Paper

SCIENTIFIC ISSUES IN RADIATION DOSE RECONSTRUCTION

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Abstract—**Stakeholders have raised numerous issues regarding the scientific basis of radiation dose reconstruction for compensation. These issues can be grouped into three broad categories: data issues, dosimetry issues, and compensation issues. Data issues include demographic data of the worker, changes in site operations over time (both production and exposure control), characterization of episodic vs. chronic exposures, and the use of coworker data. Dosimetry issues include methods for assessment of ambient exposures, missed dose, unmonitored dose, and medical x-ray dose incurred as a condition of employment. Specific issues related to external dose include the sensitivity, angular and energy dependence of personal monitors, exposure geometries, and the accompanying uncertainties. Those related to internal dose include sensitivity of bioassay methods, uncertainties in biokinetic models, appropriate dose coefficients, and modeling uncertainties. Compensation issues include uncertainties in the risk models and use of the 99th percentile of the distribution of probability of causation for awarding compensation. A review of the scientific literature and analysis of each of these issues distinguishes factors that play a major role in the compensation decision from those that do not.**
Health Phys. 95(1):26–35; 2008

Key words: dose reconstruction; dosimetry, external; dosimetry, internal; nuclear workers

INTRODUCTION

THIS PAPER addresses some of the scientific issues involved in the National Institute for Occupational Safety and Health (NIOSH) Radiation Dose Reconstruction Program and the decisions made to deal with them. Numerous issues have been raised in stakeholder comments on the dose reconstruction process, and a comprehensive literature review has identified the scientific basis for the ways the project has resolved them. For purposes of discussion, the issues are divided into three general subject areas: data issues, dosimetry issues, and compensation issues.[†] This paper discusses a few of the issues in detail, while others are discussed only briefly here, with the details presented in the accompanying papers in this issue of *Health Physics*.

While dose reconstruction for a compensation program must be based on sound science, it also must be conducted in a timely manner and ensure accurate compensation decisions. Consequently, there are significant differences in approaches to dose reconstruction for compensation as compared to dose reconstruction for use in epidemiology studies, incident assessment, or litigation. Compensation-oriented dose reconstruction is usually oriented toward overestimates, so as to preclude the denial of benefits to a deserving claimant. Thus, when equally plausible choices are available among exposure scenarios, the dose reconstructor should choose the scenario that is favorable to the claimant (U.S. DHHS 2002a), or assign an upper-bound estimate of the dose (U.S. DOD 1985).

DATA ISSUES

Federal directives (U.S. DHHS 2002a) establish priorities for data to be used in dose reconstruction. The top priority is assigned to individual monitoring data for the worker, followed by monitoring data for coworkers, area monitoring data, and finally, process data, such as the types and quantities of radioactive materials handled in the workplace. A frequent public comment is that a worker's dose cannot be reconstructed because his or her individual monitoring records do not exist, were falsified, or were otherwise inaccurate. However, this begs the question; if complete and accurate records are available, dose reconstruction is not necessary—all the required information already exists. The essence of dose reconstruction is to fill the voids in the monitoring records by using a combination of science and professional judgment to generate a value of the appropriate dose parameter that is adequate for an unambiguous compensation decision.

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(*Manuscript accepted* 6 *August* 2007)

0017-9078/08/0

[†] Compensation statistics are available at http://www.dol.gov/esa/regs/compliance/owcp/eeoicp/WeeklyStats.htm. Accessed 22 March 2007.

Furthermore, the Energy Employees Occupational Illness Compensation Program Act (EEOICPA) specifically calls for the inclusion of all occupational sources of radiation exposure incurred during nuclear weapons testing, manufacture, or maintenance. Consequently, the dose of record (if any) is only the starting point, and all other significant sources of exposure must be characterized and included. For example, individual monitoring records usually do not include doses from medical radiography required as a condition of employment, nor data on the sensitivity of the dosimeters or bioassay methods employed.

Demographic data

Several types of demographic data are critical to the process of dose reconstruction and compensation determination. The starting point is the worker's personal data: date of birth, facility (or facilities) at which employed, date of first employment at a covered facility, date of cancer diagnosis, and cancer type. Age at both first employment and at cancer diagnosis must be known because the cancer risk models used to determine the probability of causation (PC) are age-dependent. In addition, for skin cancers, ethnicity must be known, and for lung cancers, smoking history.

Demographic data of interest, based on a random sample drawn from the first 16,500 cases, show that 41% of cases referred for dose reconstruction began employment prior to 1951, 18% worked for less than 5 y, 43% worked for 5–25 y, and 39% worked for more than 25 y. As would be expected, latent periods for cancer diagnosis, defined as the interval between start of employment and diagnosis, exceeded 15 y for 87% of the cases. The most common cancer types were lung, skin, and prostate, and approximately 60% of the cases had one of the 22 "presumptive" cancers that are automatically compensated under the provisions of the Special Exposure Cohort (U.S. DHHS 2004). As of this writing, the Department of Labor (DOL) has referred 24,700 cases to NIOSH for dose reconstruction, and there is no reason to suspect any significant changes in the percentages reported above.

Site operations

Perhaps the greatest challenge in a worker dose reconstruction program is obtaining adequate data to characterize site operations and all plausible sources of significant occupational exposure to radiation and radioactive materials. During the Manhattan Project and the height of the Cold War, most site records focused on production issues. To convert this information into data useful for dose reconstruction, profiles have been prepared for all the major U.S. Department of Energy (DOE) sites (Kenoyer et al. 2008; Rollins 2008). Even having access to the data in a site profile, a dose reconstructor frequently must make claimant-favorable assumptions on parameters such as solubility, particle size, and relative equilibria of progeny radionuclides. However, the assumption that is favorable to the claimant depends on the type of cancer involved. For an inhalation exposure, assuming an insoluble form of the radionuclide is claimant-favorable for cancers of the respiratory tract, but for cancers of systemic organs, assumption of a soluble form is usually more favorable.

Data capture

The NIOSH Radiation Dose Reconstruction Program required an extensive data capture effort. This was accomplished through visits by experienced health physicists and records specialists to covered sites, Federal records repositories, libraries, and other sources of relevant data. Because of the need to validate the data, and because in many instances, independent corroborating data are not available, a decision was made to focus on collecting the "rawest" data available. That is, the most useful data are individual dosimeter readings, bioassay measurement results, air concentrations, and so on, rather than processed data such as assigned dose, radionuclide intakes, and the like. Hard copy data are preferred, although electronic data may be the only data available in some cases. Because of the limited storage capacity of early electronic data processing systems, such data often are in coded form and can be difficult to interpret. Consequently, code books, user manuals, and other reference materials describing electronic data systems were also collected. More details on this subject are presented elsewhere in this issue (Martin et al. 2008).

Individual monitoring data

The availability and quality of individual monitoring records decreases with time since exposure, and related data may be completely lacking for workers at Atomic Weapons Employer (AWE) facilities and at government-owned contractor-operated (GOCO) facilities operated under the auspices of the Manhattan Engineer District (1942–1947). The DOE was required by Presidential Executive Order 13179 (2000) to provide all available individual monitoring records for workers. As of December 2006, DOE had provided NIOSH with over 19,500 responses to requests for individual monitoring records. An analysis of the monitoring records provided for the above-mentioned random sample of the first 16,500 cases shows that the quantity and quality of the records provided are highly dependent on the facility and dates of the worker's employment. External monitoring records covering most, if not all of the worker's employment

period have been provided for 52% of the examined cases, and some external monitoring data have been provided for another 15%. Cases with no data were either not monitored in earlier eras, or were not classified as radiation workers while employed. Internal monitoring data are available for 48% of the cases examined, but records of positive bioassay data are available for only 10% of these.

Coworker data

Extensive sets of worker monitoring data gathered for epidemiology studies are readily available from NIOSH's Division of Surveillance, Hazard Evaluations, and Field Studies [DSHEFS, formerly the Health Effects Research Branch (HERB)]; from the Comprehensive Epidemiologic Data Repository (CEDR) created by DOE and maintained at Lawrence Berkeley National Laboratory (U.S. DOE 1999); and from other data collections. In general, these data sets do not contain complete monitoring data for every worker at a given facility, but are sufficiently robust to generate statistical distributions of the exposure data for a given worker population. Procedures for the analysis and application of these data have been developed for both external and internal exposures. Sites for which such data are available include Fernald, Hanford, Los Alamos, Linde, Mound, Oak Ridge National Laboratory (X-10), Oak Ridge Y-12 Plant, Pantex, Rocky Flats, and Savannah River. Additional worker exposure data have been captured for some of these and other sites from onsite and offsite records repositories.

The use of coworker data for dose reconstruction is discussed in more detail by Merwin et al. (2008a and b). Briefly, a lognormal distribution is fitted to each set of data, and an unmonitored worker can be assigned an appropriate percentile of the distribution, depending on the likelihood of exposure as determined from employment records and interviews. For external dose, the assigned percentile is considered to be an upper bound estimate. For internal dose, the geometric standard deviation of the fitted distribution can be used to develop the uncertainty bounds for the assigned dose.

DOSIMETRY ISSUES

External dosimetry

There are four significant scientific issues involved in assessing the completeness, accuracy, and uncertainty of the recorded external dose. A similar assessment must be performed for the missed external dose. Merwin et al. (2008a) discuss external dose reconstruction in detail.

Badge sensitivity and accuracy. The sensitivities and accuracies of both film and thermoluminescent dosimetry (TLD) badges to various types of radiation have been well established in the scientific literature (NRC 1989; Thierry-Chef et al. 2002). However, some authors may disagree on specific issues, such as the response of neutron track emulsion (NTA) film to neutrons of energy less than 1 MeV. In such cases, a claimant-favorable assumption is made when available data are insufficient to settle the issue. For example, neutron dose may be assigned on the basis of neutron-to-photon ratios, even though not all photon exposure was coincident with neutron exposure. Conversion of badge readings to organ dose requires the development of tables of dose coefficients relating dosimeter-measured quantities [e.g., deep dose equivalent, $H_p(10)$], to organ dose equivalents as functions of energy and exposure geometry (NIOSH 2002a).

Badge response as a function of energy and geometry. Dosimeter responses are dependent on both the energy and the angle of incident radiation, and numerous authors have investigated these properties for different types of dosimeters. A comprehensive study by Thierry-Chef et al. (2002) measured dosimeter response to 5 mGy air kerma and compared the defined value of $H_p(10)$ for that exposure to the dosimeter response at three different photon energies (118, 208, and 662 keV) and three irradiation geometries (anterior posterior, rotational, and isotropic). For older film dosimeter badges, the response as a function of photon energy and irradiation geometry varied from 0.45–3.60 of $H_p(10)$, while for newer TLD badges, the response varied from 0.8–1.2 of $H_p(10)$. With these and similar data, plus the characterization of workplace source geometries, the necessary corrections to the recorded doses can be made as part of the dose reconstruction.

Missed external dose. In many cases, the missed external dose, defined as the dose that could have been received by the worker that was not registered by the dosimeter, is likely to exceed the dose of record, especially in the early days when dosimeter sensitivity was relatively low and badge exchange was more frequent. Taulbee et al. (2001) examined a number of methods for estimating missed dose and determined that the assignment of half of the limit of detection of the dosimeter for each monitoring interval in which the dosimeter did not register a dose yields an estimate that tends to be favorable to the claimant (i.e., an overestimate of the true missed dose). Consequently, this value is routinely assigned in external dose reconstruction.

Internal dosimetry

Scientific issues pertaining to internal dose reconstruction concern the appropriateness of bioassay methods used for monitoring, including sampling frequency, sensitivity, specificity, validation, uncertainty, and reporting; the accuracy, applicability, and uncertainty of biokinetic models used to relate bioassay results to intake and dose; the accuracy and applicability of air monitoring practices; and the methods to estimate unmonitored dose. Brackett et al. (2008) discuss internal dose reconstruction in detail.

Bioassay methods. The internal dosimetry sections of the site profiles provide summaries of the bioassay methods in routine use at the various sites. These may be divided into two classes, direct (in-vivo) and indirect (in-vitro) measurements. The former comprise whole- or partial-body counts (i.e., direct measurements of photon-emitting radionuclides in the body with external detectors). The latter comprise measurements of radionuclides in excreta samples, or samples collected from the worker, such as a nasal swab. Both types of measurement have been in routine use for almost 100 years, dating back to measurements of ^{226}Ra in the early radium workers (Schlundt et al. 1929). External measurements of high-energy photons emitted by internal radionuclides have had an accuracy of approximately 30% or better since the 1950's (Spiers 1962). However, direct measurement of inhaled transuranics, which emit only low-energy photons, remains a formidable problem. Measurements in the 1970's had an uncertainty of more than a factor of three, while modern techniques have improved the accuracy to better than a factor of 1.5 (IAEA 1995). Similarly, radiochemical methods used for the analysis of excreta samples after a known intake (as opposed to routine screening) were intended to be capable of detecting radionuclides in the sample at levels corresponding to 25% of the established reference level (i.e., the maximum permissible body burden) (Harley 1964). However, indirect bioassay requires the use of a biokinetic model to estimate what fraction of the body content is contained in the sample. Both direct and indirect measurement results depend on biokinetic models to estimate intake and resulting dose.

An issue frequently raised by stakeholders regarding the accuracy of bioassay measurements is their validation against air monitoring results. Although some authors have reported positive correlations between air monitoring and urinalysis for uranium workers (Lippmann 1958; Chase 1989), others have reported no correlation (Schultz and Becher 1963; Spitz et al. 1984; West et al. 1995). Consequently, some authors maintain that air samples are primarily useful for detecting that an intake has occurred, but are too uncertain to quantify the intake (Gibson 1994). Because of the difficulty in determining the spatial relations of a given worker and a given air sampler at any particular point in time, it was decided to use air monitoring data only if no bioassay data (either individual or coworker) were available.

Perhaps the most significant problem with the interpretation of bioassay data is estimating the missed dose, especially when the data were censored (i.e., all results below some level were recorded as zero). As in the case of interpretation of positive bioassay data, the Integrated Modules for Bioassay Analysis (IMBA) computer software (Birchall et al. 2003) is used to determine the maximum chronic intake level that could have continued to occur while simultaneously yielding bioassay results below the detection limit. The resulting intakes are used to compute the relevant organ doses that could have been missed for both workers with no positive bioassay data, and workers with some positive data that declined to levels less than the detection limit.

Biokinetic models. The *Internal Dose Reconstruction Implementation Guideline* (NIOSH 2002b) calls for the use of current biokinetic models published by the International Commission on Radiological Protection (ICRP) for the interpretation of bioassay data and dose assessment. It must be noted that none of these models pertains to an actual individual; they are all based on Reference Man, an idealized, average human (ICRP 1975). In the case of the biokinetic model for plutonium excretion (ICRP 1993), estimates of the inter-individual variability in the transfer coefficients defining the model have geometric standard deviations in the range of 1.75 (Luciani et al. 2003), while Leggett (2001) reported geometric standard deviations up to 2.0 in the data on which the models are based. However, these uncertainties can be propagated through the dose calculations and entered into the IREP software (Kocher et al. 2008) to be combined with the uncertainty in the risk coefficient.

Several specific issues have been raised by stakeholders with regard to the use of ICRP's human respiratory tract model (ICRP 1994); these include concerns about oro-nasal breathing, breathing rates, physicochemical form and solubility of inhaled radionuclides, particle sizes, and chronic vs. acute intakes. Fortunately, the model is sufficiently robust to address all these issues, and where data are lacking, assumptions favorable to the claimant are made. All parameters of the model may be adjusted as needed in the IMBA software. For example, concern was raised over the inhalation of high-fired plutonium oxide by workers at the Rocky Flats plant, and its relative insolubility. For cases where the reconstruction of internal dose is based on direct

bioassay data, the effect of particle solubility on intake and dose is easily determined by adjusting the retention time of the long-term lung component from a default half-time of 7,000 d to 50,000 d or more (Carbaugh and La Bone 2003). For equal intakes, the assumption of a highly insoluble material is highly unfavorable to the claimant for all cancers except those of the respiratory tract.

However, if internal dose reconstruction is based on urine bioassay data, it is important to note that the estimate of an inhalation intake is inversely proportional to the solubility of the inhaled material for a given excretion level. Consequently, the determination of what constitutes claimant-favorable assumptions must frequently be determined by using the IMBA software to generate dose coefficients (i.e., Sv Bq^{-1} intake) for various organs as functions of particle size and solubility, and combining those with intake retention and excretion fractions (functions of particle size, solubility, and time post-intake), to generate dose coefficients per unit content of radionuclide in a bioassay measurement (Berkovski et al. 2003).

Air monitoring. As mentioned above, air monitoring data are considered by many researchers to be primarily useful to assess the potential for inhalation intakes, but not to quantify them. However, in some cases, particularly at AWE facilities where no personnel monitoring was performed, air monitoring data are likely to be the only quantitative data available on which to base an exposure model. Fortunately, numerous air samples from AWEs were collected and analyzed by the U.S. Atomic Energy Commission's Health and Safety Laboratory (later the Environmental Measurements Laboratory) (U.S. AEC 1949). Obviously, air monitoring data collected from the facility of interest is preferred, but if such data are sparse or completely lacking, data from another facility conducting similar operations may be used, based on reasonable and scientific assumptions (U.S. DHHS 2002a).

An example of this method is provided by the site profile for Bethlehem Steel in Lackawanna, NY (NIOSH 2006a). Bethlehem Steel size-reduced rough-rolled uranium rods to a smaller diameter rod for the Hanford plutonium production reactors on 14 documented occasions of one or two days duration, and some air samples were collected during some of the rollings. Another rolling mill, Simonds Saw and Steel in Lockport, NY, did both rough and finish rolling of uranium rods from 1948–1956, and processed a total of between 25 and 35 million pounds of uranium, on approximately 1,000 work days (ORAUT 2005). Air samples were collected at Simonds Saw and Steel on 15 different dates from a number of locations within the facility. The early monitoring data from Simonds Saw, when rough rolling was conducted, were used to construct an inhalation exposure model for workers at Bethlehem Steel in the 1949–1950 time period that consisted of a log-normal distribution with a 95th percentile of 553 MAC (maximum allowable concentration, which at the time was equal to 50 μg U m^{-3}).

Unmonitored dose

Unmonitored dose may be divided into two broad categories: dose received by workers who were not monitored, and generally did not need to be monitored because of low exposure potential; and dose received by workers who were, or should have been monitored, but at least some of whose doses were not monitored, or perhaps not recorded. In the former case, it is often sufficient to assign only the ambient environmental dose received on the site, especially for administrative personnel who did not enter production areas. In the latter case, the use of coworker data is the preferred dose assessment method, and it is particularly useful to address the frequent claim that workers deliberately removed their badges before conducting a "hot" job. A plot of coworker data on semi-logarithmic probability paper produces a straight line if the data are lognormally distributed. Frequently, the high end of the line will tail off, usually because of a paucity of high-level exposures resulting from both exposure control practices and the exposure mechanisms themselves (Kumazawa and Numakunai 1981; Daniels et al. 2004). The observed distributions are better fitted by a hybrid lognormal distribution, consisting of a lognormal portion at lower doses and a normal distribution at the highest doses. Because the variation in dose of a given worker will be proportional to the dose, the law of proportionate effect applies (Aitchison and Brown 1957), and the dose distribution, in the absence of radiation controls, will be a true lognormal (Kumazawa and Numakunai 1981). Therefore, the unmonitored dose, if due to removal of monitors for high dose jobs, can be reliably determined from the lognormal distribution of recorded doses. It is a straightforward matter to extrapolate the linear portion of the probability distribution to obtain percentiles of the dose distribution that would have been observed, and the same method is reliably used to determine missed dose due to censoring of exposure data below the detection limit of the monitor (Daniels and Yiin 2006). Similar methodologies are applied to unmonitored internal doses, with the exception that lognormal fits are made to bioassay results, which then are converted to doses with the appropriate biokinetic models.

Medical screening dose

The NIOSH Radiation Dose Reconstruction Program is unique in that radiation doses received from medical radiographic screening procedures that were a condition of employment are included in the occupational dose assessment. The required medical examinations typically included annual chest radiographs and, in cases of fluoride exposure, pelvic radiographs or other bone densitometry. These doses were never individually monitored, and so must always be reconstructed, but fortunately the radiography procedures used at the various sites are reasonably well documented and typical of the time. Each site profile includes a medical dose section (Shockley et al. 2008), and medical doses typically do not make a significant contribution to the occupational dose, with the notable exception of chest radiography performed with mobile photofluorographic units at Savannah River and other sites in the 1950's (Cardarelli et al. 2002).

Ambient dose

At most of the major DOE sites, ambient external doses were monitored with control film or TLD badges, normally kept in the same badge racks where worker badges were stored. Because these values were subtracted from the worker badge readings, they need to be added back into the recorded external doses. However, the issue of how representative the control badges were of the ambient external doses received by the workers has to be addressed on a site-by-site basis. This issue is further complicated by the fact that the control badges also record natural background radiation, which is to be excluded from the occupational dose, unless there was exposure to enhanced natural background radiation due to weapons-related work. As described by Rollins (2008), the ambient environmental doses received by workers at the various sites have been computed from raw stack monitoring data wherever available. The standard methodologies used for computing offsite environmental doses to nearby populations are applied to the onsite population by developing dispersion coefficients (χ/Q values) for occupied locations onsite. These values are combined with the monitored stack release rates to validate external dose estimates (Merwin et al. 2008a) and to calculate radionuclide intakes for internal dose estimates (Brackett et al. 2008).

COMPENSATION ISSUES

Probability of causation

One fundamental scientific principle upon which the process of dose reconstruction for compensation rests is that radiation is a weak carcinogen. In the Life Span Study through 1997 of 49,114 Japanese atomic bomb survivors who received significant doses (>5 mSv), there were 18,049 deaths, of which 5,502 were from solid cancers; of these, only 8% (440) were attributable to radiation exposure (Preston et al. 2003; NAS 2006). There was a significant difference in attributable risk between solid tumors and leukemias and other haematopoietic cancers, however, with an attributable risk of 44% for the latter (Pierce et al. 1996).

Summarized in Table 1 is a tabulation of minimum annual organ equivalent doses (from photons >250 keV) that produce a median value of the PC of at least 50% for the hypothetical scenario of a worker who is diagnosed with a specific cancer at age 60 y and received a constant annual equivalent dose for 30 y (ages 21–50 y). The cancers listed include the 22 presumptive cancers automatically compensated in the Special Exposure Cohort (U.S. DHHS 2004) and two common, nonpresumptive cancers, skin and prostate. The organ for which the equivalent dose was calculated for each type of cancer listed was selected in accord with NIOSH guidance (NIOSH 2002b). As may be noted, the annual organ equivalent doses range from 22 mSv for the liver to 500 mSv for multiple myeloma/non-Hodgkins' lymphoma. The corresponding cumulative doses range from 0.66–15 Sv. This tabulation is specific for the stated scenario and is intended only to illustrate the wide range of doses required to yield a median PC of 50% for cancers in the organs listed. At the same time, it must be remembered that the PC depends on age at starting exposure, age at diagnosis, cancer type, and gender as well as dose.

The values in Table 1 are those that result in a given cancer being "as likely as not" to be caused by the radiation exposure. As such, it would meet the legal standard of causation (U.S. DHHS 2002b). In point of fact, these are extremely high doses, and exceed by a wide margin the actual doses received by most workers in the nuclear weapons complex. Perhaps the most comprehensive epidemiological study of workers in the nuclear industry is that conducted by the International Agency for Research on Cancer (IARC) (Cardis et al. 1995). A review of this report by Jose et al. (2001) indicated that only about 1% of all workers were members of the highest dose category, namely a cumulative dose equivalent of over 400 mSv. No cancer in Table 1 has a median PC of 50% at a cumulative dose of 400 mSv; the lowest cumulative dose is 700 mSv for leukemia. However, the IARC study was limited to workers with the potential for external exposure; the lifetime doses in the IARC report were based on external monitoring records and do not include internal, medical, missed, and unmonitored doses. Nevertheless, even

Table 1. Annual and cumulative organ equivalent doses from photons >250 keV, which result in a median value of probability of causation of 50%[a] for a white male worker exposed from the ages of 21–50 y and diagnosed at age 60 y.[b]

Cancer type	Annual equivalent dose (mSv)[c]	Cumulative equivalent dose (Sv)[c]
Bone	130	3.9
Kidney	110	3.4
Lung (nonsmoker)	90	2.7
Lung (15–30 pack-years)	200	6.0
Multiple myeloma/Non-Hodgkins' lymphoma	500	15
Gall bladder/bile duct	170	5.1
Brain	470	14
Breast	90	2.7
Colon	110	3.3
Esophagus	150	4.5
Liver	42	1.3
Pancreas	490	15
Pharynx/salivary gland	370	11
Small intestine	170	5.1
Stomach	220	6.6
Thyroid	76	2.3
Urinary bladder	150	4.5
Leukemia[d]	22	0.66
Prostate	450	20
Skin (basal cell)	100	3.1
Skin (melanoma)	100	3.1
Skin (squamous cell)[e]	—	—
Breast[f]	130	3.8
Ovary[f]	150	4.3

[a] Calculated with the NIOSH-IREP program, using 2,000 iterations, and setting the random seed equal to 99.
[b] Values are specific to the stated hypothetical scenario only; they do not apply to any other scenario; they should not be used for purposes of screening.
[c] Doses are rounded to two significant figures.
[d] Excludes chronic lymphocytic leukemia.
[e] No value provided at median.
[f] Female worker, other conditions as stated.

allowing for a factor of five increase in cumulative dose (i.e., 2 Sv), leukemia remains the only cancer that meets the legal standard of causation.

Concurrently, it must be recognized that (1) the PC has an associated uncertainty distribution, due to uncertainty in both the assigned dose and the radiation risk factor (the latter is usually dominant) (Kocher et al. 2008), and (2) the NIOSH dose reconstruction process requires that the compensation decision be based on the 99th percentile of the distribution of the PC value (U.S. DHHS 2002b). Tabulated in Table 2 are the minimum annual organ equivalent doses that will produce a median PC value of at least 50% at the 99th percent credibility level of the distribution of the PC for the same scenario used in Table 1. The doses in Table 2 are significantly lower than the corresponding doses in Table 1, by a factor of two for leukemia, and a factor of four or more for the others.

Note that the values of annual organ equivalent dose in Table 2 are all in excess of 10 mSv, except for cancers of the liver and thyroid. Thus, exposures that could only contribute annual equivalent doses on the order of 0.1 mSv or less are usually insignificant for compensation purposes, depending on the type of cancer, and the other parameters that affect the PC (age, latency, etc.). Minor sources of exposure that contribute a small fraction of a mSv y^{-1}, such as eating wild nuts and berries that may have caused workers to incorporate radioactive materials from environmental releases, are insignificant in terms of their impacts on whether compensation is warranted. Granted, continued research can quantify such exposures and include them in the dose reconstruction, but only at the cost of delays in processing claims, and any compensation program must weigh scientific completeness against timely decision-making. In practice, such exposures might need to be considered only for those few cases where the reconstructed dose results in a PC (at the 99th percentile) of 45% or more, but less than 50%.

Because of random statistical fluctuations in the Monte Carlo process that the IREP program uses to calculate the uncertainty in the PC (Kocher et al. 2008), the 99th percentile credibility level of the PC can sometimes fluctuate about 50%. In these rare instances, more detailed exposure analyses are required to further refine both the assigned median value and the uncertainty of the reconstructed dose (NIOSH 2006b).

Table 2. Annual and cumulative organ equivalent doses from photons >250 keV, which result in a probability of causation of 50% at the 99% credibility limit[a] for a white male worker exposed from the ages of 21–50 y and diagnosed at age 60 y.[b]

Cancer type	Annual equivalent dose (mSv)[c]	Cumulative equivalent dose (Sv)[c]
Bone	30	0.90
Kidney	29	0.87
Lung (nonsmoker)	15	0.45
Lung (15–30 pack-years)	49	1.5
Multiple myeloma/Non-Hodgkins' lymphoma	45	1.4
Gall bladder/bile duct	14	0.42
Brain	62	1.9
Breast	23	0.69
Colon	25	0.75
Esophagus	28	0.84
Liver	7	0.21
Pancreas	64	1.9
Pharynx/salivary gland	62	1.9
Small intestine	36	1.1
Stomach	20	0.60
Thyroid	8	0.24
Urinary bladder	30	0.90
Leukemia[d]	11	0.33
Prostate	38	1.1
Skin (basal cell)	14	0.42
Skin (melanoma)	14	0.42
Skin (squamous cell)	140	4.2
Breast[e]	14	0.42
Ovary[e]	26	0.78

[a] Calculated with the NIOSH-IREP program, using 2,000 iterations, and setting the random seed equal to 99.
[b] Values are specific to the stated hypothetical scenario only; they do not apply to any other scenario; they should not be used for purposes of screening.
[c] Doses are rounded to two significant figures.
[d] Excludes chronic lymphocytic leukemia.
[e] Female worker, other conditions as stated.

A common misconception among claimants and other stakeholders is that the decision criterion is linearly related to dose. Thus, a claimant who is denied compensation at a PC of 40% (99th percentile) may erroneously conclude that he or she would have been compensated if the assigned dose were increased by 25%, which is expected to produce a PC of 50% (99th percentile), but this is not the case. In reality, the PC is a sigmoid function of dose, and the 99th percentile of PC is highly nonlinear in dose, in large part because the uncertainty in the risk factor is the primary contributor to the uncertainty in the PC. Merwin et al. (2008b) provide more detail and additional examples of the dependence of the PC on dose, uncertainties in the dose estimate and the risk coefficients, and the effect of basing compensation decisions on the 99th percentile credibility level of the PC distribution.

The efficiency process of assigning a maximum credible dose to a likely non-compensable case (Merwin et al. 2008a) has led to an unintended effect on claimants' perception of the dose reconstruction process. Typically the maximum dose estimate exceeds a realistic estimate of the actual dose by an order of magnitude or more, and could result in an estimated PC of say, 40% (99th percentile) for a given case. If the energy employee developed a second cancer in the interval between the original submittal of the case and the completion of the dose reconstruction, the case will be returned by DOL for re-work. Frequently, the PC for multiple cancers based on the previously assigned maximum dose will be greater than 50%, but because a case cannot be compensated with a maximum dose estimate, a more realistic dose estimate must be generated (U.S. DHHS 2002a). The revised dose reconstruction will usually report a dose and PC value that are much lower than the original ones, understandably leading to claimant dissatisfaction with the process.

"Best science" vs. "adequate science"

It should be noted that the concept of "claimant favorability" implies that a false positive (i.e., a finding for compensability that is in fact erroneous) is acceptable, while a false negative (i.e., a denial of compensation that is in fact deserved) is not. The overall goal of the dose reconstruction process under EEOICPA is to make the correct compensation decision, with false positives being acceptable. Therefore, the focus of the scientific basis of dose reconstruction needs to be on assessing and assigning all doses for a case that could produce a

positive finding. As noted above, that value of dose is dependent on the type of cancer, the age at first exposure, the latent period, and other factors. The dose reconstruction program also requires timely decision-making, and so the time spent in characterizing all possible, or even all probable, sources of exposure must be balanced against the magnitude of the potential doses from each source. As with many other phenomena, 90% of the effort could be expended on the last 10% of the dose. Consequently, the entire dose reconstruction process has been tailored to be adequate, in the sense of being sufficiently detailed to ensure that no sources of exposure not addressed could possibly move a case from below 50% PC at the 99th percentile to compensability (i.e., to above 50%).

COMMENTARY AND CONCLUSION

A tremendous amount of data has been collected and scientifically analyzed to develop a robust system of dose reconstruction methodologies for EEOICPA cases. The NIOSH dose reconstruction process requires decisions to be favorable to the claimant when there is a choice of options that are not determined by available data (U.S. DHHS 2002a). Consequently, the resulting dose reconstructions are well suited to their purpose of informing compensation decisions, but not appropriate for epidemiolological studies or other applications. Through the site profiles, the history of exposure conditions and radiation protection practices in the nuclear weapons complex has been compiled and evaluated. It is clear that the intent of dose reconstruction under EEOICPA, namely to provide scientifically-based but claimant-favorable estimates of radiation doses so as to compensate workers in the nuclear weapons complex who developed cancer as a result of their radiation exposures incurred, has been accomplished.

Acknowledgments—The work described in this paper was supported by the National Institute for Occupational Safety and Health under contract no. 200-2002-0593.

Disclaimer: The findings and conclusions in this paper have not been formally disseminated by the National Institute for Occupational Safety and Health and should not be construed to represent any agency determination or policy.

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