F_p,$  or  $F_e$  for the urine, feces, perspiration, and exhalation pathways, respectively), one finds

$$q'(j,\text{all } P,t_i) = \frac{q'(j,P,t_i)}{FP} \quad (6-24)$$

It generally is assumed, in the absence of renal disease, that all radionuclides excreted in urine have passed through the kidneys and have therefore come from extracellular fluid. For feces, the case is different. Klaassen has stated,

Appearance [of toxic compounds] in the feces can be due to a number of factors: (1) the chemical was not completely absorbed after oral ingestion, (2) it was excreted into the bile, (3) it was secreted in the saliva, in the gastric or intestinal secretory fluid, or in the pancreatic secretion, and/or (4) it was secreted by the respiratory tract and then swallowed (Doull, Klaassen, and Amdur 1980).

From a knowledge of excretion rate of radioactivity in feces, one cannot unambiguously infer blood content, nor bone content, in the absence of knowledge of time of intake (intake history) or in the presence of the possibility of chronic or sporadic exposure (ICRP 1971). Thus, deducing an organ burden from bioassay of feces is more tenuous than deducing one from urinalysis, unless the target sites of interest are stomach, small intestine, upper large intestine, or lower large intestine. Because of the large uncertainties involved in inference of systemic organ burdens from fecal analysis, we recommend that results of fecal analysis not be used for purposes of dose assignment in epidemiology, although they may form a reasonable semiquantitative index of exposure to a radionuclide.

It should be noted that  $q'(j, \text{all } P, t_i)$  (for this individual) is analogous to the ICRP's  $E(t)$  function for the daily elimination rate via all pathways of nuclide  $j$  evaluated at time  $t_i$  and is identical to  $E(t)$  when  $q_s$  is replaced by  $q_T$ . The function  $E(t)$  is defined by the ICRP as "the amount ( $\mu\text{Ci}$ ) of radionuclide excreted per unit time at time  $t$ " (ICRP 1968), but it is for a single radionuclide and for all routes of biological removal and for all organs containing the radionuclide. To keep notation as consistent as possible with that of the ICRP,  $E(t)$  and related functions  $q(t)$ ,  $Y(t)$ , and  $R(t)$  will be used from here on, bearing in mind that the functions refer to radionuclide  $j$ , total body as source organ, and all routes of elimination combined (ICRP 1968). The function  $q(t)$  is the activity in total body,  $R(t)$  is the fractional retention function equal to  $q(t)/q_0$ , and  $Y(t)$  is the fractional excretion function equal to  $E(t)/q_0$  where  $q_0$  is the initial

intake value. The relationships between these functions are clearly laid out in equations B1 through B4 of ICRP Publication Number 10 (1968), followed by a discussion of the most elementary forms of  $E(t)$ . The value of  $q(t)$  can be inferred from measurements of  $E(t)$  under some circumstances, as discussed below. To keep clearly separate the mathematical construct  $q(t)$  and an inferred value of  $q(t)$ , the latter is denoted by  $q$  (total body,  $t_i$ ) in the discussion that follows.

#### Mathematical Relations Between Excretion and Organ Burden

In ICRP notation, both  $R(t)$  and  $E(t)$  include the effects of radioactive decay.  $R(t)$  as used in ICRP Publication 10 (1968, pp. 1-29) is more correctly  $R_e(t)$ , where the subscript  $e$  denotes "effective." As explained in the note (ICRP 1968, p. 29), the retention functions given in the Appendices to that report are  $R_b(t)$ , only the biological component of  $R_e(t)$ . The radiological retention function,  $R_r(t)$ , is always  $\exp(-\lambda_r t)$ . The effective retention function,  $R_e(t)$  is always the product of  $R_r(t)$  and  $R_b(t)$ , that is,  $\exp(-\lambda_r t)R_b(t)$ . The ICRP's failure to use subscripts  $e$ ,  $b$ , and  $r$  for the various retention functions or to use consistent notation from one publication to another has led to confusion among those doing and teaching internal dosimetry. The subscripts are used here to minimize confusion.

Since  $R_e(t) = R_b(t)R_r(t)$ , and using primes to denote time derivatives,

$$\begin{aligned}
 R'_e(t) &= d/dt[R_b(t)R_r(t)] \\
 &= R'_b(t)R_r(t) + R_b(t)R'_r(t) \quad [\text{by the chain rule}] \\
 &= R'_b(t)\exp(-\lambda_r t) - \lambda_r R_b(t)R_r(t) \\
 &= -Y(t) - \lambda_r R_e(t) .
 \end{aligned}
 \tag{6-25}$$

This is equation B2 from ICRP Pub. 10 (1968), with intermediate steps added to emphasize the interrelationships between the various quantities. From the definitions of  $q(t)$ ,  $E(t)$ ,  $R_b(t)$ , and  $Y(t)$ ,

$$\begin{aligned}
 q(t)/E(t) &= R(t)/Y(t) \\
 &= -R_b(t)/R'_b(t) .
 \end{aligned}
 \tag{6-26}$$

The last term is the number that the daily, all-pathway excretion rate is to be multiplied by to calculate the instantaneous body burden  $q(t)$  needed for dosimetry. The quantity  $q(t)/E(t)$  is called the "body burden per unit excretion rate" and has the units of time determined by  $E(t)$  (e.g., days, if  $E(t)$  is in activity/day). [In ICRP Publication 10A (1971),  $R_s(t)$  is used to denote  $R_b(t)$ , and  $r_s(t)$  is used for  $R_e(t)$ . The "s" denotes "single deposition".]

#### Effect of Form of Biological Retention Function on Inference of $q(t)$ Following a Single Intake

In 1967 Committee 4 of the ICRP pointed out that three forms of biological retention functions are commonly encountered: sums of exponential functions (of which a single exponential is a special case), a power function, or an exponential function and a power function (ICRP 1968). As theories of internal dosimetry have evolved, the above functions have proved to be over-simplifications in many cases (ICRP, Publication 19, 1972; ICRP, Publication 20, 1972; ICRP 1979; Eckerman, Ford, and Watson 1981), and a fourth general form was introduced for alkaline earths. Equations for the  $Y(t)$  and  $R_e(t)$  (and therefore for  $R_b(t)$ ) are given in ICRP Publication 10 (ICRP 1968) and ICRP Publication 20 (IRCPA 1972), and are not derived here. Results for the ratio  $[-R_b(t)/R_b'(t)]$ , that is, the body burden per unit excretion rate following a single intake, are given below for five cases. In each case,  $q(t) = q(0)R_e(t)$  and  $E(t) = -q(0)R_e'(t)$ .

##### 1. Single Exponential Biological Retention Function

The simplest case of a biological retention function is the single exponential. The model underlying a single exponential retention function is simply that all of the radionuclide taken in goes to a single compartment or pool. A constant fraction of the activity in the compartment is removed to excreta per unit time. The rate constant for removal of chemical element  $j$  by normal biological processes is  $\lambda_b$ ; and  $R_b(t) = \exp(-\lambda_b t)$ . The effective rate constant for removal by both radioactive decay and biological processes is  $\lambda_e = \lambda_r + \lambda_b$ . Under this model, the activity per unit time

(e.g.,  $\mu\text{Ci/day}$  or  $\text{Bq s}^{-1}$ ) eliminated by biological processes (the part that can be measured in excreta) at any time  $t$  following an intake is

$$E(t) = \lambda_b q(t) , \quad (6-27)$$

(ICRP 71, p. 27) so that

$$q(t)/E(t) = 1/\lambda_b ,$$

or

$$q(\text{total body}, t_i)/q'(j, \text{all P}, t_i) = 1/\lambda_b . \quad (6-28)$$

Thus the body burden can be inferred from excretion data by use of the single constant parameter  $\lambda_b$  when the biological retention function has the form of a single exponential.

## 2. Multiple Exponential Biological Retention Function

When  $R_b(t)$  is a series of exponentials,

$$R_b(t) = \sum_i K_i \exp(-\lambda_{i,b} t) ,$$

or

$$R_e(t) = R_r(t) \sum_i K_i \exp(-\lambda_{i,b} t) \quad (6-29)$$

(equation B12 from ICRP 1968) the body burden  $q(t)$  is not a constant function of  $E(t)$  over time and can vary by orders of magnitude depending on the values of individual parameters. In general,

$$q(t)/E(t) = \frac{\sum_i K_i \exp(-\lambda_{i,b} t)}{\sum_i \lambda_{i,b} K_i \exp(-\lambda_{i,b} t)} . \quad (6-30)$$

### 3. Simple Power Law Biological Retention Function

When the biological retention function is

$$R_b(t) = A(t + x)^{-n} , \quad (6-31)$$

where  $A = x^n$ ,  $0.01 \leq x \leq 10$  days,  $0 < n < 1$  (equation B14 from ICRP 1968). Equation 6-31 then yields

$$q(t)/E(t) = (t + x)/n . \quad (6-32)$$

This, too, is a distinct function of time, increasing linearly with time for times  $t \gg x$ .

### 4. Combination Exponential and Power Law Biological Retention Function

When the biological retention function is

$$R_b(t) = K \exp(-\lambda_b t) + (1 - K)(t + x)^{-n} , \quad (6-33)$$

where terms are defined as above (equation B15 from ICRP 1968), then

$$q(t)/E(t) = \frac{K \exp(-\lambda_b t) + (1 - K)(t + x)^{-n}}{\lambda_b K \exp(-\lambda_b t) + n(1 - K)(t + x)^{-(n + 1)}} . \quad (6-34)$$

### 5. Alkaline Earth Biological Retention Function

The ICRP Publication 20 (ICRP 1972) biological retention function is

$$R_b(t) = (1 - p) \exp(-mt) + p \epsilon^b (t + \epsilon)^{-b} [\beta \exp(-r\lambda t) + (1 - \beta) \exp(-\sigma r \lambda t)] \quad (6-35)$$

where the various terms are defined in that reference. For this form,

$$q(t)/E(t) = \{(1 - p)\exp(-mt) + p\varepsilon^b(t + \varepsilon)^{-b}[\beta\exp(-r\lambda t) + (1 - \beta)\exp(-\sigma r\lambda t)]\} / \{m(1 - p)\exp(-mt) + p\varepsilon^b(t + \varepsilon)^{-(b + 1)}[\beta\exp(-r\lambda t) + (1 - \beta)\exp(-\sigma r\lambda t)] + p\varepsilon^b(t + \varepsilon)^{-b}[r\lambda\exp(-r\lambda t) + \sigma r\lambda(1 - \beta)\exp(-\sigma r\lambda t)]\} .$$

(6-36)

This function changes dramatically with time.

#### Effect of Form of Biological Retention Function on Inference of $q(t)$ for Chronic Intakes

The problem of the temporal evolution of  $q(t)/E(t)$  may be avoided for the single and multiple exponential biological retention functions if exposures are assumed to be chronic and if the half-lives associated with the  $\lambda_{i,e}$  parameters are small compared to the time period of the study. The latter assumption is equivalent to assuming that intake and excretion are in equilibrium, that is  $E(t) = I'$ , where  $I'$  is the daily rate of intake of the radionuclide into the body through the lung, G.I. tract, or skin. For the special case of the single exponential model,

$$q(t) = I'/\lambda_{i,e} , \quad (6-37)$$

while for the multiple-exponential model,

$$q(t) = I' \sum_1 K_i / \lambda_{i,e} . \quad (6-38)$$

Thus the body burden per unit excretion rate becomes

$$q(t)/E(t) = 1/\lambda_{i,e} \quad (6-39)$$

for the single exponential model and

$$q(t)/E(t) = \sum_1 K_i / \lambda_{i,e} \quad (6-40)$$

for the multiple exponential model. Use of this equation clearly requires that changes in the intake rate occur on a scale that is large compared to the longest significant retention component in  $R(t)$ . Values for  $K_i$ ,  $\lambda_{i,e}$ , and  $\lambda_{i,r}$  may be found in ICRP Publication 30 (ICRP 1979), from the ORNL-ICRP Data Base (Watson, Fore, and Ford 1981) or from other tabulations (Killough et al. 1978; Dunning et al. 1979).

An alternative to the use of ICRP values for  $\lambda_b$  is to consult the literature directly. For example, data published in the open literature by the Y-12 health physics staff revealed that the ICRP value of 500 days for clearance from lung of the uranium compounds found at Y-12 was too high; a value of 115 days was a more appropriate value based on the literature survey. It must be noted that the choice of  $\lambda_b$  will be too high for some workers and too low for others, sometimes randomly and sometimes systematically, sometimes changing from too low to too high systematically over a worker's lifetime as some radioactivity accumulates in slow turnover compartments (e.g., uranium in bone). The biologic half-life is the source of a great deal of uncertainty in internal doses inferred from occupational records.

When hundreds of thousands of records, made in the course of monitoring a population of thousands of workers, are used as a basis for inferring doses, a single exponential retention function has two attractive features for computational purposes. First, the simplicity of computation is important, because complex calculations may be too expensive and time consuming. Secondly a single exponential retention function can be used to infer  $q(t)$  unambiguously with no assumption about the intake or excretion history of the individual. (There still remains, however, the problem of calculating values of  $q(t)$  at times prior to the measurement, as intake time must be known to extrapolate  $q(t)$  backwards in time from the measurement.) Since time(s) of intake may not be known in an epidemiologic study, power function, or power function and multiple exponential retention functions, require more assumptions of parameter values than does a single exponential retention function. Of course, choice of a given retention function must be based on empirical findings and not on consideration of convenience. An alternative to forcing a simple retention function into a study is to assume that the organ burden for each individual is a smoothly and slowly varying

function of time, so that the measurements may be assumed to be representative of all times between measurements.

It is natural to ask why each individual's retention function cannot be inferred directly from his or her bioassay measurements. This inference can be made only if there are no intakes after a certain time, followed by a series of measurements, or if intakes are of known magnitude and time. Since information about individual intakes generally is not available with occupational records, and since intakes can be expected to occur chronically or sporadically between bioassays, a worker's retention function cannot, in general, be inferred from his or her occupational records. If an attempt is made to infer  $R(t)$  from a worker's bioassay data, intakes between measurements will result in an apparent increase in the tenacity with which the material is retained. Large sporadic intakes also negate the assumption stated above that organ burdens vary smoothly and slowly with time.

#### Estimating Burdens for Specific Organs

The discussion to this point has focused on the relation between the bioassay result and an estimate of total body burden, defined as the amount of radionuclide anywhere in the body. For dosimetric purposes, it is necessary to obtain an estimate of the burden in each separate body organ, since for most radionuclides (except, for example, tritium) distribution is highly inhomogeneous.

Of the total amount of radionuclide  $j$  excreted via pathway  $P$  per day, it is necessary to estimate what fraction,  $F(s)$ , came from each possible source organ  $S$ . Such information may be sought from the ICRP (ICRP 1960; ICRP 1968; ICRP 1971; ICRP 1975; ICRP 1979). When each organ is characterized by retention functions with single exponentials, the fraction of the excreted radionuclide that came from source organ  $S$  is

$$F(s) = \frac{\lambda_{b,s} q_s(t)}{\sum_i \lambda_{b,i} q_i(t)}, \quad (6-41)$$

where  $\lambda_{b,s}$  is the removal rate constant from organ  $S$ ,  $q_s(t)$  is the activity in organ  $S$ ,  $\lambda_{b,i}$  is the removal rate constant for organ  $i$ ,  $q_i(t)$  is the

activity in organ  $i$ , and the summation is over all organs. The organ burden of radionuclide  $j$  in organ  $S$  at time  $t$  is

$$q_s(j,t) = \frac{F(s)}{\lambda_{b,s}} \sum_i \lambda_{b,i} q_i(j,t) = \frac{F(s)E(t)}{\lambda_{b,s}}, \quad (6-42)$$

where

$$\sum_i q_i(j,t) = q(j, \text{total body}, t) .$$

Note that in cases where a radionuclide is removed with the same half-life from all organs

$$q_s(j,t) = F(s) q(j, \text{total body}, t) .$$

Often one organ will predominate, such as the thyroid for iodine, bone for the alkaline earths, or lung for insoluble (class Y) alpha or beta emitting particulates that have been inhaled; in these cases,  $F(s) = 1$  and  $q_s(j,t) = q(t)$ . In more complicated cases involving significant uptakes by several organs, the temporal pattern of individual organ burdens must be reconstructed for each year prior to a year in which doses are being computed. Equation 6-41 is employed, with the relative values for  $q_i(t)$  being computed as in equation 6-15 with  $I'$  being made an explicit function of time. In the absence of data for preceding years during which significant exposures are known to have occurred, no meaningful doses can be reconstructed unless a constant intake rate is assumed.

For the special case of inhalation of radionuclides, the activity in lung can be inferred from urine measurements only by using data or assumptions about the solubility or solubility class (ICRP 1979) of the inhaled material. Soluble material (ICRP solubility class D) is assumed to clear from the lung quite quickly (effective half-life less than one day), while less soluble material is assumed to clear more slowly (effective half-lives on the order of weeks or years) (ICRP 1979). It is clear that the retention function for lung has a very strong dependence on solubility class, and thus inference of lung burden from urinalysis is strongly dependent on solubility class.

In a case where workers are exposed to a mixture of solubility class D and solubility class Y compounds of a radionuclide for which the physical half-life is on the order of tens or hundreds of days, taking a urine sample at the end of a weekend, before a worker returns to potential exposures, is a way of attempting to separate class Y from class D exposure. In principle, most of the class D activity should be gone from the body 48 to 60 hours after exposure ends, so that the after-weekend sample should represent primarily excretion of class Y radioactivity. Since class Y activity contributes much more dose per unit activity than does class D activity in cases of mixed exposures, inferring long-term lung burdens from after-weekend urinalysis will account for most of the dose.

Lippmann has stated that "an after-weekend urine sample does not provide an accurate indication of uranium body burden" (Lippmann 1959). The ICRP has stated that "only in individuals who do not have an appreciable preexisting body content is it possible to relate urinary levels (of uranium) to body or organ content" (ICRP 1968). Both statements are true if organ burdens that are off by a factor of 2 or more are unacceptable, as would be the case in health physics practice, where individual doses, and not group means, are essential. However, organ burdens can indeed be computed using simple assumptions, bearing in mind the fact that each assumption of a parameter represents some central tendency for a population that has a wide distribution for individuals. This approach can be summarized by the idea that urine measurements contain at least some information about body or organ burdens, and that doses inferred from such measurements, along with a range of confidence for the inferences, may prove more useful for epidemiology than no inferences, or categorization based on raw, unreduced data (hazard index grouping).

A summary of the inference of organ burden from bioassay measurements is beneficial at this point. For the case of a radionuclide that is eliminated in proportion to volume (or mass) of excreta,

$$q_s(j, t_i) = \frac{F(s) V N n'}{\lambda_b FP v Yr Yc} \quad (6-43)$$

For the case of a radionuclide that is eliminated at a more or less constant rate per day, regardless of total volume or mass of excreta,

$$q_s(j, t_i) = \frac{24 F(s) M N n'}{\lambda_b F P T m Yr Yc} \quad (6-44)$$

It may be possible to relate  $E(t)$  and  $q(t)$  directly through paired measurements of organ burden (in vivo counting) and urinalysis results. There are times (such as at the Y-12 facility) when the health physics staff at a facility will perform in vivo measurements and urinalyses on an individual at approximately the same time. If both measurements are directed towards the same radionuclide, the ratio of the paired measurements might provide a better indication of the ratio  $E(t)/q(t)$  than is available by modeling. To develop this ratio, all in vivo and urinalysis results with identical dates and worker ID's are withdrawn from the data base and ratios obtained. The mean ratio (not the ratio of the means) where possible may then be used for the entire population by multiplying all urinalysis results by the above ratio (in vivo result/urinalysis result). Care should be taken, however, to ensure that the sample of paired measurements is representative. For example, a common problem is that one measurement scheme often is used as a check on high values determined from the other. At the Y-12 facility, accidents were accompanied by such paired measurements, so that the ratio  $E(t)/q(t)$  thus obtained would be weighted towards values representative of times immediately after significant intakes.

#### Calculating Cumulated Organ Activity from Calculated Organ Burdens

The preceding sections have detailed the calculation of the activity  $q_s(j, t_i)$  of radionuclide  $j$  in source organ  $S$  at an instant in time  $t_i$ , starting with either bioassay measurements or in vivo counts. This section outlines the considerations involved in the calculation of a cumulated activity,  $U(j, S, t_2 - t_1)$ . The cumulated activity is the time integral of the organ burden:

$$U(j,S,t_2 - t_1) = \int_{t_1}^{t_2} q_s(j,t)dt . \quad (6-45)$$

Conceptually  $U(j,S,t_2-t_1)$  is the number of nuclear transformations (n.t.) of radionuclide  $j$  in organ  $S$  between time  $t_1$  and time  $t_2$ . In general, one is interested in computing  $U(j,S,t_{i+1}-t_i)$ , which represents the cumulated activity during the interval separating the  $i^{\text{th}}$  and  $i + 1$  organ burden measurements in a temporally ordered sequence of measurements on a given individual. Strictly speaking,  $U(j,S,t_2-t_1)$  is dimensionless.

The cumulated activity  $U(j,S,t_2-t_1)$  is calculated from quantities having the dimensions of activity ( $q(t)$ ) and of time ( $t$ ), which are found in a considerable variety of units in occupational records. In order of increasing magnitude,  $U(j,S,t_2-t_1)$  can be found expressed in Bq-s (= 1 n.t.; note that Bq-s is the SI unit (ICRU 1980)), dpm-days (= 1440 n.t.), Bq-days (= 86,400 n.t.),  $\mu\text{Ci}$ -hours (= 133,200,000 n.t.), or, more traditionally,  $\mu\text{Ci}$ -days (= 3,196,800,000 n.t.).

Two general categories of problems involving the calculation of cumulated activity (also known as integral organ burden) may be encountered. The first concerns the use of bioassay data to compute organ doses specific to an individual, either in the form of annual doses or doses delivered during some longer interval in a study. The second category concerns the use of bioassay data to determine the mean organ dose (or mean annual dose) delivered to a group of workers with some common characteristic such as job title. This latter category is explored here first since it often is easiest to deal with in a large epidemiologic study. It is recognized, however, that studies often will require the development of individual-specific doses, particularly when there is reason to suspect large variability in doses between workers in specific job titles or departments.

#### Calculating Mean Cumulated Activity for Groups

Consider a population of  $N$  workers grouped together by some common classification such as job title. It is assumed that the classification

arose from a prior determination that the N individuals worked in similar environments with respect to the contaminant under study. The bioassay data for these workers are pooled into discrete sets according to the time of measurement. By way of example, the measurements might be pooled by year, with a resulting pool for each year of monitoring. In this case, computation of annual doses might be the goal.

Consider now the pool of bioassay data for the N workers for a given year. It is assumed that these data have already been converted to organ burden measurements, as described in the previous section. It is necessary to make use of these individual organ burden data in calculating the mean organ burden expected to hold throughout the year for the study group of workers. If it can be assumed that the bioassay samples were drawn completely at random from the population, then the mean of the organ burdens above is the best estimate of the mean organ burden for the population during the year. By "completely at random" we mean:

1. each individual in the group was equally likely to be sampled during the year, and
2. each bioassay measurement was made at a time uncorrelated with known large intakes of the contaminant and the measurements are distributed uniformly throughout the year.

The first criterion ensures that the individuals being monitored are representative of the entire population (job title, etc.). This criterion would be violated for example, if some of the individuals were chosen for sampling because of prior knowledge by the facility health physics staff that they were at increased risk of exposure. If this were true, the mean bioassay result for the sample population would be an overestimate of the mean bioassay result that would have been obtained if criterion 1 had held true.

Criterion 2 is important due to the common health physics practice of starting bioassay procedures on an individual following a known intake. Such bioassay measurements are not representative samples of the organ burdens holding in the study population throughout the year. These measurements lead, instead, to an overestimate of the mean organ burden for the study population.

If neither criterion has been violated, then all bioassay results (organ burdens) for the year of interest are pooled for the sample population. Let these individual organ burden measurements constitute a set  $X$  of size  $n$ , consisting of the individual measurements  $(x_1, x_2, x_3, \dots, x_n)$ . The best estimate of the mean organ burden for the entire population then is simply

$$\bar{X} = \frac{\sum_{i=1}^n x_i}{n} \quad (6-46)$$

Assuming that  $x_i$  is in units of  $\mu\text{Ci}$ , the annual cumulated activity for each worker in the population is

$$U(j, S, \text{annual}) = 365\bar{X} = \frac{365}{n} \sum_{i=1}^n x_i \quad (6-47)$$

where  $U(j, S, \text{annual})$  is the cumulated activity in units of  $\mu\text{Ci-days}$  for the year of interest as resulting from irradiation of organ  $S$  by radionuclide  $j$ . This cumulated activity is specific to a radionuclide and an organ, and no attempt should be made to add cumulative activity estimates from differing radionuclides and organs. This restriction arises from the fact that  $S$ -factors (discussed in the next section of this report) may differ widely between radionuclides and organs.

The above discussion assumes that there is reasonable confidence that no bioassay measurements were initiated as a result of known intakes. In addition, it assumes that the sample population (monitored individuals) had organ burdens representative of the entire population. Assume for the moment that the latter assumption is true, but that the pool of bioassay data contains measurements known to have arisen from an accident or other large intake. For example, measurements at the Y-12 facility were, at times, accompanied by a code letter or number indicating that they had been performed because a worker or supervisor had determined that a large intake had occurred.

Again, let  $X$  designate that set of bioassay measurements of size  $n$  for the sample population. The number of workers is equal to  $m$ . In addition, allow these bioassay results to be pooled by individual worker for each year

of monitoring and in order of increasing date of measurement. The set X now is broken into subsets

$$\begin{aligned}X_1 &= (x_1, x_2, x_3) \\X_2 &= (x_4, x_5, x_6) \\X_m &= (x_{n-2}, x_{n-1}, x_n) .\end{aligned}$$

This illustrative example assumes that there were three measurements for each individual during the year, which in turn have been ordered chronologically for the year. The number of measurements can vary widely in real situations.

It is now possible to further divide each of the subsets ( $X_1$ ,  $X_2$ , etc.) into two discrete subsets. The first of these would consist of all measurements not associated with a suspected "accidental" intake of large magnitude. The second consists of all measurements associated with such an intake. These must be identified either on the basis of an accident report or by a code that might accompany the bioassay record itself (such as occurred at the Y-12 facility). This leaves two groups of bioassay measurements as described above, those associated with a known intake and those associated with times representative of the more chronic exposures to the average worker in the job title under study.

What is meant by the term "associated" as employed above? Clearly, any bioassay result accompanied by a code or flag indicating an accident falls into the first category. In the absence of such a code, it should be determined whether large measurement results were systematically followed by a series of subsequent measurements spaced closely in time. These results should also be considered as accompanying an intake if they were obtained as a follow-up to the intake and not as part of the routine sampling schedule. By "closely following," we mean simply results that were deliberately obtained during some interval of time following the intake, where this interval is equal to the length of time necessary to return the organ burden to the "chronic" values typified by the measurements not associated with known (large) intakes. Technically, this interval is infinite in extent, but some finite value may be chosen as representative of the time necessary to allow the intake to decrease by some fraction. For example, the interval

might be chosen to allow the organ burden to fall to 25 percent of the peak value. Assuming a single exponential model for removal from the organ, the associated interval of time would be equal to two half-lives ( $2 \times T_e$ ). As a result, all bioassay results following a known intake by two half-lives would be assumed to accompany that intake unless the records indicate that the worker had been placed back onto a routine sampling schedule prior to this time. In general, if  $R(t)$  is the fractional retention function for the organ, then the interval is chosen such that  $R(t)$  drops to the prescribed fraction. For the example above, the time interval,  $t$ , is chosen to satisfy the relation

$$R(t) = 0.25 . \quad (6-48)$$

Remember that placement of any measurement result into the category assumed associated with an intake requires reasonable confidence that this measurement would not have been performed in the absence of a known intake.

This separation of bioassay results (into those associated and those not associated with known intakes) then is accomplished for each monitored individual during the year of interest. Results not associated with a known intake then are re-pooled for the entire monitored population and the mean ( $\mu Ci$ ) computed as in equation 6-46, with the exception that the value of  $n$  is now smaller by the number of bioassay results placed into the category "associated with a known intake." Letting  $\bar{X}_{na}$  be the mean ( $\mu Ci$ ) organ burden for the entire population during times of normal exposure, and  $X_{i,na}$  be the  $i^{th}$  bioassay result obtained during such times, it may be seen that

$$\bar{X}_{na} = \frac{\sum_{i=1}^{N_{na}} X_{i,na}}{N_{na}} , \quad (6-49)$$

where  $N_{na}$  is the number of bioassay results obtained during times of normal exposure. The mean cumulated activity during such normal periods within a year then is

$$\bar{U}(j,S,t_{na}) = t_{na} \bar{X}_{na} = \left( \frac{t_{na}}{N_{na}} \sum_{i=1}^{N_{na}} X_{i,na} \right) . \quad (6-50)$$

Here,  $t_{na}$  is the mean number of days per year represented by normal exposure conditions for the population of interest ("normal" simply means not associated with a known intake).

The question then arises as to how  $t_{na}$  can be estimated. Consider first the case where a known intake results in the immediate onset of bioassay procedures. Let  $T_t$  be the total number of person-days (during the year of interest) of employment in the job title or other category of interest. In other words, if three persons each worked  $t_1$  days in the job title during the year, and two persons each worked  $t_2$  days, then  $T_t = 3t_1 + 2t_2$ . Further, let  $T_a$  be the total number of person-days (during the year of interest) in this population for which bioassay results were associated with known intakes. In general,

$$T_a = \sum_{i=1}^Z t_{i,a} \quad (6-51)$$

where  $Z$  is the number of distinct measurement intervals in the population associated with known intakes (which will, in turn, be equal to the number of known intakes in the population). For example, if  $t_{i,a}$  in equation 6-51 is 36.5 days or 0.1 year, and if 10 workers had records associated with known intakes, then

$$T_a = \sum_{i=1}^{10} (0.1) = 365 \text{ person-days.}$$

This assumes that all of the intervals  $t_{i,a}$  are contained within the year of interest. In general, only the fraction of  $t$  falling within the year of interest should be employed for calculation of  $T_a$ . Returning to equation 6-50, it would then hold that

$$\bar{U}(j, S, t_{na}) = \frac{365(T_t - T_a)}{N_{na} T_t} \left( \sum_{i=1}^{N_{na}} x_{i,na} \right) \quad (6-52)$$

If an individual worked less than a year in an environment (job title), then 365 is replaced by the number of days during which the individual was employed in the environment (job title).

It now remains to calculate the mean cumulated activity assumed to hold during periods of time associated with known intakes,  $\bar{U}(j,S,t_a)$ . This may be performed in several manners. If known intakes were treated identically in all individuals, with subsequent sampling (during  $t$ ) that occurred at fixed (and identical) intervals following the intake, then all bioassay values associated with known intakes can be averaged. In other words,

$$\bar{X}_a = \frac{\sum_{i=1}^{N_a} x_{i,a}}{N_a} , \quad (6-53)$$

where the index  $a$  refers to measurements associated with a known intake. In this case,

$$\bar{U}(j,S,t_a) = \frac{356T_a}{N_a T_t} \sum_{i=1}^{N_a} x_{i,a} . \quad (6-54)$$

The best estimate of the mean cumulated activity for the entire population during the year then is

$$\bar{U}(j,S,\text{annual}) = \bar{U}(j,S,t_{na}) + \bar{U}(j,S,t_a) . \quad (6-55)$$

A more realistic case is one in which known intakes did not result in a standard pattern of bioassay measurements, since larger known intakes might require closer scrutiny by the health physics staff, resulting in more measurements on highly exposed individuals. The calculation of  $t_{na}$  and  $\bar{U}(j,S,t_{na})$  still proceeds as before, but  $\bar{U}(j,S,t_a)$  now is computed for each individual,  $i$ , prior to computing the mean cumulated activity  $\bar{U}(j,S,t_a)$ .

To compute  $\bar{U}_i(j,S,t_a)$ , the cumulated activity for the  $i^{\text{th}}$  worker during the year of interest and during times associated with known intakes, the bioassay data must be arranged into a linear array characterized by increasing time of measurement. Let  $(q_{i,j}, t_{i,j})$  represent the  $j^{\text{th}}$  bioassay measurement of organ burden ( $q_{i,j}$ ) obtained on the  $i^{\text{th}}$  worker at time  $t_j$  during the interval  $t$  following a known intake. A given worker is assumed to have  $m$  such measurements during time  $t$ , yielding a discrete set of data pairs:  $\{(q_{i,1}, t_{i,1}), (q_{i,2}, t_{i,2}), \dots, (q_{i,m}, t_{i,m})\}$ . Assuming that linear interpolation between data points is a valid approximation, it follows that

$$\begin{aligned}
U_i(j, S, t_a) &= \frac{(q_{i,1} + q_{i,2})}{2} (t_{i,2} - t_{i,1}) \\
&+ \frac{(q_{i,2} + q_{i,3})}{2} (t_{i,3} - t_{i,2}) \\
&+ \dots + \frac{(q_{i,m-1} + q_{i,m})}{2} (t_{i,m} - t_{i,m-1}) \\
&= \sum_{j=1}^{m-1} \frac{(q_{i,j} + q_{i,j+1})}{2} (t_{i,j+1} - t_{i,j}) . \quad (6-56)
\end{aligned}$$

Remember that the integration of the  $q$  values continues until the end of the year or until the end of the interval  $t$ , whichever occurs first. If the end of the year occurs first, the remainder of  $U_i(j, S, t_a)$  must be included in the next year. This carry-over holds also for the computation of  $t_{na}$ .

This calculation is performed for each interval  $t$  within the year of interest and for all individuals in the population. The mean cumulated activity associated with intervals following known intakes then is

$$\bar{U}(j, S, t_a) = \frac{\sum_{k=1}^{N_a} U_k(j, S, t_a)}{N_a} \quad (6-57)$$

where  $N_a$  is the number of intervals of length  $t$  associated with known intakes in the population of workers. By analogy with equation 6-55, it may be seen that

$$\begin{aligned}
\bar{U}(j, S, \text{annual}) &= \bar{U}(j, S, t_{na}) + \bar{U}(j, S, t_a) \\
&= \frac{365(T_t - T_a)}{N_{na} T_t} \left[ \sum_{i=1}^{N_{na}} X_{i, na} \right] \\
&+ \sum_{k=1}^{N_a} U_k(j, S, t_a) / N_a . \quad (6-58)
\end{aligned}$$

Equation 6-58 gives the mean annual cumulated activity for workers employed in the exposure environment throughout the year and includes only the cumulated activity delivered during employment within the category of interest such as job title or department.

#### Extrapolating Backward in Time

The preceding discussion assumed that the first measurement associated with a known intake occurred immediately after the intake. This may not prove to be true in some instances. If this assumption is not true, but no record was kept of the actual time of intake, and if  $R(t)$  is characterized by a single exponential retention function, then nothing can be done to correct for this problem. The investigation is simply left with a source of bias in the data, resulting in an uncertainty in the mean cumulated activity in the population. In cases of more complicated retention functions, the methodology outlined by Skrable (1981) may be employed to determine the time of intake from a time series of bioassay measurements. Fortunately, however, large intakes typically result in the generation of an accident report containing information as to the time of intake.

Let this time of intake be given as  $t_0$ , with the first subsequent bioassay measurement being performed at time  $t_1$  and resulting in an organ burden  $q_1$ . The retention function (including both biological and radiological removal) for the radionuclide in the organ is assumed to be given by  $R_e(t)$ . In this case, the best estimate of the organ burden at time  $t_0$  would be given by

$$q_0 = q_1 / R_e(t_1 - t_0) \quad (6-59)$$

where  $(t_1 - t_0)$  is the argument of the function  $R$ . The cumulated activity during the interval  $(t_0, t_1)$  then is

$$\begin{aligned} U_i(j, S, t_1 - t_0) &= \frac{(q_0 + q_1)}{2} (t_1 - t_0) \\ &= \frac{q_1 [1 + 1/R_e(t_1 - t_0)]}{2} (t_1 - t_0) \end{aligned} \quad (6-60)$$

This cumulated activity then is added to the value computed in equation 6-57 and the analysis proceeds as indicated. The time interval,  $(t_0, t_1)$ , is added to the estimate of  $T_a$ . A more precise determination of  $U_i(j, S, t_1 - t_0)$  may be obtained by fitting the retention function to all of the data in the interval and determining the integral

$$U_i(j, S, t_1 - t_0) = q_0 \int_{t_0}^{t_1} R_e(t) dt . \quad (6-61)$$

Care must be taken to ensure, however, that no further intakes occurred in the interval, which may not prove to be the case. This concludes discussion of the calculations necessary in computing mean cumulated activity for a group of workers assumed to be exposed to similar conditions.

#### Calculating Cumulated Activity for Individuals

The above calculations presume that each worker was equally likely to be sampled during normal periods of exposure and equally likely to experience a given level of accidental intake. Such conditions are likely to be the exception rather than the rule, since job titles or department codes usually are only rough indicators of exposure potential. In addition, the selection of individuals and frequency for monitoring may be unconsciously influenced by impressions of the health physics staff. In other words, the health physicist may have been led to select for monitoring those individuals within a job title who are expected to have the greatest potential for exposure, or to increase the sampling schedule in those workers.

Because of these considerations, it may prove necessary to compute total annual cumulated activity for each individual monitored or to not attempt the computation of doses. This former necessity could also arise from a desire to employ regression analysis on individual annual doses when calculating risk factors and dose-response curves. Alternatives to dose estimation (such as semiquantitative hazards grouping) are deferred to a later point in this discussion.

It may be necessary to compute cumulated activity for each monitored worker separately. This calculation proceeds much as before, with measurements on an individual during the year being placed in order of increasing time of measurement, although there is no separation of "accident" and "non-accident" results. This results in a discrete set of data points on the  $i^{\text{th}}$  worker for the year of interest,

$$\{(q_{i,1}, t_{i,1}), (q_{i,2}, t_{i,2}), \dots, (q_{i,m}, t_{i,m})\} .$$

If it is assumed that exposure continued throughout the time between  $t_{i,1}$  and  $t_{i,m}$ , then linear interpolation between the individual organ burden measurements in each year may be employed,

$$U_i(j, S, \text{annual}) = \sum_{j=1}^{m-1} \frac{(q_{i,j} + q_{i,j+1})}{2} (t_{i,j+1} - t_{i,j}) . \quad (6-62)$$

Note that if  $t_{i,j+1} - t_{i,j}$  is constant for all  $j$  (i.e. systematic and constantly spaced sampling throughout the year), then  $U_i(j, S, \text{annual})$  above may be replaced by the product of the mean organ burden and length of the measurement interval

$$U_i(j, S, t_{i,m} - t_{i,1}) = \left[ \frac{\sum_{j=1}^m q_{i,j}}{m} \right] (t_{i,m} - t_{i,1}) . \quad (6-63)$$

Unfortunately, such systematic sampling is rarely, if ever, the case in reality. One should remember that equation 6-63 gives the cumulated activity only during the time interval in which measurements occur.

A few words are in order concerning the effect of boundaries between consecutive years. The above equations assume that all  $m$  data points fall within the year of interest. Of more general concern is the instance where data also are available during the years immediately prior to and following the year of interest. Let  $q_{i,0}$  be the last organ burden measurement in the

year immediately prior to the year of interest, with associated time  $t_{i,0}$ . Also, let  $q_{i,m+1}$  be the first organ burden measurement in the year immediately following the year of interest, with associated time  $t_{i,m+1}$ . The cumulated activity between times  $t_{i,0}$  and  $t_{i,1}$  then is

$$\begin{aligned}
 & U(j, S, t_{i,1} - t_{i,0}) \\
 &= \frac{(q_{i,0} + q_{i,1})}{2} (t_{i,1} - t_{i,0}) . \qquad (6-64)
 \end{aligned}$$

Clearly, the cumulated activity assigned to the year of interest (from activity at the beginning of the year) is

$$\begin{aligned}
 & U_i(j, S, t_{i,1} - t_{\text{boy}}) \\
 &= \frac{(q_{i,0} + q_{i,1})}{2} (t_{i,1} - t_{i,0}) \frac{(t_{i,1} - t_{\text{boy}})}{(t_{i,1} - t_{i,0})} , \qquad (6-65)
 \end{aligned}$$

where  $t_{\text{boy}}$  is the time associated with the beginning of the year of interest. The cumulated activity between times  $t_{i,m}$  and  $t_{i,m+1}$  then is

$$\begin{aligned}
 & U_i(j, S, t_{i,m+1} - t_{i,m}) \\
 &= \frac{(q_{i,m} + q_{i,m+1})}{2} (t_{i,m+1} - t_{i,m}) . \qquad (6-66)
 \end{aligned}$$

The cumulated activity assigned to the year of interest (from activity at the end of the year) is

$$\begin{aligned}
 & U_i(j, S, t_{\text{eoy}} - t_{i,m}) \\
 &= \frac{(q_{i,m} + q_{i,m+1})}{2} (t_{i,m+1} - t_{i,m}) \frac{(t_{\text{eoy}} - t_{i,m})}{(t_{i,m+1} - t_{i,m})} , \qquad (6-67)
 \end{aligned}$$

where  $t_{\text{eoy}}$  is the time associated with the end of the year of interest.

The preceding paragraphs assume that organ burden may be approximated by a smoothly varying function between organ burden measurements. The measurements for an individual may then be integrated using the linear interpolation described previously.

The question arises as to how to deal with a year in which no results are available for an individual. One possibility is to assign a cumulated activity of zero, simply assuming the worker was unexposed. An alternative approach is possible if it is ascertained that other workers (with the same job title), monitored at some rate equal to or less than yearly, constitute a population with similarly low exposures. If the dose assessment process reveals this to be the case, then organ burden measurements ( $q_j$ ) for all individuals sampled at a low frequency within a job title and during the year of interest are pooled contingent upon the measurements not being associated with a known intake. If  $b$  such organ burden measurements are available for the population during the year, then the mean organ burden,  $\bar{q}_{\min}$ , holding during times of minimal exposure is

$$\bar{q}_{\min} = \left( \sum_{j=1}^b q_j \right) / b . \quad (6-68)$$

The values  $q_j$  are assumed to be representative of the organ burdens holding in all individuals (within job title, etc.) exposed to minimal levels of the contaminant during the year of interest. This has assumed that workers (in a job title) who have very low sampling schedules receive exposures that are representative of unmonitored workers in that job title.

#### A Simple Example for Calculating Cumulated Activity

A hypothetical set of data for an individual should best illustrate the preceding points, as well as a few remaining topics of discussion. Consider an individual with the following bioassay measurements, all given in units of  $\mu\text{Ci}$  :

Measurement Number	Measurement Result ( $\mu\text{Ci}$ )	Time (days) since entry into the workforce
1	1	10
2	3	150
3	2	410
4	30*	560
5	25	570
6	15	600
7	6	700
8	2	750
9	4	2000

\*Indicates a known or suspected acute intake.

For simplicity, let the worker enter the workforce on the first day of year 1. The best estimate of the cumulated activity between this entry date and the day of the first measurement is

$$\frac{(1 + 0)}{2} (10 - 0) = 5 \mu\text{Ci-days.}$$

The best estimate between the second and third measurements is

$$\frac{(3 + 1)}{2} (150 - 10) = 280 \mu\text{Ci-days.}$$

The best estimate between the second and third measurements is

$$\frac{(3 + 2)}{2} (410 - 150) = 650 \mu\text{Ci-days,}$$

of which

$$\frac{650 (365 - 150)}{(410 - 150)} = 538 \mu\text{Ci-days}$$

is allocated to the first year. This indicates a cumulated activity of  $5 + 280 + 538 = 823 \mu\text{Ci-days}$  during the first year.

The cumulated activity allocated to the second year from the value of 650 above is

$$\frac{650 (410 - 365)}{(410 - 150)} = 112 \mu\text{Ci-days}.$$

If the fourth measurement were not flagged as being indicative of an intake, then the cumulated activity between the third and fourth measurements would be equal to

$$\frac{(30 + 2)}{2} (560 - 410) = 2400 \mu\text{Ci-days}.$$

Since the fourth measurement is assumed to have resulted from a very recent intake, the organ burden from the third measurement (2  $\mu\text{Ci}$ ) is assumed to hold until  $t = 560$ , at which time it rises to the value of 30. Therefore, the cumulated activity between the third and fourth measurements is taken as

$$2(560 - 410) = 300 \mu\text{Ci-days}.$$

The cumulated activity between the fourth and seventh measurements is

$$\begin{aligned} & \frac{(30 + 25)}{2} (570 - 560) + \frac{(25 + 15)}{2} (600 - 570) + \frac{(15 + 6)}{2} (700 - 600) \\ & = 275 + 600 + 1050 = 1925 \mu\text{Ci-days}, \end{aligned}$$

all of which is allocated to the second year. The cumulated activity between the seventh and eighth measurement is

$$\frac{(6 + 2)}{2} (750 - 700) = 200 \mu\text{Ci-days},$$

of which

$$\frac{200[(365 \cdot 2) - 700]}{(750 - 700)} = 120 \mu\text{Ci-days}$$

is allocated to the second year and 80  $\mu\text{Ci-days}$  to the third year. The total for the second year then is

$$112 + 300 + 1925 + 120 = 2457 \mu\text{Ci-days}.$$

The cumulated activity between the eighth and ninth measurements is

$$\frac{(2 + 4)}{2} (2000 - 750) = 3750 \mu\text{Ci-days.}$$

Of this amount, the third year is allocated

$$\frac{3750[(365 \cdot 3) - 750]}{(2000 - 750)} = 1035 \mu\text{Ci-days.}$$

The total in the third year then is

$$80 + 1035 = 1115 \mu\text{Ci-days.}$$

The cumulated activity during the fourth year (prior to the ninth measurement) is

$$\frac{3750[(365 \cdot 4) - (365 \cdot 3)]}{(2000 - 750)} = 1095 \mu\text{Ci-days.}$$

Similarly, the cumulated activity during the 5<sup>th</sup> year is

$$\frac{3750[(365 \cdot 5) - (365 \cdot 4)]}{(2000 - 750)} = 1095 \mu\text{Ci-days.}$$

The cumulated activity during the sixth year prior to  $t = 2000$  is

$$\frac{3750[2000 - (365 \cdot 5)]}{(2000 - 750)} = 525 \mu\text{Ci-days.}$$

Suppose now that the individual has no further records after measurement 9, but remains in the same job title until  $t = 5000$ . He then switches to another job title until  $t = 6000$ , at which time he retires. What cumulated activity should be assigned during these subsequent years?

The fact that no results were obtained between  $t = 2000$  and  $t = 5000$  days may be interpreted in two broad manners. The first assumes that the individual remained at the same level of exposure as previously but simply was not sampled. If this may be shown to be the case, then the best

estimate of the organ burden for this individual during this interval is simply the value last encountered, such as 4  $\mu\text{Ci}$ , contingent upon this last value not being associated with a known intake.

The cumulated activity during the remainder of the sixth year then is

$$4(365 \cdot 6) - 2000 = 760 \mu\text{Ci-days},$$

for a sixth year total of

$$525 + 760 = 1285 \mu\text{Ci-days}.$$

The cumulated activity during the seventh year is

$$4(365 \cdot 7) - (365 \cdot 6) = 1460 \mu\text{Ci-days},$$

which also is true of the eighth through thirteenth years.

Consider now the fourteenth year. The cumulated activity during this year prior to the change in job titles (assumed to indicate a change in exposure status) is

$$4(5000) - (365 \cdot 13) = 1020 \mu\text{Ci-days}.$$

After  $t = 5000$ , the individual presumably worked in a new exposure environment, but no individual-specific bioassay measurements are available. In this case, the bioassay results from other workers in the same (new) job title are used to obtain an estimate of the mean organ burden for this job title. As described earlier, only measurements not associated with an intake are employed for this purpose, assuming that large intakes in the individual of interest would not have gone unmonitored. In addition, only bioassay results obtained from workers with low sampling frequency (one record in the year of interest) should be employed, unless it can be assured that sampling frequency in the job title was not determined by the level of exposure.

Assume that such bioassay results have been obtained from other workers with the same job title for each year between  $t = 5000$  and  $t = 6000$ . The

mean for each year is found to be 1  $\mu\text{Ci}$ . During the remainder of the fourteenth year, therefore, the worker is assumed to receive

$$1(365 \cdot 14) - 5000 = 110 \mu\text{Ci-days},$$

due to the chronic exposure at a level of 1  $\mu\text{Ci}$ . However, the previous organ burden of 4  $\mu\text{Ci}$  does not cease immediately at  $t = 5000$  days, continuing instead until it has disappeared. More precisely, it drops to 1  $\mu\text{Ci}$  within some finite interval of time.

What cumulated activity is contributed to times following  $t = 5000$  from the 4  $\mu\text{Ci}$  burden at  $t = 5000$ ? This 4  $\mu\text{Ci}$  burden drops to a value of 1  $\mu\text{Ci}$  as specified above. The total cumulated activity (to infinite time) from the 3  $\mu\text{Ci}$  difference in initial and final organ burdens is

$$U = \int_0^{\infty} 3R(t)dt . \quad (6-69)$$

If  $R(t)$  is a single exponential, then

$$U = 3/\lambda = (1.44 T_e)3.$$

To what year is this cumulated activity allocated? If  $R(t)$  decreases rapidly, so that most of the original activity is gone within a year, then it is reasonable to allocate all of the quantity  $U$  to the year containing the change in exposure status. For example assume that  $R(t)$  is a single exponential with  $T_e = 10.0$  days. Then for the worker in the hypothetical example,

$$U = (1.44)(10)(3) = 43.2 \mu\text{Ci-days}.$$

This is added to the cumulated activities of 1020 and 110  $\mu\text{Ci-days}$  computed earlier to yield a total of

$$1020 + 110 + 43.2 = 1173.2 \mu\text{Ci-days}$$

during the fourteenth year. The cumulated activity during the fifteenth year then is

$$1(365 \cdot 15) - (365 \cdot 14) = 365 \mu\text{Ci-days},$$

which also is true for the sixteenth year.

If  $R(t)$  did not drop rapidly with time, then the fourteenth year would be allocated only

$$\int_0^{(365 \cdot 14) - 5000} 3R(t)dt$$

for a total cumulated activity of

$$1020 + 110 + \int_0^{(365 \cdot 14) - 5000} 3R(t)dt$$

The fifteenth year would receive (from the decaying  $4 \mu\text{Ci}$  burden alone)

$$\int_{(365 \cdot 14) - 5000}^{(365 \cdot 15) - 5000} 3R(t)dt$$

and so on throughout the remaining years.

Finally, one arrives at the time  $t = 6000$  days, when the worker terminates employment and (presumably) is removed from conditions of exposure. The organ burden at  $t = 6000$  was assumed to be  $1 \mu\text{Ci}$ , which then decays by the retention function  $R(t)$  with no further intakes. Again, the total cumulated activity present over all time following  $t = 6000$  is simply

$$\int_0^{\infty} 1R(t)dt$$

which is allocated to the appropriate years according to the considerations above.

On a final note, if measurement 9 had been associated with a known intake, then continuing the organ burden of 4  $\mu\text{Ci}$  until  $t = 5000$  days would have been inappropriate. In this case, the best estimate of the organ burden would be to employ the mean value for the job title as indicated by workers with infrequent sampling. The organ burden of 4  $\mu\text{Ci}$  at  $t = 2000$  would be assumed to continue to drop with the characteristic function  $R(t)$ . The cumulated activity from this 4  $\mu\text{Ci}$  burden during ensuing years yields a total of

$$\int_0^{\infty} 4R(t)dt .$$

This cumulated activity then is allocated to the appropriate years and added to the cumulated activity computed from use of the mean organ burden for the job title. This problem should arise only rarely, since individuals with known large intakes typically are measured until the results drop to normal values. As a result, the last measurement for a worker should be indicative of his or her organ burden during times of normal exposure. In addition, at many facilities measurements are made when job titles or exposure situations change for an individual.

#### Calculating Doses to Target Organs from Cumulated Activity in Source Organs

Once the value of the annual cumulated activity of radionuclide  $j$  in source organ  $S$  during year  $y$ ,  $U(j,S,y)$ , has been calculated using the methods outlined above, dose or dose equivalent can be inferred to any target organ  $T$  using "S-factors" obtained from the work of Snyder and co-workers (Snyder et al. 1975) or Dunning and co-workers (Dunning, Pleasant, and Killough 1977). The work of Snyder et al. emphasizes radionuclides of interest in medicine, while that of Dunning et al. emphasizes radionuclides of interest in the nuclear fuel cycle. The units of S-factors are either absorbed dose (Snyder et al. 1975), or dose equivalent (Dunning, Pleasant,

and Killough 1977), to a target organ T per unit cumulated activity of radionuclide j in source organ S. For high LET radiations, the source and target organs are identical due to the short range of the radiation in tissue.

The annual dose (denoting either absorbed dose or dose equivalent) D to target organ T due to radionuclide j in source organ S is

$$D(T,j,S,y) = S\text{-factor}(T,j,S) U(j,S,y) . \quad (6-70)$$

The total dose to target organ T from all radionuclides j and from all source organs S is

$$D(T,y) = \sum_j \sum_S D(T,j,S,y) . \quad (6-71)$$

Machine-readable records containing values of D(T,y), the worker's ID, and other flags or notes constitute the inferred internal exposure values to be used in epidemiologic analyses.

#### Method for Inferring Internal Doses from Air Samples and Time-In-Area Data

At times, it may be deemed desirable to calculate internal organ doses based on air monitoring data. This situation could arise, for instance, if the bioassay data were of questionable value due either to: (1) bioassay data not having been collected from personnel suspected of having significant exposures and for whom job titles or work assignments were not randomly sampled; (2) bioassay data having been obtained by a method inappropriate or insensitive to the detection of a significant radionuclide; or (3) bioassay data having been reported consistently as simply falling within semiquantitative categories (such as "less than the minimum detectable amount" or "less than one body burden"). In any of these cases, the dose assessor may find that the air sampling results are more complete and choose to calculate organ doses based upon air monitoring data. In the following discussion it is assumed that air monitoring data are available; that they may be assigned to a specific job title, department, or other

worker classification; and that they have been deemed representative of airborne exposures for that classification.

Factors relating dose (absorbed dose or dose equivalent) commitment to intake of radionuclides via inhalation have been published by Dunning and co-workers (Dunning et al. 1979). These factors, denoted here by  $D^*(j,T,c,AMAD,50\text{-year})$ , are "fifty-year dose [equivalent] commitment" to target organ T "per microcurie intake" of radionuclide j inhaled as an aerosol of particles of ICRP solubility class c and Activity Median Aerodynamic Diameter AMAD. A description of these factors is included in the "Highlights" section (Dunning et al. 1979), reproduced here in its entirety:

This report is the second of a two-volume tabulation of internal radiation dose conversion factors for man for radionuclides of interest in environmental assessments of light-water-reactor fuel cycles. This volume treats 78 radionuclides, all of mass number greater than 200. Intake by inhalation and ingestion are considered. The International Commission of Radiological Protection (ICRP) Task Group Lung Model has been used to simulate the behavior of particulate matter in the respiratory tract. Results corresponding to activity median aerodynamic diameters (AMAD) of 0.3, 1.0, and 5.0 micrometers are given. The gastrointestinal (GI) tract has been represented by a four-segment catenary model with exponential transfer of radioactivity from one segment to the next. Retention of radionuclides in other organs is characterized by linear combinations of decaying exponential functions. Fifty-year dose commitment per microcurie intake of each parent nuclide is given for 22 target organs with contributions from specified source organs plus surplus activity in the rest of the body; cross irradiation due to penetrating radiations has also been incorporated into these tabulations. Dose conversion factors are also presented in four sets of summary tables for easy reference. In addition to the foregoing tabulations, in which the value  $Q(\alpha) = 10$  has been assumed for quality factor corresponding to alpha emissions, an alternative table of dose conversion factors for alpha-emitting nuclides with

Q(alpha) = 20 is provided. Specific computational details in which the present calculations depart from the general methodologies presented in the previous volume of this series are discussed.

The D\* factors, referred to as "dose per unit intake" factors, can be used with air sample data and information about how long a worker breathed air at the concentration measured by the air sampler to infer doses to various organs for use in epidemiology. This method has been described by Beck and co-workers (Beck et al. 1983), and is referred to here as the "Air Sample/Time-In-Area" method.

Beck's method involves first calculating an intake for each worker for each period of time that the worker had a given job title. By relating job titles to work areas (building numbers) and processes, individual workers are linked to specific air samples taken during the time they worked. The selection of solubility class(es) to be used in calculations is based on knowledge of processes carried on in a building or area, along with a knowledge of the chemical form(s) of the radionuclide(s) used in those processes. Often more than one chemical form of a radionuclide will be encountered by a particular worker. Tables relating ICRP solubility class to chemical forms of radionuclides are available from several sources (Dunning et al. 1979; ICRP 1979; Alexander 1974). Personnel records with job titles and dates of employment permit calculation of days of exposure to air at the concentration indicated by the appropriate air samples. Detailed job descriptions permit calculation of the fraction of each work day that a worker was exposed. Intake of radionuclide j between time t<sub>1</sub> and time t<sub>2</sub>, I(j, t<sub>1</sub>, t<sub>2</sub>, c, AMAD) (in units of μCi), is simply the product of the weekdays worked, (5/7)(t<sub>2</sub>-t<sub>1</sub>); the fraction of the time interval exposed, f; the worker's breathing rate, I<sub>b</sub> (in cubic meters/workday); and the average concentration in air of radionuclide j during the time interval between t<sub>1</sub> and t<sub>2</sub>, C(j, t<sub>1</sub>, t<sub>2</sub>, AMAD), in units of μCi/cubic meter:

$$I(j, t_1, t_2, c, AMAD) = \frac{5}{7} (t_2 - t_1) f I_b C(j, t_1, t_2, AMAD) \quad (6-72)$$

Note that the number of hours worked per day is multiplied by the hourly breathing rate to obtain the workday breathing rate,  $I_b$ .

Multiplication of the intake of radionuclide  $j$ ,  $I(j, t_1, t_2, c, AMAD)$ , by the appropriate "dose per unit intake" factor,  $D^*(j, T, c, AMAD, 50\text{-year})$ , gives the "50-year dose commitment" to target organ  $T$  due to the intake:

$$D(j, T, c, AMAD, 50\text{-year}) = D^*(j, T, c, AMAD, 50\text{-year}) \cdot I(j, t_1, t_2, c, AMAD). \quad (6-73)$$

For reasons of epidemiologic validity, Beck et al. found it necessary to truncate the 50-year period implicit in the dose per unit intake factors (Beck et al. 1983). Such truncation requires evaluating the retention functions for the various organs that contribute to irradiating target organ  $T$ . Even with truncation, the results of the Air Sample/Time-In-Area method are still dose commitments, not annual doses, unless the retention half-life is small compared to a year.

The total committed dose equivalent to target organ  $T$ ,  $D(T, 50\text{-Year})$ , is obtained by summing  $D(j, T, c, AMAD, 50\text{-year})$  over all radionuclides  $j$ , solubility classes  $c$ , and Activity Median Aerodynamic Diameters  $AMAD$

$$D(T, 50\text{-year}) = \sum_j \sum_C \sum_{AMAD} D(j, T, c, AMAD, 50\text{-year}) . \quad (6-74)$$

It must be emphasized that the Air Sample/Time-In-Area method produces dose commitments, not annual doses, and thus differs fundamentally from the methods of inferring annual doses from bioassay measurements or in vivo counts. Because the 50-year dose commitment includes all the dose received over the 50 years following the intake, some or even most of the dose may occur after the incidence of radiogenic disease, if the disease occurs less than 50 years after the intake and if the retention halftime is very long. Since it is not plausible for dose occurring after the disease begins to have caused the disease (Hill 1965; Reissland 1982), the disease can be attributed to only that part of the dose commitment occurring prior to the beginning of the latent period preceding the disease diagnosis. The drawback of using dose commitments in epidemiology is particularly serious

for radionuclides such as plutonium or radium that have very long effective half-lives in some compartments such as respiratory, lymph, or bone. The use of 50-year dose commitments in epidemiology thus may violate the antecedent-consequent requirement for the inference of causality (Hill 1965), and for nuclides retained tenaciously by the body and having long effective half-lives, may result in serious underestimation of the risk per unit dose. The underestimation of risk occurs because an observed amount of disease is attributed to an overestimated dose some of which was received after the onset of disease.

It should also be noted that truncation of 50-year dose commitments is not as simple as correcting for the half-life in lung of the various chemical forms of radionuclides. The  $D^*$  factors use refined, multi-compartment models for retention in other organs besides lung, so that the correction for half-life in lung is not correct for other organs (Dunning et al. 1979). For the lung, however, simple corrections can be made to the 50-year dose commitment. The dose,  $D$ , delivered over a time,  $T$ , is simply

$$D = D^*(1 - e^{-\lambda T}) / (1 - e^{-\lambda 50}) ,$$

where  $\lambda$  is the effective lung removal rate constant in units of years<sup>-1</sup>. Clearly, the lung retention function is being characterized by a single exponential.

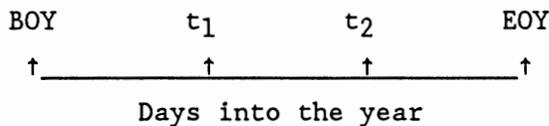
The problems associated with the use of dose commitments can be avoided if the air sampling results are used to calculate annual cumulated activity in the organs followed by use of the appropriate S-factor.

Consider a worker who is exposed to a radioactive atmosphere (or any other contaminated atmosphere) during the  $n^{\text{th}}$  year of exposure. The exposure is assumed to occur during the interval from time  $t_1$  to time  $t_2$  within a given year. For purposes of explication, the target organ is assumed to be the lung and, hence, the retention function reduces to a simple exponential form. Let the following quantities be defined during year  $n$  in the following manner:

1.  $C$  = the concentration of the radionuclide in the inhaled atmosphere (units of  $\mu\text{Ci}/\text{l}$ ).
2.  $f$  = the fraction of the interval  $(t_2, t_1)$  during which the worker was exposed to the atmosphere.

3.  $I_b$  = the intake rate of the contaminated atmosphere (liters per day).  $I_b$  is given only as  $I$  in equations 6-75 to 6-90.
4.  $d$  = the deposition fraction in the organ of interest (here the lung) for the inhaled contaminant. In the example presented here,  $d$  is the fraction of inhaled contaminant deposited in the pulmonary region of the lung.
5.  $S$  = the S-factor for the target organ (here the lung) and for the specific contaminant. Units are rad or rem per microcurie-day of residence in the lung.
6.  $\lambda$  = the removal rate constant ( $\text{days}^{-1}$ ) for the contaminant from the organ. In general, this constant is equal to  $0.693/T_e$  where  $T_e$  is the effective half-life in units of days.

Let the interval  $(t_2, t_1)$  occur at some time within the year. For example,



Here, EOY means "End of Year" and is equal to 365 days. During the interval  $(t_2, t_1)$ , the contaminant is assumed to enter the organ at a constant rate  $R$  ( $\mu\text{Ci}/\text{day}$ ).

$$R_n = f_n I_n C_n d_n , \quad (6-75)$$

where the subscript refers to the  $n^{\text{th}}$  year. The organ burden at any time  $t$  during the interval is then

$$\frac{f_n I_n C_n d_n}{\lambda_n} \left( 1 - e^{-\lambda_n t} \right) \quad (6-76)$$

where  $t$  is the elapsed time since the start of exposure, which begins at  $t_1$ . A simple, single exponential retention function has been employed here. The integral organ burden during the interval is simply the integral of the above expression from 0 to  $(t_2 - t_1)$ ,

$$\begin{aligned}
& \int_0^{t_2 - t_1} \frac{f I C d}{\lambda_n} \left( 1 - e^{-\lambda_n t} \right) dt \\
&= \frac{f I C d}{\lambda_n} (t_2 - t_1) \\
&= \frac{f I C d}{\lambda_n^2} \left( 1 - e^{-\lambda_n (t_2 - t_1)} \right) . \tag{6-77}
\end{aligned}$$

The limits of integration arise from the fact that intake occurs during the interval of length  $t_2 - t_1$ . The dose equivalent delivered during this time interval is the product of equation (6-77) and the S-factor, or

$$\begin{aligned}
& \frac{S f I C d}{\lambda_n} (t_2 - t_1) \\
&= \frac{S f I C d}{\lambda_n^2} \left( 1 - e^{-\lambda_n (t_2 - t_1)} \right) . \tag{6-78}
\end{aligned}$$

Consider now the dose equivalent delivered during the remainder of the year. As of time  $t_2$ , the organ burden is given by

$$\frac{f I C d}{\lambda_n} \left( 1 - e^{-\lambda_n (t_2 - t_1)} \right) . \tag{6-79}$$

Between  $t_2$  and EOY, this organ burden will decrease exponentially. The organ burden at time  $t$ , where  $t$  is in the interval  $(t_2, \text{EOY})$ , is given by

$$\frac{f I C d}{\lambda_n} \left( 1 - e^{-\lambda_n (t_2 - t_1)} \right) e^{-\lambda_n t} . \tag{6-80}$$

The integral organ burden during the interval  $(t_2, \text{EOY})$  is then

$$\begin{aligned}
& \int_0^{(EOY - t_2)} \frac{f_n I_n C_n d_n}{\lambda_n} \left[ 1 - e^{-\lambda_n(t_2 - t_1)} \right] e^{-\lambda_n t} dt \\
&= \frac{f_n I_n C_n d_n}{\lambda_n^2} \left[ 1 - e^{-\lambda_n(t_2 - t_1)} \right] \\
&\cdot \left[ 1 - e^{-\lambda_n(EOY - t_2)} \right]. \tag{6-81}
\end{aligned}$$

The dose equivalent delivered during this interval is

$$\frac{S f_n I_n C_n d_n}{\lambda_n^2} \left[ 1 - e^{-\lambda_n(t_2 - t_1)} \right] \left[ 1 - e^{-\lambda_n(EOY - t_2)} \right]. \tag{6-82}$$

Combining equations (6-78) and (6-82), it may be noted that the dose equivalent delivered to the organ during the year, and as a result of intakes in the interval  $(t_2, t_1)$ , is

$$\begin{aligned}
D_n &= \frac{S f_n I_n C_n d_n}{\lambda_n} (t_2 - t_1) \\
&- \frac{S f_n I_n C_n d_n}{\lambda_n^2} \left[ 1 - e^{-\lambda_n(t_2 - t_1)} \right] \\
&+ \frac{S f_n I_n C_n d_n}{\lambda_n^2} \left[ 1 - e^{-\lambda_n(t_2 - t_1)} \right] \\
&\cdot \left[ 1 - e^{-\lambda_n(EOY - t_2)} \right]. \tag{6-83}
\end{aligned}$$

Remember that all times are assigned by setting the beginning of the year to zero time. If several different intake intervals occur during the year, then separate values of  $D_n$  are computed for each intake interval and the results are summed.

Consider now the case of the dose equivalent delivered during this same year (n) from intakes that occurred in previous years. Assume that a previous intake occurred in year i, where i is less than n. Further assume that the intake occurred in the interval (t<sub>2</sub>, t<sub>1</sub>) within year i, where t<sub>1</sub> and t<sub>2</sub> are given numerical assignments relative to the first day of year i. In other words, if t<sub>1</sub> occurs 100 days into year i, then t<sub>1</sub> is set equal to 100 days.

What will be the organ burden at the end of this intake interval in year i? By analogy with equation (6-79), it is simply

$$\frac{f_i I_i C_i d_i}{\lambda_i} \left( 1 - e^{-\lambda_i(t_2 - t_1)} \right) \quad (6-84)$$

This organ burden will decrease exponentially to the beginning of year n, at which time the organ burden will be

$$\frac{f_i I_i C_i d_i}{\lambda_i} \left( 1 - e^{-\lambda_i(t_2 - t_1)} \right) \left( e^{-\lambda_i(\text{EOY} - t_2)} \right) \left( e^{-\lambda_i(n - i - 1)365} \right) \quad (6-85)$$

Bear in mind that EOY is in units of days<sup>-1</sup>.

The integral organ burden during the year n as a result of the intakes during year i is

$$\int_0^{365} \frac{f_i I_i C_i d_i}{\lambda_i} \left( 1 - e^{-\lambda_i(t_2 - t_1)} \right) \left( e^{-\lambda_i(\text{EOY} - t_2)} \right) \cdot \left( e^{-\lambda_i(n - i - 1)365} \right) \left( e^{-\lambda_i t} \right) dt - \frac{f_i I_i C_i d_i}{\lambda_i^2} \left( 1 - e^{-\lambda_i(t_2 - t_1)} \right) \left( e^{-\lambda_i(\text{EOY} - t_2)} \right) \cdot \left( e^{-\lambda_i(n - i - 1)365} \right) \left( 1 - e^{-\lambda_i 365} \right) \quad (6-86)$$

This yields a dose equivalent of

$$\frac{S_i f_i I_i C_i d_i}{\lambda_i^2} \left( 1 - e^{-\lambda_i(t_2 - t_1)} \right) \left( e^{-\lambda_i(\text{EOY} - t_2)} \right) \cdot \left[ e^{-\lambda_i(n - i - 1) 365} \right] \left[ 1 - e^{-365\lambda_i} \right] \quad (6-87)$$

This is the dose equivalent during year n from an intake interval in year i. The total dose equivalent during year n must consider all intake intervals during year i, as well as all intake intervals in each year prior to year n. The total dose equivalent in year n from intake intervals in prior years is

$$\sum_{i=1}^{n-1} \left[ \frac{S_i f_i I_i C_i d_i}{\lambda_i^2} \left( 1 - e^{-\lambda_i(t_2 - t_1)} \right) \left( e^{-\lambda_i(\text{EOY} - t_2)} \right) \cdot \left[ e^{-\lambda_i(n - i - 1) 365} \right] \left[ 1 - e^{-365\lambda_i} \right] \right] \quad (6-88)$$

The total dose equivalent in year n then is the sum of equations (6-83) and (6-88).

$$D_n = \frac{S_n f_n I_n C_n d_n}{\lambda_n} (t_2 - t_1) - \frac{S_n f_n I_n C_n d_n}{\lambda_n^2} \left( 1 - e^{-\lambda_n(t_2 - t_1)} \right) + \frac{S_n f_n I_n C_n d_n}{\lambda_n^2} \left( 1 - e^{-\lambda_n(t_2 - t_1)} \right) \left[ 1 - e^{-\lambda_n(\text{EOY} - t_2)} \right] + \sum_{i=1}^{n-1} \left[ \frac{S_i f_i I_i C_i d_i}{\lambda_i^2} \left( 1 - e^{-\lambda_i(t_2 - t_1)} \right) \left( e^{-\lambda_i(\text{EOY} - t_2)} \right) \cdot \left[ e^{-\lambda_i(n - i - 1) 365} \right] \left[ 1 - e^{-365\lambda_i} \right] \right] \quad (6-89)$$

Values assigned to  $t_1$  and  $t_2$  are relative to the beginning of the year in which they occur, with this beginning always set equal to zero for each year. This means that EOY is always 365 days. Values of  $I$  must be in units of volume per day, where volume must be in the same units as appear in values for  $C$ .

A single exponential retention function has been assumed here. For multiple exponential functions,  $d_i$  in equation 6-85 is replaced by the deposition fraction,  $d_{i,k}$ , in the  $k^{\text{th}}$  compartment and  $\lambda_i$  replaced by  $\lambda_{i,k}$ . Equation 6-89 then is summed over all such compartments.

In practice, the subscripts may be dropped for some parameters. For instance, if the exposures are to the same material throughout the exposure history of the worker, then  $S_n$  may be replaced by the constant  $S$ . Usually, values of  $f$ ,  $I$  and  $d$  will remain constant for an individual adult. Care must be taken, however, to ensure that the aerosol particle diameter does not change with time, as this will affect values of  $d$ . If the radionuclide and its solubility class also remain constant in time, then the subscript of  $\lambda$  may also be dropped. This would, of course, imply that there is no significant age-dependence of  $\lambda$ , which typically is a reasonable assumption for working adult populations. Values of  $C$  must always be specific for the period during which intakes are assumed to occur.

There are a large number of sources for values to be employed in the above equations. The  $S$ -factors for most radionuclides of interest may be found in the reference by Dunning, Pleasant, and Killough (1977). A good, easily obtained, reference for values of  $\lambda$  is the Publication 30 report by the International Commission on Radiological Protection (ICRP 1979), although it is recommended that more accurate values be obtained from the primary literature when possible. An example of this is the rather extensive literature on uranium half-lives in lung for many uranium compounds. Values for  $I$  may be found in Publication 23 of the ICRP (ICRP 1975), in which typical physiological and metabolic parameter values are given for an assumed reference adult. It is recommended that respiratory values be obtained for light activity rather than for the more typically reported resting state. Finally, values of  $d$  for any particle AMAD (and age) may be obtained from a number of references (Crawford 1982; Crawford and Eckerman 1983; ICRP 1972).

The reader is cautioned that equation (6-89) assumes the intake is reasonably constant over each intake interval ( $t_2, t_1$ ). If there is reason to believe that this is not the case, then it may become necessary to further subdivide the intake interval into smaller intervals over which intake is reasonably constant. In addition, use of equation 6-89 may often present logistical problems if a computer code is to be employed (such as might occur in a large epidemiology study). A good test of the program is to allow an intake to occur during some interval and then compute the dose equivalent in that year and all subsequent years. The sum of these annual dose equivalents must equal

$$\frac{SfICd}{\lambda} (t_2 - t_1) \quad (6-90)$$

which is simply the expression for the dose equivalent commitment over an infinite time for intakes occurring in the intake interval.

Finally, use of area air monitoring results for the estimation of C should be approached with caution. Past studies indicate that area air monitors systematically underestimate worker-specific values of contaminant air concentration by approximately a factor of 3 (Caldwell 1967; Marshall and Stevens 1980). It might prove prudent to adjust such measured values accordingly, although extrapolation of this underestimate from one facility to another may not be valid.

#### Analysis of Dose Distributions from Internal Monitoring

Distributions of annual doses due to exposure to internal radioactivity are analyzed for lognormality or other functional forms in the same manner as external doses. Fractions of workers receiving doses below the cutpoints previously listed are computed for each target organ and radionuclide type by year. For each radionuclide and target organ combination, the number of workers monitored each year, the collective dose each year, and the mean and median dose each year are plotted versus year. If dose distributions appear to be lognormal, that is, if logprobit plots appear to be straight lines between -2 and +2 standard deviations, the geometric standard deviations of the annual distributions are plotted versus year. Otherwise, the standard deviations and/or variances are reported.

## Alternatives to Computing Doses

There may occur times when both bioassay data (urinalyses) and in vivo counting data (whole body or lung measurements) are deemed inadequate for purposes of dose estimation. In the following discussion, it is assumed that these forms of data are inadequate to compute both individual-specific doses and mean doses to a worker classification. If this is the case, an alternative must be found to dose estimation, one example of which was presented in the previous section on the use of air monitoring data. Several other alternatives are described below. In each case, the result is not an estimation of dose or dose equivalent but rather a semiquantitative measure of the potential for having received an exposure.

Consider first the case of air sampling data. The data themselves may be adequate to estimate exposure to each worker and for each radionuclide and chemical form, but metabolic or dosimetric models may be deemed inadequate or too speculative. The assessor may determine, therefore, that dose estimates are not warranted and choose to develop epidemiologic analyses based on exposure estimates. If only the dosimetric model is under question, then the analysis performed in the section "Method for Inferring Internal Doses from Air Samples and Time-in-Area Data" is utilized, with the exception that multiplication by the S-factor is not employed. The result then is an estimate of integral organ burden for each radionuclide.

If either the metabolic or lung deposition models are under question, then estimates of integral organ burden would not be obtained. Each worker, or group of workers, then is assigned a measure of exposure based on the air sampling data. Separate exposure estimates must be assigned for each radionuclide and for each chemical form. Since each radionuclide and chemical form will result in differing levels of dose or dose equivalents, these separate exposure estimates cannot be further collapsed and epidemiologic analyses must proceed through the use of stratification on each exposure estimate.

Finally, consider the case where air sampling data, bioassay data, and in vivo counting data are inadequate for fully quantitative estimation of either exposure or dose. In this case, a semiquantitative estimate of internal exposure must be obtained based on the limited, and perhaps biased,

measurement data. Several possibilities present themselves for generating such estimates:

1. A review of processes encountered within each job title or department may yield information as to the relative amounts of each radionuclide and chemical form employed by each process. An ordinal ranking of exposures to each radionuclide/form may then be possible, with subsequent epidemiologic analyses being performed with stratification. It is recommended also that this ordinal ranking be checked by comparison with available monitoring data. The best that can be done under these circumstances is to perform ordinal rankings of job titles/departments using means of each of the sets of available monitoring data specific to the job title/department. The ordinal ranking of these means should correspond to the ranking developed through the process review.
2. The means of a particular monitoring method, deemed to represent the most complete and systematic set of data, may be computed for each job title/department. These means then would be used to generate an ordinal ranking of job titles/departments. Problems arise, however, in that these means may be biased towards high measurement results, as discussed in earlier sections. It should not be assumed, therefore, that the means used here are accurate reflections of the (quantitative) relative hazards associated with these jobs. In other words, the fact that the mean for one job title is twice that of another does not indicate that doses differ by a factor of 2.
3. One could also rank job titles ordinally based on the fraction of measurement results (specific to that job title/department) exceeding certain bounds. A good general rule is to categorize job titles according to the fraction of results between 0 and 25 percent of a maximum permissible organ burden, 25 to 50 percent MPOB, 50 to 100 percent MPOB, and greater than 100 percent MPOB. Once again, separate analyses must be performed for each radionuclide/chemical form and stratified analysis

performed in the epidemiologic study. No attempt should be made to interpret the resultant ordinal ranking as a quantitative estimate of the relative hazards associated with job titles/ departments.

4. One could also rank job titles/departments according to the highest measurement (of a particular type) obtained from workers within that job title/department. Again, only ordinal rankings are possible and stratification must be employed. This approach is the weakest of the four outlined here and should result in the greatest degree of misclassification, since isolated, spurious, high results within a job title/department may be expected to occur periodically. Such spurious results may be due to machine error, fluctuations in background, or other factors, and will not be representative of results under normal operating conditions.

Clearly, these four approaches share some potential for large amounts of misclassification and should be avoided where possible. In addition, they result in exposure groupings which cannot be collapsed across radionuclides/chemical forms. If the sampling methods are not identical within a given group of facilities, it may also prove impossible to combine several facilities into a single epidemiologic study, since rankings obtained at one facility may differ (both qualitatively and quantitatively) from the others. Finally, these ordinal rankings cause problems in epidemiologic studies in which workers stay in job classifications for differing periods of time. Numerical "hazard or exposure indices" based on cases (3) and (4) above probably will bear only minimal resemblance to the relative indices that would be obtained under complete estimates of internal doses based on detailed sampling data. Approaches (1) and (2) are better in this regard, in that they employ mean values for each form of sampling result. In these cases, the correspondence between generated numerical "exposure or hazard indices" and estimates of internal dose based on detailed sampling data will be better as the sampling protocol approaches complete randomness (as defined on page 59 of this report).

## Uncertainties

The result of a radiation epidemiology study is an estimate of any increase or decrease in risk associated with exposure to radionuclides. While the risk itself typically is expressed quantitatively, the assumed causative factor (radiation exposure) might be expressed in any manner extending from the purely qualitative to the essentially quantitative. Ideally, the cause of increased risk in a group of individuals is assigned to some quantitative, objective property of the radiation, such as exposure, dose, or dose equivalent. This risk then may be assumed to be associated with any other circumstances characterized by an identical quantity of radiation, all other factors being equal. In this manner, risk factors derived from a given study can be extended to other situations of radiation exposure.

The question then arises as to how well a radiation epidemiology study can specify the precise cause of any perceived change in risk in a population. Clearly, this is possible only to the extent that the underlying exposure or dose estimates (including categorizations) are accurate, precise, and meaningful. A statistically strong risk estimate in a large population can be rendered useless if there is no way for determining the conditions that gave rise to the increased risk. If there is uncertainty as to the cause of a risk, there is also uncertainty in the implications of a study towards future action. The intent of this section is to explore, briefly, the components of uncertainty that enter into attempts to specify exposures or doses to a group of individuals. No attempt is made to present a mathematical development of the propagation of errors in dose estimation, since the methods themselves currently are under development and the use of error calculations in epidemiologic studies is rare.

For purposes of discussion, uncertainty in exposure or dose estimates will be placed into three broad categories. Each category contains components of the overall uncertainty associated with these estimates. As will be seen, the categories differ in the extent to which the uncertainties are expressed qualitatively or quantitatively and in the source of the uncertainty. How these various components of uncertainty are combined will

depend on the kind of analysis performed with the monitoring data and, hence, this section is designed only to outline the sources of uncertainty that should be explored in a study. Some recommended procedures for specifying the components of uncertainty also are provided. Support from statistical staff is essential in determining the combined uncertainty associated with a given dose or exposure estimate.

Within the first category of uncertainty there lies the question of whether a monitoring program (air samples, bioassay results, etc.) was appropriate to the task at hand. The researcher must review documentation on the monitoring procedures and ensure that the instrumentation and methods employed by the study facility were capable of measuring significant sources of exposure. If no such documentation were available, there is some uncertainty as to the relationship between reported measured values and actual exposures.

Consider, for example, the case of using air monitoring results for the estimation of lung doses. Documentation should be searched to determine whether the results were obtained at representative work locations and times and whether the air monitor was capable of detecting all significant radionuclides. If these facts cannot be determined, then one is faced with uncertainty as to whether the monitoring results are truly representative of air concentrations to workers. This uncertainty probably would be expressible only as written statements (a text) describing why the researcher is not certain that the measured value was the most meaningful measurement value to use in an assessment of exposure. This same textual supplement should be developed for each kind of monitoring result, with perhaps separate assessments for each use of the monitoring method in different work locations and at different times.

Included in the textual assessment is any question regarding the chemical form of a monitored radionuclide, the presence of other contaminants that might interfere with interpretation of the results, particle size of inhaled radionuclides, and ambiguities in such factors as degree of enrichment (in uranium). Where documentation or common knowledge cannot assure the insignificance of such questions, the resulting uncertainty should be described. This description should include an assessment of the cause of the uncertainty and some semiquantitative

determination of the possible impact of these errors on exposure or dose estimates. Where possible, it should be specified whether the source of uncertainty would lead to systematic errors in the study population or to random errors that might be expected to cancel when many workers are studied as a group. Usually, the best that can be done is to report the range of possible values that might be expected for the uncertain quantity or quantities.

Also included here is any descriptive background of the procedure used to assign job titles, work locations and individuals to the monitoring program. To the extent possible, the researchers should explain the degree to which they are certain that the more highly exposed populations or locations were monitored. Since unmonitored groups (if they exist) often are used as controls due to their assumed low exposures, an attempt should be made to determine how certain it is that the unmonitored group contains no highly exposed individuals. This can be done by displaying, through documentation, that the monitored workers or locations were chosen on the basis of clearly traceable and valid procedures. Such procedures might, at times, rest heavily on the subjective judgment of the health physics staff at a facility, and this should be noted in the text. A good check of the efficacy of these procedures is to determine the extent to which mean values of a measurement drop as the sampling frequency decreases. Workers with low sampling frequency are assumed (although not known) to be associated with low exposures, and it is a simple matter to test this empirically with the data. Such a test is performed by: (1) determining the number of monitoring results obtained for each person during a year; (2) pooling results on all people in that year who were monitored a given number of times; (3) determining the mean for each pool; and (4) plotting means against the number of times people in each group were monitored during the year. This is then repeated for each year of monitoring.

Similar remarks apply to attempts at grouping workers according to job title, department, or some other labels. If air monitoring results are employed, the researcher should assess the confidence associated with the assignment of a worker to a specific set of air measurements. If mean values of bioassay results are employed for a group, it is necessary to assess the confidence that the monitored workers in the group were

representative of the entire group. If there is some indication that monitored workers in a group were chosen on the basis of having the highest exposures, this should be noted and the resulting bias described. In this example, the bias would be towards an overestimate of the mean exposure in the group, although the bias would have an unknown magnitude.

Almost all dose estimates require the use of metabolic and dosimetric models obtained from literature sources. This would be true of dose estimates arising from both external and internal exposures, although uncertainties are larger in the second instance. These necessary models rarely are based on detailed understanding of the processes by which radionuclides and radiations interact with the human body and typically are formed on the basis of limited data. A complete assessment of the uncertainty in a dose assessment should include, therefore, some description of the extent to which metabolic and dosimetric models are firmly based in empirical findings. If a model is taken from the literature, it should be reported whether it differs significantly from other reported models. The extent of this difference can be determined quantitatively by starting with the same measurement value (such as air concentration) and calculating doses or organ burdens with each potential model. Assigning quantitative estimates of the relative validity of competing models is not possible at present, but it is possible to choose a common set of data (such as metabolic retention data from the literature) and determine a "goodness of fit" value for each model (for an example, see Williams and Leggett 1984).

It is also important to determine whether the models apply over lengths of time typical of epidemiologic studies, since many reported metabolic models are valid over restricted periods of time. In addition, models for the radionuclide of interest may have been extrapolated from other radionuclides or other chemical forms. This introduces uncertainty into the study since the researcher is not certain that the employed model is valid for the particular substance under study. If the models were generated from data obtained from nonhuman animals, or from data that were highly variable, this should also be noted as a possible source of error. It is doubtful that quantitative estimates of uncertainty can be developed for these possible sources of error, although the uncertainties in model parameters can be estimated if the original data on which the model was based are available.

In a similar vein, it should be understood that metabolic and dosimetric models are functions of age and the physical state of the individual. Reported models are almost always intended for use in normal adult humans. If a model can be shown to be a strong function of age or physical condition, the researcher must note the impact this might have in the present study. Alternatively, it might suffice to show simply that the distribution of age and conditioning is identical in all groups and doses appearing in the study.

Finally, the quality of the data reporting system must be assessed. The best measurement schemes can be negated if the researcher has no confidence that the results were correctly recorded. If hardcopy validation was performed, then the results of this validation should be noted, including the frequency with which transcription errors were detected and their magnitudes. Any residual ambiguities as to the units associated with measurement results should be described in detail and comments made on the possible impact of this problem. If there is any reason to believe that records may have been lost, this should be noted, especially if the lost records might be reports of large accidents or in any other way biased towards high or low values. Since several of the procedures outlined in the previous sections require separation of measurements resulting from accidents, the researcher must assess the certainty with which this separation can be performed. In cases where no accident codes were assigned by the facility and where hardcopy records cannot be used to complete this assignment, the researcher should describe the possible extent to which any upward bias might exist in exposure or dose estimates. The same comments apply to cases where the facility did not adequately flag results that were suspected to be errors in measurement.

The second category of uncertainty considers errors that might arise as a result of the measurement procedures themselves. Physical measurements contain random or systematic errors due to the process of measurement and the stochastic nature of the processes giving rise to the measured quantity. Basically, the researcher is asked to determine how accurate and precise a measurement could be, conditional on the facility's staff having correctly identified which workers ought to be monitored and what quantities ought to be measured. This source of uncertainty typically is divided into

considerations of the accuracy and precision of the measurements (Colle et al. 1980).

Using standard definitions (Colle et al. 1980), precision is taken here to mean the extent to which repeated measurements of an invariant physical property (such as mean air concentration or urine content) result in similar measured values of that property. In other words, the measured values are precise if they fall within a narrow range of values, regardless of whether this range contains the correct value for the property. Accuracy refers to the ability of a measurement process to yield a mean estimate that is close to the correct value, with no systematic departure from that correct value. For example, monitoring for uranium in urine by electrodeposition might be precise yet inaccurate if a constant and unknown fraction of uranium is lost to evaporation (this usually is not the case).

The researcher must determine the accuracy and precision associated with each measurement value employed in the exposure or dose calculations. Unfortunately, it often is difficult to separate these two components in practice and the separation should proceed only with advice from expert statisticians. To develop a measure of precision, it is necessary to find a set of measured values arising from measurements on the same, invariant, physical quantity. For example, the precision of in vivo lung counts might be ascertained by searching for multiple counts performed on an individual worker during some period of time that is short compared to the scale of time over which significant changes occur in organ burden. This procedure was followed in the paper by Crawford-Brown and Wilson (1984) for the precision of in vivo lung counts. Assuming that such repeated counts are available, the data for each individual are pooled separately, and means and standard deviations for each pool calculated. Plotting means versus standard deviations then yields a functional relation between measurement results and precision (not accuracy).

An alternative source for this information may be found in literature supplied by the instrument manufacturer, the precision reported may have arisen under highly controlled conditions in a laboratory. The precision obtainable in practice may be less by a significant amount due to poor quality control, changes in operating procedures at the facility, or varying exposure geometries and energies. Regardless of the source of precision

information, however, the researcher must guarantee that the measurements all were made on the same physical quantity under identical conditions. In addition, precision may change with time and with the magnitude of the measured quantity. Estimates of precision should, therefore, be made throughout the range of possible values of the measured quantity and during each year of monitoring.

Determining the accuracy associated with a measurement procedure is more difficult to obtain in practice. Some of the simpler causes of inaccuracy arise from calibration procedures designed to specify the relationship between a measurement result and the underlying organ burden, exposure, or dose. For example, calibration factors for film badges yield the relationship between optical density and exposure. Such a calibration factor would be inaccurate (as would the calculated exposures) if the gamma energy and exposure geometry used in calibration differed significantly from that found in the working environment. In this case, there would be a systematic bias in the exposure values for the entire population. The same comments would apply to air monitoring results obtained with an air sampler that failed to draw air at the assumed rate, and to urinalysis results that employed an incorrect background value or an inaccurate calibration sample.

For some measurement schemes, such as film badges, the facility will have employed functions relating the measured quantity (optical density) to exposure or dose. These functions may have been obtained by curve fitting to some set of data points obtained experimentally. Since the available data points usually are few, the coefficients of the resulting equation will be characterized by uncertainties. This source of inaccuracy may be determined by examining the original data used in generating the curve and estimating the uncertainty in the coefficients. As in the cases above, these inaccuracies would lead to a systematic bias in the dose estimates within a certain range of doses, although it must be noted that a fitted curve can yield an overestimate of the quantity in one part of the range and an underestimate in another.

In vivo counting schemes are beset by yet another source of uncertainty due to variability in body thickness among individuals (Cofield 1960). Conversion factors relating counts (in an external detector) to organ burden usually are obtained for a phantom representing the standard human body. If

the body thickness of an individual is larger than the standard value, then the calculated organ burden will be underestimated systematically. The opposite would be true for a worker who is thinner than the assumed standard body. This results in uncertainty as to the true organ burden that should be assigned on the basis of a measurement result on the individual. (Fortunately, there is a trend in health physics practice to try to incorporate body thickness, particularly for very low energy emitters like Pu and Am.) For a good review of this topic, see Palmer 1984. A measure of this inaccuracy can be obtained only when a group of workers can be identified as having identical organ burdens. This usually will be possible only for persons who have received no exposures or in instances where the "true" organ burdens can be obtained by some other measurement scheme. A possible source of the latter information is the reporting of both in vivo counting data and autopsy data for a group of individuals, while a source of the former information is measurements performed on newly hired workers who have had no previous exposures above background. In each case, several measurements should be available for each individual, thereby allowing separation of components of precision and accuracy. Precision is determined in this case by the procedure discussed on page 102. Accuracy is determined by calculating the mean measurement value for each worker and computing the standard deviation of these mean values about the true mean (which must be known prior to the calculations). This approach to estimating accuracy assumes the precision error is small.

In vivo counting is beset by another form of inaccuracy (Crawford-Brown and Wilson 1984) due to the simultaneous presence of a radionuclide in several organs. For example, uranium in the lung may be accompanied by uranium in the rib cage. External detectors will respond to both sources, leading to an inaccurate assessment of lung burden if all measured activity is assigned to the lung. Where possible, the researcher must determine the extent to which such problems may arise in particular instances at a facility and determine the extent to which these problems might bias dose estimates. Bias typically will be in a systematic direction for the entire population in the study, although the degree of bias may vary due to differing lengths of exposure. For the example of uranium in lung, use of chest counting (in vivo) data would yield a systematic overestimate of lung

burden that would grow larger as an individual worked for longer periods of time. If the resulting inaccuracy is not large, it might be satisfactory to ignore it and simply report its direction and magnitude. In other cases, some correction of the data might be performed to lower the inaccuracy if the extent of the bias is known.

This then, leads to the question of how well an exposure or dose could be specified for an individual given perfectly precise and accurate measurements at a facility. The third and final category of uncertainty concerns the fact that most dosimetric quantities are not measured directly but are calculated on the basis of some equations (see preceding sections). The dosimetric quantity then is a function of the measured quantity and of a number of other quantities that must be specified a priori. An example here is the use of urinalysis results to calculate lung doses, where the ratio of lung burden to urine content must be specified. This ratio will vary among individuals (intersubject variability) and within an individual over time (intrasubject variability). These components should be separated where possible.

The parameters necessary in converting from measurements to dose estimates usually involve biological properties of the exposed worker. These include, for example, lung mass, biological retention half-life, organ uptake fractions, and lung deposition fractions. Calculated doses usually employ the mean values for these quantities in the exposed population. Past experiences (Hofmann 1983; Cuddihy, McClellan and Griffith 1979; Hoffman et al. 1982; Dunning and Schwarz 1981; Schubert, Brodsky and Tyler 1967) suggest that the distribution of these quantities within a working population typically is approximately lognormal, with geometric standard deviations in the range of 2 to 3 and with truncation at 2.5 to 3 geometric standard deviations from the median value. For a listing of the pertinent mean values of biological parameters, the reader is referred to the appropriate ICRP publications (e.g. ICRP 1975). These values will deviate systematically for an individual.

In cases where urinalysis results are employed, calculation of organ burdens requires the use of a ratio between organ burden and urine content. Due to differences in organ retention and excretion factors, this ratio will vary between individuals, usually with both intrasubject and intersubject

components. These components of variability can be determined if organ burden measurements are available by both urinalyses and some independent (and accurate) method. A good external and independent method is in vivo lung counting at organ burdens significantly larger than the minimum detectable activity. In this case, the ratio of organ burden to urinalysis result is obtained for all such paired results in each worker. The variability of this ratio is characterized for each worker by examining variability inherent in the set of ratios obtained on the worker, yielding a measure of intrasubject variability for each worker. This is performed for all workers with sufficient sets of paired data. Similarly, the mean value of the ratio for each worker is determined and intersubject variability characterized by the distribution of these mean values among the workers (assuming the mean for each worker has been well determined).

If organ burden measurements separated in time are connected by interpolation functions, followed by integration of the function, an estimate should be made of the uncertainty in this process. Basically, this uncertainty arises due to the possibility that the true organ burden between the times of measurement may have deviated from that assumed in the interpolating function. It usually will not be possible to develop a distribution of the frequency with which the interpolating function yields an integral organ burden that is in error by a given factor. As an alternative, the researcher can develop this frequency for instances where a single intake occurred between two measurement times:

Assume that a single intake occurs between times  $t_1$  and  $t_2$ , which in turn are associated with organ burdens  $m_1$  and  $m_2$ . Let the effective retention function be  $R(t)$  for the radionuclide and organ of interest. Further assume that a linear interpolating function is used and that the resulting approximation to the integral organ burden is termed  $A$  such that

$$A = \frac{(m_1 + m_2)}{2} (t_2 - t_1) . \quad (6-91)$$

Consider now the case where an intake occurs at time  $t_i$  where  $t_1 < t_i < t_2$ . In this case, the exact integral organ burden,  $\Sigma$ , would be equal to

$$\Sigma = \int_0^{t_2 - t_1} m_1 R(t) dt + I \int_0^{t_2 - t_i} R(t) dt \quad (6-92)$$

where I is the magnitude of the intake in units of the organ burden.  
Clearly,

$$m_1 R(t_2 - t_1) + IR(t_2 - t_i) = m_2$$

or

$$I = \frac{m_2 - m_1 R(t_2 - t_1)}{R(t_2 - t_i)} \quad (6-93)$$

Therefore

$$\Sigma = \int_0^{t_2 - t_1} m_1 R(t) dt + \frac{m_2 - m_1 R(t_2 - t_1)}{R(t_2 - t_i)} \int_0^{t_2 - t_i} R(t) dt \quad (6-94)$$

Given  $m_1$ ,  $m_2$  and  $R(t)$ , it is then possible to compute the ratio  $A/\Sigma$  for any value of  $t_i$ . By assuming that intakes are equally likely at any point between  $t_1$  and  $t_2$ , it is possible to determine the frequency with which a particular value of the ratio  $A/\Sigma$  may be expected in a population of workers characterized by identical values of  $m_1$ ,  $m_2$ ,  $t_1$ ,  $t_2$ , and  $R(t)$ . If  $R(t)$  varies between workers, then this variability should be folded into the variability for  $A/\Sigma$ . As noted earlier, the dispersion of the resulting distribution of values of  $A/\Sigma$  will be an overestimate in cases when more than one intake gave rise to  $m_2$  (which is probably the usual case). This procedure will, at least, provide an upper estimate of the dispersion associated with this ratio, thereby giving aid in setting upper bounds for confidence intervals.

## Creating of Analysis Files and Assessment Report

### Analysis Files for External Monitoring

Since external monitoring files generally have one or more of the five categories of data (gamma, beta, neutron, penetrating, and skin) on each record, it is reasonable to maintain this structure at the analysis file stage. The analysis file has one record for each ID for each year that the worker was monitored or for each year for which doses have been calculated for that worker. Each annual record has variables ID, year, gamma dose equivalent, variance of gamma dose equivalent, beta dose equivalent, variance of beta dose equivalent, neutron dose equivalent, variance of neutron dose equivalent, penetrating dose equivalent, variance of penetrating dose equivalent, skin dose equivalent, variance of skin dose equivalent, number of records that were used to make annual records, and counts of flags for various problems or features of the data, such as number of "NEARBY" interpolations, number of unusable records and any other judgment flags that are needed. If monitoring was not performed for any of the five categories, then that field and its variance are left blank, not set to zero.

Complete documentation should be provided for such files and should be included in the final assessment report. Documentation includes length, position, and format information for each variable, as well as a description of the variable and its units. Those documents, as well as others collected during dose assessment, should be placed in a central file for ready retrieval.

### Analysis Files for Internal Monitoring

Internal monitoring results are organized by target organ. One analysis file is created for annual doses to each target organ for which doses have been calculated. Internal dose files have one record per ID for each year for which internal doses have been calculated for that worker. The reader is reminded that, for radionuclides with long effective clearance half-lives, a single intake may result in the inference of nonzero doses for

that worker for each year for the rest of his or her life. Each internal dose record contains the variables ID, year, dose equivalent, variance of dose equivalent (if feasible), and counts of flags.

Documentation for internal analysis files is similar to that described above for external analysis files.

#### Assessment Report

A final assessment report is prepared for each facility where dose assessment is performed. The report summarizes program evaluation findings, conversion algorithms, uncertainty analysis, and problems; and contains references to all primary and secondary literature concerning the site and its measurement procedures. Detailed documentation of all computer files used in dose assessment is referenced. The final assessment report documents what has been done to the data and why; what has not been or could not be done and why; questions, concerns, problems, and limitations; and caveats for the use of the data in epidemiologic studies. Systematic errors that may be present in the data are also discussed.

With the completion of the final assessment report, the dose assessment process is complete. Care should be taken to ensure that the assessment report details all steps in the assessment process, with particular concern to outline all assumptions made in the calculation of doses, exposures, or other groupings. This would include explanations of model choices, choices of parametric values, and assumed relations between monitoring results and exposures.

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APPENDIX A

DOSIMETRY RECORDS AND RADIATION HAZARDS  
QUESTIONNAIRE

Please address any questions regarding this questionnaire to:

William G. Tankersley, M.S.  
Oak Ridge Associated Universities  
P.O. Box 117  
Oak Ridge, TN 37830

Phone FTS xxx-xxxx or commercial (xxx) xxx-xxxx

\*This information will be used as part of the Department of Energy's Health and Mortality Study. This study is being conducted by Oak Ridge Associated Universities' Center for Epidemiologic Research with the collaboration of the University of North Carolina's School of Public Health under DOE Contract DE-AC05-76OR00033.

May 1982

I. FACILITY \_\_\_\_\_

A. Person in charge of dosimetry records

Name \_\_\_\_\_

Title \_\_\_\_\_

Address \_\_\_\_\_

\_\_\_\_\_

\_\_\_\_\_

Area Code & Phone No. \_\_\_\_\_

FTS Phone No. \_\_\_\_\_

B. Person filling out this form

Name \_\_\_\_\_

Title \_\_\_\_\_

Address \_\_\_\_\_

\_\_\_\_\_

\_\_\_\_\_

Area Code & Phone No. \_\_\_\_\_

FTS Phone No. \_\_\_\_\_

C. Please provide names and addresses of persons who have transferred or retired who may be helpful in locating or interpreting old records:

D. Are there written documents such as research reports, technical memos, internal evaluation memos, procedure manuals, etc., that would provide additional information on your dosimetry systems? Please list titles, authors, dates, document numbers, etc., and location where these documents may be found.

II. RADIATION HAZARDS AS A FUNCTION OF TIME

Please enter the approximate years during which your facility was engaged in the following operations which may have led to personnel exposure to ionizing radiation. Please indicate NA, for Not Applicable, where necessary.

CATEGORY	OPERATIONS	YEARS
NUCLEAR FUEL CYCLE	Uranium Mining, Milling, or Processing	
	Uranium Enrichment	
	Thorium Mining, Milling, or Processing	
	Nuclear Fuel Fabrication	
	Nuclear Reactor Operation	
	Nuclear Reactor Overhaul/Modification/Maintenance	
	Nuclear Reactor Refueling	
NUCLEAR WEAPONS	Nuclear Fuel Reprocessing	
	Nuclear Weapons Fabrication	
OTHER	Nuclear Weapons Testing	
	Industrial Radiography	
	Radionuclides: Sealed Sources	< 1 mCi
		1 mCi to 1 Ci
		1 Ci to 1 kCi
		> 1 kCi
	Radionuclides: Radiochemistry, Unsealed Sources	
	Accelerators: 0-1 MeV	
		1 to 100 MeV
		> 100 MeV
	Radiation Therapy at Your Site: Brachytherapy	
		Teletherapy
	Medical & Dental X Rays & Fluoroscopy at Your Site	
Other Operations, Please Specify		

III. EXTERNAL, PENETRATING DOSE  
(hard x ray and gamma ray, excluding neutrons)

A. For what years were employees exposed to this type of radiation?

For what years were employees NOT monitored for external penetrating doses?

B. For each type of personnel monitoring method, please enter in the following table the YEARS for which records are available in a given form.

		FORM RECORDS ARE IN					
EXTERNAL PEN. DOSE		Paper	Microfilm, Microfiche	Magnetic Tape	Punched Cards	On Line	Other, Specify below
M	Pocket						
O	Chamber						
NM							
IE	Film						
TT							
OH							
RO	TLD						
ID							
N							
G	Other, Specify below						

Please specify "Other Monitoring Method," if needed:

Please specify "Other Form Records Are In," if needed:

C. Please circle your entries in the above table which represent the primary dosimetry method which was used for legal records. If this changed over time, please attach descriptions of these changes.

D. If some or all of these records are not stored on site, please specify which records, and where they are located, or if they have been destroyed or lost.

E. If commercial dosimetry services were used, please specify companies, monitoring methods, and dates of service.

IV. EXTERNAL, SKIN DOSE (beta, soft x rays)

A. For what years were employees exposed to this type of radiation?

For what years were employees NOT monitored for external skin doses?

B. For each type of personnel monitoring method, please enter in the following table the YEARS for which records are available in a given form.

EXTERNAL SKIN DOSE	FORM RECORDS ARE IN					
	Paper	Microfilm, Microfiche	Magnetic Tape	Punched Cards	On Line	Other, Specify below
M   Pocket O   Chamber						
NM						
IE   Film						
TT						
OH						
RO   TLD						
ID						
N						
G   Other, Specify below						

Please specify "Other Monitoring Method," if needed:

Please specify "Other Form Records Are In," if needed:

C. Please circle your entries in the above table which represent the primary dosimetry method which was used for legal records. If this changed over time, please attach descriptions of these changes.

D. If some or all of these records are not stored on site, please specify which records, and where they are located, or if they have been destroyed or lost.

E. If commercial dosimetry services were used, please specify companies, monitoring methods, and dates of service.

V. FAST NEUTRON DOSE

A. For what years can fast neutron doses be separated from thermal neutron doses in your records?

For what years were employees exposed to this type of radiation?

For what years were employees NOT monitored for external penetrating doses?

B. For each type of personnel monitoring method, please enter in the following table the YEARS for which records are available in a given form.

FAST NEUTRON DOSE		FORM RECORDS ARE IN				
		Paper	Microfilm, Microfiche	Magnetic Tape	Punched Cards	On Line
M	Pocket					
O	Chamber					
NM						
IE	Film					
TT						
OH						
RO	TLD					
ID						
N						
G	Other, Specify below					

Please specify "Other Monitoring Method," if needed:

Please specify "Other Form Records Are In," if needed:

C. Please circle your entries in the above table which represent the primary dosimetry method which was used for legal records. If this changed over time, please attach descriptions of these changes.

D. If some or all of these records are not stored on site, please specify which records, and where they are located, or if they have been destroyed or lost.

E. If commercial dosimetry services were used, please specify companies, monitoring methods, and dates of service.

VI. THERMAL NEUTRON DOSE

A. For what years were employees exposed to this type of radiation?

For what years were employees NOT monitored for thermal neutron doses?

B. For each type of personnel monitoring method, please enter in the following table the YEARS for which records are available in a given form.

THERMAL NEUTRON DOSE	FORM RECORDS ARE IN					
	Paper	Microfilm, Microfiche	Magnetic Tape	Punched Cards	On Line	Other, Specify below
M   Pocket O   Chamber						
NM   IE   Film						
TT   OH						
RO   TLD ID						
N   G   Other, Specify below						

Please specify "Other Monitoring Method," if needed:

Please specify "Other Form Records Are In," if needed:

- C. Please circle your entries in the above table which represent the primary dosimetry method which was used for legal records. If this changed over time, please attach descriptions of these changes.
- D. If some or all of these records are not stored on site, please specify which records, and where they are located, or if they have been destroyed or lost.
- E. If commercial dosimetry services were used, please specify companies, monitoring methods, and dates of service.

VII. INTERNAL EXPOSURES

A. In your opinion, did internal radiation exposures constitute a significant fraction of exposures to workers when compared to external exposures? If yes, estimate the fraction of population dose (person-rem) which is due to internal exposures.

B. Please check the types of internal exposures which may have occurred at your facility:

- |  |  |
|--|--|
| <input type="checkbox"/> Fission Products    | <input type="checkbox"/> Plutonium                 |
| <input type="checkbox"/> Tritium             | <input type="checkbox"/> Other Transuranics        |
| <input type="checkbox"/> Iodine              | <input type="checkbox"/> Radium & Daughters        |
| <input type="checkbox"/> Activation Products | <input type="checkbox"/> Other Beta-Gamma Emitters |
| <input type="checkbox"/> Uranium             | <input type="checkbox"/> Other Alpha Emitters      |

C. Please estimate the number of persons who are known to have had radionuclides in their bodies. These radionuclides are broken down into categories of Maximum Permissible Body Burdens (MPBB's).

Radioactive Material	Number of Persons Who Had (Have)		
	10-99% of MPBB	100-999% of MPBB	>1000% of MPBB
Tritium			
Radium & Daughters			
Uranium			
Plutonium			
Other:			

D. For each type of personnel monitoring method, please enter in the following table the YEARS for which records are available in a given form.

INTERNAL MONITORING	FORM RECORDS ARE IN					
	Paper	Microfilm, Microfiche	Magnetic Tape	Punched Cards	On Line	Other, Specify below
Urin- M   alysis						
O						
N   Fecal						
I   Analysis						
T						
O   Whole						
R   Body						
I   Counting						
N						
G   Partial						
Body						
Counting						
(lung)						
M   Blood						
E   Tests						
T						
H   Air						
O   Samples						
D						
Other,						
Specify						
Below						

Please specify "Other Monitoring Method," if applicable:

Please specify "Other Form Records Are In," if applicable:

- E. If some or all of these records are not stored on site, please specify which records, and where they are located, or if they have been destroyed or lost.
- F. For what years were employees NOT monitored for internal radiation exposure?

VIII. UNITS AND QUALITY FACTORS FOR HIGH-LET RADIATIONS

A. For various kinds of HIGH-LET radiation, and for various time intervals, please specify the units of measurement, the quality factors used, if any, and the units in which radiation doses are recorded.

RADIATION TYPE	YEARS	UNIT OF MEASUREMENT	QUALITY FACTOR	UNIT OF RECORDS
THERMAL NEUTRONS				
FAST NEUTRONS				
ALPHA PARTICLES				
OTHER, SPECIFY				

Please make additional comments which may help us to use these dosimetry records.

IX. MISCELLANEOUS

A. Can neutron exposures be separated from x ray and gamma exposures?

B. Can internal and external doses be separated?

C. By present-day standards, were there significant radiation hazards which were not monitored for, or that were inadequately monitored? If so, please specify sources and types of radiation.

D. For what years do the following personal identifiers accompany the primary dose records?

IDENTIFIER	(EXTERNAL) YEARS	(INTERNAL) YEARS
Name		
Social Security No.		
Date of Birth		
Employee No.		
Dosimetry No.		
Race		
Sex		
Other:		

E. Are there some exposure levels below which personnel are not monitored? If so, what are those levels? Please break down by internal and external dose categories.

F. How are exposures at "less than minimum detectable level" reported? Please check:

Hard X Gamma	Soft X Beta	Therm. n	Fast n	Inter- nal	
_____	_____	_____	_____	_____	As equal to the "minimal detectable level"
_____	_____	_____	_____	_____	As "less than the minimum detectable level"
_____	_____	_____	_____	_____	As zero
_____	_____	_____	_____	_____	Other: please specify below

G. Have occupational exposure histories (that is, workers' doses from previous employers) been compiled at your site for radiation workers?

If yes, for how many workers has this been done?

Can these previous exposures be distinguished from on-site exposures in your records?

Please comment if needed:

PLEASE PHOTOCOPY THIS QUESTIONNAIRE AFTER YOU HAVE COMPLETED IT, AND RETAIN IT FOR YOUR RECORDS.

THANK YOU FOR YOUR TIME AND EFFORT!

Please return to: William G. Tankersley, M.S.  
Oak Ridge Associated Universities  
P.O. Box 117  
Oak Ridge, TN 37830

## APPENDIX B

### STANDARD ASSESSMENT REPORT

William L. "Jack" Beck, Paul S. Stansbury, and

James E. Watson

The following information is needed to evaluate the completeness and accuracy of dosimetry data required for the Department of Energy Health and Mortality Study.

#### I. HISTORY OF "HAZARDS" TO ASSESS OVERALL MONITORING PROGRAM

- A. What were the radiation hazards as a function of time?
- B. By present day standards, were there significant radiation hazards that were not monitored for or that were inadequately monitored?
- C. Describe in general the monitoring systems that have been used and the dates for each system.
- D. Are there other written documents such as research reports, technical memos, internal evaluation memos, procedure manuals, etc., that would provide additional information on your dosimetry systems? Where can these documents be found?

#### II. EXTERNAL MONITORING DATA

- A. Personnel Monitoring Badges
  1. What type of badge was used (film, TLD, etc.)?
  2. If more than one type, please give data for each different type used.
  3. What different modes of measurement were made (skin, penetrating, photon, beta, etc.)?
  4. Were dosimeters evaluated by commercial processor(s) or "in house"? If by commercial, give names and addresses and dates used.

5. If done in house, is there a procedure manual(s) available? If manual is not available, the following information is needed. Please give dates, etc.
  - a. Describe the calibration procedure and frequency.
  - b. Can the calibration be traced to NBS?
  - c. Describe the dosimeter evaluation process.
  - d. How often were "test" dosimeters evaluated and were they blind tests?
  - e. For test dosimeters, how accurate and precise were the results?
  - f. What other quality assurance procedures were used?
  - g. Were there specific training requirements for the dosimetrist?
6. What is the consensus of personnel operating the dosimetry service as to the accuracy and precision of the monitoring measurements?

B. Use of Badges

1. What part of the total worker population was badged?
2. What were the criteria for badging?
3. Were monitoring badges also security badges?
4. What percentage of the time did workers probably wear their badges?
5. Did workers tend to leave badges in desks, in cars, etc., often?
6. What procedure was used to provide monitoring if worker left his badge at home or lost his badge?
7. At what location did most workers wear their badge (shirt pocket, waist, collar)?

C. Other External Monitoring Techniques

1. Were pocket ionization chambers used? If yes, describe the type of chambers, procedure, quality assurance program, testing, and give overall estimate of accuracy and precision of results if possible.
2. Were other external personnel monitors such as NTA neutron film, activators, glass, or chemical dosimeters used? If yes, describe system as in part C-1 above.
3. Were are monitoring devices used? If yes, describe devices, etc., as in C-1 and explain how data were used in personnel monitoring program.

D. Administration and Record Keeping

1. What units were used in reporting results? Describe any conversion calculations.
2. Were quality factors (QF) or other modifying factors used to evaluate dose equivalent? If yes, describe procedure and QF's used, etc.
3. How were unusually high or low readings handled for determining if they were true readings or artifacts?
4. How were lost or obviously damaged dosimeters compensated for in dosimetry records of an individual worker?
5. Was there any compensation for natural background?
6. Are records known to be complete or are there known to be periods of lost data or records?
7. How were lost or unobtainable past personnel monitoring records compensated for in your record system?
8. Are monitoring data computerized? If yes, describe format of computer records. If no, describe or provide a copy of the form on which monitoring data are recorded.
9. What length of monitoring period(s) was used?
10. Are quarterly or yearly summaries available?

### III. INTERNAL MONITORING DATA

#### A. Bioassay Program

1. What types of bioassays were used (urinalysis, fecal, breath, etc.)?
2. What were the criteria for requiring bioassays?
3. How was the frequency of bioassays determined?
4. What radionuclides were analyzed for each method of bioassay analysis?
5. Are there procedure manuals available? If not, the following information is needed about each different method.
  - a. Description of method of analysis.
  - b. Units and description of any calculations or conversions used in obtaining final answers in dose or dose equivalent.
  - c. Procedure for calibration of counting equipment.
  - d. NBS traceability.
  - e. Estimated accuracy and precision of measurement technique; and limits of detection.

#### B. Whole-Body Counting

1. Was whole-body counting (WBC) used?
2. What were the criteria for requiring a whole-body count?
3. Was WBC done in house or by a commercial company? If done by others, identify company and if possible, the person responsible for measurements.
4. If in house WBC, describe counter, limits of detection, calibration procedure, calibration traceability to NBS, estimated accuracy and precision of measurement.
5. What calculations or modifications were done to counting data to determine radionuclide content of worker?

6. Were any conversions to dose or dose equivalent done? If yes, describe procedure used for conversion.

C. Other Internal Monitoring Techniques

1. Were air monitoring results used to estimate internal deposition? If yes, describe the equipment, usage procedures, calibration, and the method of interpreting measurements. Give results of accuracy or precision of monitoring, if available.
2. Were any other monitoring methods used to estimate internal deposition other than bioassay, whole-body counting or air monitoring? If yes, describe in detail as outlined in Part 1 above.

D. Administration and Record Keeping

1. Are internal monitoring reports computerized? If yes, what is the format of the data? If no, what information is available, and are there quarterly or yearly summaries?
2. How were unusually high or low values validated?
3. If artifacts were discovered, how was individual worker's record corrected?
4. What procedure was used to merge internal and external dosimetry data?

Please make any other comments that you think are needed for a better understanding of the personnel monitoring programs at your facility.

## Appendix C

### Computation of Notional Dose from Nearby Records

A computer program to generate a notional dose each time a flawed film badge record is encountered has been developed (Hudgins and Strom 1983). The program, called NEARBY, generates a dose number from records chronologically near the flawed record. The program uses a hierarchy of twelve options to generate a notional dose, and if all 12 options fail, it flags the flawed record as "unusable." The hierarchy is shown in Table C-1, with the option numbers labeled "priority."

Using a computer file sorted by ID, year, and calendar quarter (ordinary dates could be used instead of year and quarter), NEARBY examines the two records immediately preceding the flawed record, as well as the two following records (the "nearby" records). All nearby records are checked for the same ID as the flawed record, for dates within two calendar quarters of the flawed record, and for absence of "unusable" or "damaged" flags. Any record failing any of these checks is rejected for use by NEARBY. If a record with an identical date is found, either before or after the flawed record, the dose information from the nearby record is simply substituted for that in the flawed record (options 1 and 2). Failing that, the record immediately preceding the flawed record and the one immediately following the flawed record are examined for valid data, and the dose data averaged to create a notional record to substitute for the flawed record (option 3). Five other variations of two-record averaging are tried if option 3 fails (options 4 through 8). Options 3 through 8 are essentially linear interpolations between nearby doses. If no two-record options works out, then four possible one-record substitution options are tried (options 9 through 12). If none of the options works, the flawed record is flagged as unusable, or a judgment made to extend the time boundaries beyond two calendar quarters (if exposures have not evolved significantly).

Table C-1

Generation of Notional Doses to Replace Flawed Records

NEARBY OPTIONS						
Priority	-2rec	-1rec	0 rec	+1rec	+2rec	NEARFLAG
1		x	copy	(same date as 0)		A
2	(same date as 0)		copy	x		B
3		x	average	x		C
4	x		average	x		D
5		x	average		x	E
6	x		average		x	F
7	x	x	average			G
8			average	x	x	H
9		x	copy			I
10			copy	x		J
11	x		copy			K
12			copy		x	L
13			unusabl			U

- 
- 2rec Indicates the record that appeared two records before the flawed record.
  - 1rec Indicates the record that appeared immediately before the flawed record.
  - copy Means that dose data are copied without alteration from the record having an "x" under it in the table.
  - average Means that the doses from records have x's under them in the table are averaged before being substituted for the doses in the flawed record.
  - NEARFLAG A variable whose value indicates which of the NEARBY options was used to create the notional record.