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EXTERNAL DOSE RECONSTRUCTION IMPLEMENTATION GUIDELINE

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Preface

The purpose of this guide is to provide basic information on the methods employed in reconstructing doses under the Energy Employees Occupational Illness Compensation Program Act of 2000. The intent of this guide is to assist a qualified health physicist in determining annual organ dose from exposure to various sources of external radiation. Because not all possible exposure scenarios can be foreseen, this guide does not provide step by step instructions for how the dose reconstruction should be performed. It is recognized there will be situations for which the methods outlined in this guide result in underestimates or overestimates of a claimants actual dose. In these cases, care must be exercised that the doses are conservative (claimant friendly) but reasonable for the claimant's exposure scenario.

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Record of Issue/Revisions

Issue Authorization Date	Effective Date	Revision No.	Description
May 2002	May 2002	0	External Dose Reconstruction Implementation Guideline
August 2002	August 2002	1	Updated Photon Dose Conversion Factors (Appendix B) to be consistent with IREP version 5.2. The intermediate energy photons cutoff changed from 200 keV to 250 keV. Updated Occupational Medical Dose section (2.1.3) to include calculation of dose from x-ray machine parameters.

1.0 INTRODUCTION

The purpose of this section is to provide guidance on the components, standards, and methods of external radiation dose reconstruction for probability of causation calculations in support of the Energy Employees Occupational Illness Compensation Program Act (EEOICPA). External radiation dose results from exposure to a radiation source that is outside of the body. The photon or particle radiations travel through the air and are absorbed in a tissue of the body.

1.1 Dose Reconstruction Requirement

The first step in the photon dose reconstruction is to determine whether there was any potential for external radiation exposure at the facility. At most Department of Energy (DOE) facilities and Atomic Weapons Employers (AWE) there is a potential for radiation exposure. When no radioactive material was processed or stored, an external dose reconstruction is not necessary. The three groups of workers who require dose reconstruction are: 1) workers who were not monitored for radiation exposure; 2) workers who were monitored inadequately for radiation exposure; and, 3) workers whose monitoring records are incomplete or missing (42CFR82.3(a) 2002).

1.1.1 Adequately Monitored

In general, external monitoring data collected since the implementation of 10 CFR Part 835 could be considered adequately monitored. When a claimant has been monitored adequately using either film badge dosimetry or thermoluminescent dosimetry (TLD) in accordance with the Department of Energy Laboratory Accreditation Program (DOELAP), these data shall be used to compute the annual dose for the claimant. The associated uncertainty should be assumed to be normally distributed and should be obtained from the site dosimetry office.

1.1.2 Not Monitored

Many of the Atomic Weapons Employer (AWE) workers were not individually monitored for radiation exposure. At some facilities, radiation surveys were conducted and this data, in conjunction with frequency of exposure, should be used to estimate the annual dose. When no radiation monitoring data is available for a facility, scientifically reasonable estimates of exposure should be developed based on the source term or quantity of radioactive material handled at the facility.

1.1.3 Monitored Inadequately

At some facilities, only a small sample of the work force was monitored to ensure compliance with radiation exposure limits. As an example, although construction workers were often not monitored, it may be possible in some instances to use workers who received similar exposures, such as radiological control technicians who monitored the work activities, to estimate external dose. For workers at these sites, the highest recorded value for similar work group should be assigned to the unmonitored worker.

In addition to incomplete monitoring practices, most early workers at DOE facilities were monitored inadequately compared to modern standards. In most instances, the missed dose alone can exceed 500 mrem/year. At many facilities, routine monitoring for neutron exposures was not initiated until the late 1950s. In general, monitoring data prior to 1960 must be evaluated cautiously due to technological shortcomings and because monitoring programs were designed to ensure compliance with a 12 rem/year exposure limit compared to the 5 rem/year current standard. For these workers and others with uncertain dose information, an evaluation of their dosimetry (or monitoring) data in combination with estimates for missed dose, occupational medical exposures, and environmental on-site dose should be used to determine the total annual external dose.

1.1.4 Monitoring records incomplete or missing

When monitoring records are incomplete or missing, the monitoring data prior to and after the missing data can be used to interpolate the missing data. When only post monitoring data is available, extrapolation should be used with caution, accounting for engineering administrative changes that might have been instituted which reduced exposures. In addition, co-worker data can be used to fill in missing or incomplete records.

1.2 External Radiation Exposures

The absorbed dose is to be calculated for the organ where the primary cancer exists. Appendix A lists the cancer types and the organ of interest used in the NIOSH - Interactive RadioEpidemiological Program (IREP) to calculate the probability of causation for an individual worker. For external radiation, there are three types of exposure; photon, neutron, and electron. Photon exposures are divided into three energy categories (< 30 keV, 30-250 keV, and >250 keV). Neutrons are divided into 5 energy categories (< 10 keV, 10-100 keV, 100-2000 keV, 2-20 MeV, and >20 MeV). While there are two electron categories in IREP, only the > 14 keV is considered to be a source of external radiation. Electrons below 14 keV do not have sufficient energy to penetrate the epidermal layer of the skin and, therefore, are not considered an external radiation hazard. Typically, external electrons are primarily of interest in skin cancer claims, however, depending on the beta particle energy the dose can be significant for the development of breast and testicular cancer as well.

1.2.1 Photon exposures

The four basic components of photon exposures are the individual's radiation monitoring data from dosimeters (D_D), the unrecorded or unmeasured dose commonly referred to as the missed dose (D_M), the occupational medical dose from medical monitoring x-rays (D_{OM}), and the environment dose primarily from stack emissions (D_E). The sum of these doses in each calendar year comprises a worker's annual occupational photon dose (D_γ).

$$D_g = D_D + D_M + D_{OM} + D_E$$

1.2.1.1 Dosimeter Dose (D_D)

Most radiation workers have been routinely monitored for exposure to radiation to ensure compliance with health and safety standards. External radiation monitoring was typically conducted on an individual basis using pocket ionization chambers, film badges, and thermoluminescent dosimeters.

1.2.1.2 Missed Dose (D_M)

Missed Dose is the unrecorded or unmeasured external photon dose that is a result of relatively high detection limits in early years of radiation monitoring combined with short-monitoring durations or a high dosimeter exchange frequency. In some instances, the missed dose is not the result of a technological limitation, but results from a recording practice, which considered some doses *de minimus*, which resulted in positive readings below some threshold being recorded as zero. Missed dose is particularly problematic in early years of radiation monitoring. During this time interval, missed dose was considered relatively unimportant since the annual occupational limits for radiation exposure were quite high (12 rem/year). As annual exposure limits were reduced, and monitoring technology improved, the magnitude of the missed dose has significantly decreased such that when quarterly TLD monitoring was implemented the missed dose is generally less than 40 mrem/year.

1.2.1.3 Occupational Medical Dose (D_{OM})

In early years, the latent effects of radiation exposure were not well understood, and short-term tolerance dose limits were believed to be protective. With improved technology came new screening techniques such as photofluorography. Medical personnel used these new techniques to screen and diagnose patients for diseases such as tuberculosis. In addition, these screening techniques were used to monitor for excess exposure to heavy metals such as uranium. According to Parker (1947), the entrance dose in photofluorography was about 1 R. Cardarelli et al (2001) noted that the bone marrow dose from photofluorography (≈ 800 mrad) was nearly two orders of magnitude greater than that of conventional diagnostic x-rays (≈ 10 mrad). At some facilities, photofluorography or diagnostic x-rays were required as part of the routine medical monitoring program. Since these examinations were required for employment, they are considered part of the occupational radiation exposure under EEOICPA. Primarily these exposures will be in either the < 30 keV or the 30-250 keV energy group. It should be noted that only medical exposures that were required as a condition of employment are included in the occupational medical dose. Diagnostic and therapeutic procedures not required for employment are not included.

1.2.1.4 Environmental Dose (D_E)

Typically, energy employees who were not categorized as radiation workers were not monitored using personal dosimeters, however, the work environment for these employees was often routinely monitored using area dosimeters or periodically monitored using survey instrumentation to measure the "background" environmental radiation levels. At many of these facilities, routine monitoring stations have recorded the average photon dose in a general area or at the plant boundaries. At several DOE facilities, radioactive emissions from plant stacks have been known to significantly increase the "background" radiation levels on the plant site. In general the dose from increased background is rather low.

The environmental dose at each facility should be evaluated. For example, during some operations such as the "green" runs at Hanford and INEEL, the release of fission products to the atmosphere significantly increased the ambient radiation levels well above natural background. At Hanford for instance, the environmental dose in May 1947 was reported as 21.7 mrep/month and 68.2 mrep/month for the 200-W and 200-E area respectively. The monthly environment gamma dose at the 100-B area was 12.4 mrep/month, which is near natural background levels. With prevailing winds from the west to east, clearly the downwind areas had significantly elevated radiation exposures. Since most routine emissions and even many of the non-routine emissions would not result in exposures exceeding the annual occupational limits, some workers at the facility, including security and construction workers, were not monitored for this exposure.

1.2.2 Neutron Exposures

The two basic components of the neutron dose are the individual's monitored dose from dosimeters (D_D) and the unrecorded or unmeasured dose commonly referred to as the missed dose (D_M). Since neutron exposures from man-made sources do not exist in the environment, nor are neutrons used in diagnostic or medical procedures, the later two categories are not included in the external radiation dose reconstruction.

$$D_N = D_D + D_M$$

1.2.2.1 Neutron Dosimeters (D_D)

Since the beginning of nuclear operations, neutrons have been monitored in the work place through radiation surveys, typically using either moderated boron tri-fluoride (BF_3) detectors or tissue equivalent proportional counters. Individual neutron exposures have typically been measured and recorded using specially designed pocket ionization chambers, nuclear track emulsion type A (NTA) film, and thermoluminescent dosimeters (TLD).

1.2.2.2 Neutron Missed Dose (D_M)

Neutron monitoring was not fully implemented at some sites until the late 1950s. Early use of NTA film resulted in relatively large missed doses for neutron exposure. For example, at Hanford, neutron film was read on a weekly basis with a stated limit of detection (LOD) of 90 mrem. At some sites, due to the difficulty in reading the film, many monitored workers' films were not read unless the photon dosimeter exceeded a certain threshold. This administrative practice has also resulted in some significant missed dose. This dose must be evaluated and added to the overall neutron dose.

1.2.3 Electron (Beta Particle) Exposures

Generally, external electron exposures are only important for surface tissue such as skin. Thus, for skin cancer, a dose reconstruction from exposure to electrons is required. The exposure to skin can originate from either a strong unshielded electron source such as Sr-

90 or uranium daughters, or from skin contamination with beta/gamma emitters. The other two organs for which external electron exposure from high energy electrons (> 1 MeV) might be significant are the testes and the breast. For breast and testicular cancer, an evaluation of the maximum electron energy exposure should be conducted. Generally, if the electron energy is less than 1 MeV, the dose conversion factor (DCF) will be zero since the electron does not have sufficient energy to penetrate the outer layer of skin (ICRP 74, 1996.) In these cases a dose reconstruction is not necessary. As with neutron doses, there are typically only two components of the electron dose, dosimeter dose and missed dose. Electrons have not typically been used for diagnostic occupational medical monitoring. An electron environmental dose is also usually not applicable since immersion in a cloud would generally have been monitored. While occupational medical and environmental doses are not included in electron dose reconstruction, skin contamination from beta-gamma emitters poses a unique exposure scenario that should be included in skin cancer cases. The general form of the electron dose equation is as follows:

$$D_E = D_D + D_M + D_S$$

1.2.3.1 Dosimeter Dose (D_D)

In early years, beta exposures were monitored using an open window of a film dosimeter. In the mid 1970s TLDs replaced the film badges at most facilities. One major advantage of the TLD is that the detector is similar to tissue and a shallow dose could be measured more accurately.

1.2.3.2 Missed Dose (D_M)

As with most dosimeters, there was a limit of detection that has resulted in some missed dose. In addition to readings below a limit of detection, many early monitoring programs measured but did not record the open window dose in the official dose of record for the individual.

1.2.3.3 Skin Contamination (D_S)

While skin contamination can result in some deep dose from photons, the shallow dose from the electrons is usually several orders of magnitude greater and should be included in dose reconstruction for skin cancer. Data such as isotope, and quantity of activity from skin contamination incidents have typically been reported in a claimant's radiological file.

1.3 Dose Reconstruction - Hierarchy of Data

Generally, individual dosimeter data should be used whenever possible and given precedence over personal monitors, survey data or source term data. In some instances, dosimeters were not worn or, in the case of neutrons, the NTA film limit of detection (LOD) was relatively high compared to the pocket ionization chamber. In these circumstances, the personal monitor can be used, however caution should be employed to ensure the dose is not a false positive or the sum of the personal monitors exceeds the LOD of the personal dosimeter. Table 1.1 outlines the general hierarchy of data sources that should be employed for dose reconstruction under EEOICPA.

Table 1.1 Hierarchy of Data Sources for Dose Reconstruction

Hierarchy	Data Source	Examples
1	Personal Dosimeter	Film Badge, TLD
2	Personal Monitors	Pocket Ionization Chambers
3	Co-Worker Data	Film Badge, TLD, Pocket Ionization Chambers, etc.
4	Area Monitoring	Work Place Radiation Surveys, Ambient Air room monitors, duration of exposure
5	Source Term	Source strength, distance from source, duration of exposure, and shielding information
6	Radiation Control Limits	Generally, workplace posting has been required when the dose rate exceeded 0.025 mSv/hr.

1.4 Initial Dose Assessment

In order to achieve the greatest efficiency in conducting external dose reconstruction, a health physicist should review the case to arrive at a rough estimate of exposure to determine if the case falls into either a very low or very high potential exposure category. External cumulative doses across numerous facilities have been observed to follow a log normal distribution with some very high exposures and some very low exposures. In some instances, particularly with short exposure durations, reasonable and conservative upper dose estimates can be developed based on relatively little data. Likewise, based on an initial review of exposure records, a health physicist should be able to identify workers with likely high doses and may only need to conduct a partial dose assessment to definitively place a worker's exposure into a high exposure. An initial dose assessment can greatly facilitate the throughput of dose reconstructions by conservatively overestimating a low dose exposure that would likely not result in a high probability of causation and underestimating a high dose exposure that would result in a high probability of causation.

1.4.1 Estimated Low Dose

This approach is most appropriate for short duration non-radiological workers who might have been exposed to low levels of environmental radiation. For example, an unmonitored clerical worker at a facility once walked through a radiological control area to deliver a message. The duration in the area was less than one hour and the maximum allowable dose rate in the area was 2.5 mrem/hr. Instead of using co-worker data, or actual survey data, the worker's dose can be estimated to be a maximum of 2.5 mrem.

1.4.2 Estimated High Dose

Some workers have exceeded occupational limits through radiological accidents or incidents. In order to expedite their claims, a partial dose reconstruction should be conducted, provided, the outcome results in a high probability of causation. In cases where the probability of causation is not high, a more detailed dose reconstruction must be conducted.

1.5 Conversion to Organ Dose

For external dose reconstruction under EEOICPA, the organ or tissue which developed the cancer is the organ of interest. Appendix A lists the ICD9 code and the corresponding organ for which the external dose should be calculated. Generally, only the 3-digit code is sufficient, however for skin cancer, the suffix designator which identifies the location on the body is needed to properly calculate the external dose for shallow exposure to the skin.

Typically, film badge and TLDs were worn on the upper front torso of the body. Depending on the monitoring era, these devices were calibrated to measure either 1) exposure, 2) the ambient dose equivalent, or 3) the penetrating dose at 10 mm using a standard phantom. The 10 mm penetrating dose is commonly referred to as the $H_p(10)$. For film badge dosimetry calibrated to exposure in roentgens (R), the conversion to ambient dose equivalent ($H^*(10)$) can be found in ICRU 43 (1988).

In turn, the ambient dose equivalent ($H^*(10)$) can be converted to air KERMA (K_a) using data from ICRP 74 (1996). The deep dose equivalent ($H_p(10)$) can also be converted to air KERMA using data from ICRP 74 (1996). Once the monitoring data has been converted to air KERMA, the organ dose can be calculated based upon the most likely exposure geometry for each of the IREP radiation types and associated energy intervals. The methodology describing these conversions is further discussed in section 4.

While these calculations are straightforward, the conversion of early film badge data to exposure energy is not. Unless corrections were made, the calibration of the early film badges using a radium or Cs-137 (high energy) gamma source resulted in an overestimation of the low energy exposure received by a worker. Thus, when appropriate, low energy exposures should be corrected for this overestimation.

1.6 Uncertainty

The general approach to uncertainty in external dose reconstruction is to treat each variable as a distribution and then employ Monte Carlo sampling of each of the distributions to determine the overall uncertainty of the annual dose estimate. In general, the uncertainty in the measured dosimeter dose and the occupational medical dose is assumed to follow a normal distribution, while the uncertainty in the missed dose and the environmental dose is assumed to follow a log normal distribution. The uncertainty in the conversion of exposure or deep dose equivalent to organ dose is assumed to follow a triangular distribution with the upper and lower bounds determined by the most and least favorable geometry and energy.

2.0 External Dose Reconstruction – Monitoring Data

When monitoring data are available, the three types of exposure are the photon dose, neutron dose, and electron exposures. As previously discussed, electron exposures are usually relevant for skin cancer but for high-energy electron exposure, testicular and breast doses should be evaluated.

2.1 Photon Dose

As discussed in the introduction, the four components of photon dose are the dosimeter dose (D_D), the missed dose (D_M), the occupational medical dose (D_{OM}), and environmental dose (D_E). The sum of these doses in each calendar year comprises a worker's annual occupational photon dose (D_γ).

2.1.1 Dosimeter Dose

2.1.1.1 Background

Since the beginning of nuclear weapons research and production, individual workers have been monitored using personal dosimeters at many facilities. Initial monitoring was conducted using film badges with various exchange frequencies. By the late 1970s most monitoring programs transitioned to TLDs. Through the years, technological developments have greatly improved the accuracy and sensitivity of the dosimeters.

2.1.1.2 Method

In general, the dosimeter dose is a summation of the individual dosimeter readings. As listed in Table 1.1, the following hierarchy should be used to determine an individual's dosimeter dose: personal dosimeter (film badge or TLD), pocket ionization chamber, group or co-worker dosimeter(s). Within the NIOSH-IREP probability of causation program, there are three photon energy bands; 1) below 30 keV, 2) 30 to 250 keV, and 3) above 250 keV. Therefore, some separation of the dose from each energy band is required.

At most of the larger facilities, multi-shielded film badges or multi-element TLDs have been used since the mid 1960s. Since only three energy bands are used in the probability of causation calculations, the differences between various filter doses can provide insight into the gross energy distribution at the facility. To the extent possible, these differences should be used to estimate the relative energy distributions in earlier years when only two element film badges were used. If individual energy distribution information is not available for two element film badges, the open window dose should be used as a claimant friendly estimate of the 30 to 250 kev dose. It is recognized that early film badges over-responded to low-energy photons, however corrections for this over-response are only recommended when scientific studies have been conducted and the exposure geometry and energy distribution are well known. For example, Fix et al (1994) estimated the two-element film dosimeter in the anterior posterior geometry at the Hanford facility over-responded by a factor of 1.7 to 100 keV photons.

When monitoring data do not indicate the relative energy distribution, the distribution can be estimated based upon either the site radionuclide inventory or the relative energy distribution which can be estimated for most facilities based upon a review of historical operations. An exception is the chemical separations areas in which the irradiation duration of the nuclear fuel might change the general observed ratios. When an estimate based on radionuclide inventory is conducted, some consideration should be given to the degree of Compton scattering that would contribute to the 30 to 250 MeV energy range.

For modern radiological monitoring programs, especially DOELAP accredited laboratories, the energy distribution should be well characterized for the processing of the TLDs and this information should be readily available from the site dosimetry program.

2.1.1.3 Uncertainty

2.1.1.3.1 Film Badge Uncertainty

A technical committee appointed by the National Academy of Sciences outlined three components (laboratory, radiological, and environmental) of uncertainty in personal dosimetry for film badge dosimetry used during atmospheric nuclear tests (NRC 1989). The uncertainty in the environmental component is discussed in section 2.1.3, and the radiological component is discussed in the exposure geometry section 4.4. Thus the laboratory uncertainty is the only source of uncertainty addressed in this section.

The uncertainty for film badge measurements is a function of the film type or packet used at the facility. The laboratory film badge uncertainty is relatively dependent on the exposure. Brodsky et al. (1965) extensively studied the accuracy of film badge dosimeters and concluded that under good laboratory conditions, the uncertainty can be as low as 10% to 15%, even at low doses. However, at many facilities, the uncertainty was much greater at low doses. Fundamentally, the absolute uncertainty at the 95% confidence should not be less than the limit of detection (LOD). For simplicity, the approach outlined by the National Research Council (1989) will be employed for dose reconstruction under EEOICPA. However, the additional uncertainty discussed for exposures below 200 mR will not be employed, since routine monitoring is generally more precise than large sampling events such as atmospheric test monitoring. The uncertainty associated with each dosimeter reading is assumed to be normally distributed, where the dosimeter reading is the mean and the upper 95% confidence dose is calculated by multiplying the uncertainty factor K(E) by each dosimeter reading using the following equations:

$$K(E) = 1 + 1.96\left[\frac{s\,(E)}{E}\right]$$

$$s\,(E) = \frac{s\ *}{D_{\infty} g} e^{gE}$$

where:

E = Exposure in roentgen

σ^* = Densitometer reading uncertainty typically 0.015 density units

D_{∞} = Saturation Density of the Film (Dupont 502 = 2.8)

γ = film sensitivity (Dupont 502 ≈ 0.25)

2.1.1.3.2 TLD Uncertainty

The uncertainty of thermoluminescent dosimeters is generally lower than film badge dosimeters, however the uncertainty is still somewhat dependent on the dose. Several biases can occur that, when combined, contribute to the random error. The fading of the dosimeter, especially in high temperature environments, results in a slight decrease in the measured dose. Conversely, the annealing process can result in residual artificial dose and spurious luminescence from contaminants, thereby overestimating the true dose. Hirning (1992) described the variance of TLD dosimetry as follows.

$$s_t^2 = s_n^2 + s_m^2 K_t^2$$

where:

s_t = Standard deviation of the total air KERMA
s_n = Standard deviation of the null readings
s_m = Relative standard deviation observed at high air KERMAs
K_t = Total air KERMA

Data for s_n and s_m should be readily available from most DOELAP accredited programs. This simple estimate is basically divided into two components with one part based on the limit of detection, which dominates in the low dose region, and the other based on a best estimate of overall dosimeter uncertainty (generally 10-15%.) As noted by Hirning (1992), a key assumption is that the two components are uncorrelated. This is believed to be appropriate since the variance in the low dose region would be dominated by measurement or counting statistics (i.e. total counts above background on a photo multiplier tube (PMT)). Conversely, in the upper dose region, the variance from counting statistics plays a rather insignificant role, however the uncertainty associated with the calibration, energy response of the dosimeter, and fading begin to dominate (i.e. $s_n << s_m K_t$). Generally the relative uncertainty associated with radiation monitoring has been less than 10-15% at relatively high dose levels. This uncertainty increases with decreasing dose to approximately 100% at the LOD.

2.1.1.3.3 Simplified Dosimetry Uncertainty

While site-specific data, if available, should be used, in many instances this data will not be known. Rather than initiate a research project for each claimant, prolonging the dose reconstruction and claims processing, the simple estimate of uncertainty is proposed based on the general equation provided by Hirning (1992).

The minimum detection level (*MDL*), sometimes called the critical limit (L_C), is generally defined as the point when the uncertainty of the reading at the 95% confidence level is ± 100%. The standard deviation at this level can be defined as:

$$s_{MDL} = \frac{L_C}{k} = \frac{L_C}{1.96}$$

Assuming that $s_{MDL} \gg s_n$ and that the standard deviation at the high dose level (s_m) is a constant relative percentage on the order of 10-20%, a simple estimate of uncertainty based on exposure level can be defined as:

$$s(E) = \sqrt{\left(\frac{L_C}{1.96}\right)^2 + \left(\frac{s^*}{100}(E)\right)^2}$$

where:

L_C = Critical Limit

s^* = Estimated percent standard error

E = Exposure or Dose

Figure 2.1 demonstrates reasonable agreement between the 95% Uncertainty Factors, K(E) using the film badge methodology and the simplified method over a range of exposures. The simplified method was calculated using a 30 mrem detection threshold and a standard error of 10%.

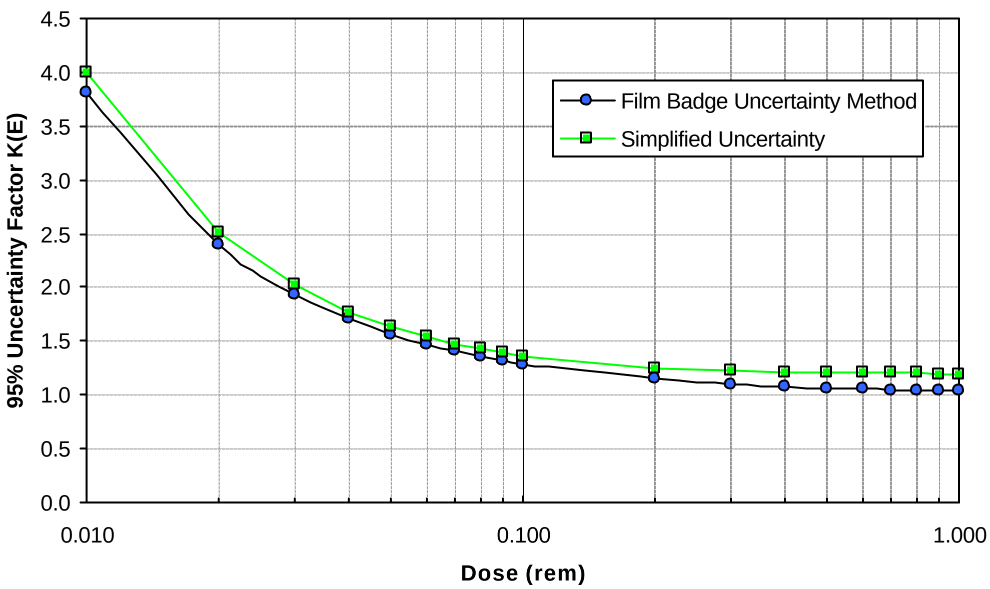

Figure 2.1 Comparison of film badge uncertainty to simplified uncertainty

2.1.1.3.4 Uncertainty Combination

The uncertainty from each film dosimeter should be calculated and the combined annual uncertainty should calculated using standard error propagation methodology (square root of the sum of the squares) as shown in the following equation.

$$s_D^2 = s_1^2 + s_2^2 + s_3^2 + ..s_i^2$$

where

s_D = Uncertainty of Annual dose

s_i = Uncertainty of a Single Dosimeter

2.1.1.4 Example

The following is an example of an individual's film dosimetry after summation and uncertainty sampling. The mean, 415 mrem, was equal to the sum of the 12 film badges and the upper 95% confidence was 513 mrem. Caution should be used when applying the normal distribution to ensure that the lower tail of the distribution is not negative.

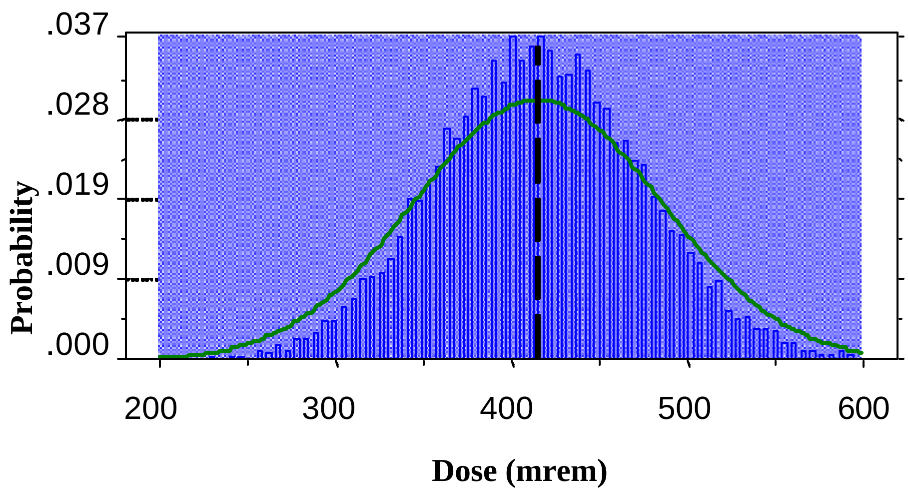

Figure 2.2 Uncertainty distribution of annual film badge exposures for a radiological worker monitored with Dupont 502 film with a mean of 415 mrem and a upper 95% confidence interval of 513 mrem.

Figure 2.2 compares the uncertainty between the simplified method (line) and the film badge uncertainty (bar) from example 2.1.1.4. The simplified method utilized a minimum detection level of 30 mrem and an estimated 20% standard error. As noted by the National Research Council subcommittee, laboratory uncertainty was never less than 1.2. As long as a standard error estimate is greater than 10%, the 95% Uncertainty Factor (K(E)) is also never less than 1.2.

2.1.2 Missed Dose

2.1.2.1 Background

In the scientific literature there are several different models that can be used to assess the missed dose (Strom 1988, Hornung and Reed 1990, Finklestein and Verma 2001). Recently, Taulbee, et al (2001) evaluated several models and concluded that the LOD/2 method resulted in a slightly positive bias (overestimate) of the true dose. While other missed dose methods might be more accurate on a large scale, for claimant friendly dose reconstruction, a bias which slightly overestimates the missed dose is acceptable.

2.1.2.2 Method

The National Research Council, in their evaluation of film badge dosimetry for compensation of atomic veterans, recommended the use of the Limit of Detection LOD/2 method. Since this scheme has been used in other compensation programs and has been shown to result in a slight positive bias, the recommended method for dose reconstruction related to EEOICPA is to assign a dose equal to the LOD divided by 2 for each dosimetry measurement (film badge, pocket ionization chamber or TLD) that is recorded as zero, below the limit of detection, or below a reported threshold. Each of these assigned doses is then summed for a given year as shown in Table 2.1. It should be noted that as the detection limit decreases and the monitoring interval increases, the missed dose becomes relatively insignificant compared to the dosimeter dose.

Table 2.1 Example of missed dose calculation

Year	Limit of Detection (mrem)	LOD/2 (mSv)	# of Zero Measurements	Estimated Missed Dose (mSv)
1956	30	15	25	375
1957	30	15	20	300
1958	20	10	10	100
1959	10	5	10	50
1960	10	5	5	25

2.1.2.3 Number of Zero Measurements is Unknown

When the number of zero measurements cannot be determined, the missed dose becomes more complicated. When only the annual dose is known, the number of zero doses should be estimated based on the dose level and the monthly, quarterly, or annual limits for that year, and the number of possible zero monitoring intervals. This would be the situation, for example, if an individual received a cumulative dose of 2140 mrem in a given year, at a facility that had a monthly monitoring frequency and the maximum permissible exposure limit was 1000 mrem per month. The minimum number of months in which this dose could have been received is 3. Therefore, the maximum number of missed dose months would be 9, and the minimum would be 0 since the dose could have been received evenly throughout the year. The central estimated number of months would be the median or 5, however the upper bound would be 9.

2.1.2.4 Uncertainty

As with all uncertainties within this document, the missed dose uncertainty will be combined with the measurement, energy and geometric uncertainties in a Monte Carlo sampling technique described in Section 1. Since the “true” missed dose is not known, there is some probability that the actual missed dose could be as great as the LOD times the number of zero measurements. Likewise, there is some probability that the “true” missed dose is actually zero. According to Strom (1988), and as verified by Taulbee et al (2001), the log normal distribution dominates in the low dose region. Therefore the log normal distribution should be used for the uncertainty distribution for missed dose. The central tendency should be calculated using the LOD/2 method, and the upper 95% dose should be the LOD times the number of zero measurements. In the scenario discussed above where the actual number of zero measurements was unknown, the central estimate would be calculated by multiplying the LOD/2 by 5, while the upper 95% estimate would be the LOD multiplied by 9.

2.1.2.5 Example

A simpler example using the data from 1957 in Table 2.1, the geometric mean is calculated to be 300 mrem, and the upper 95% confidence is 600 mrem.

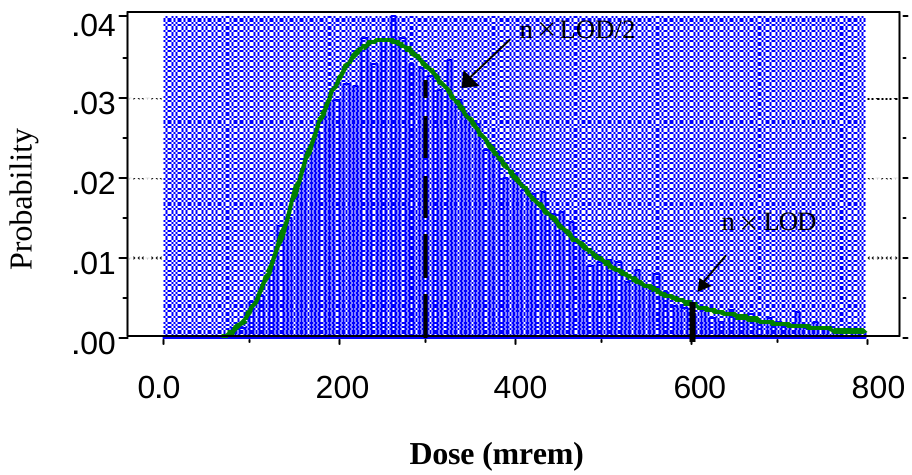

Figure 2.3 Log normal distribution of Missed Dose with a geometric mean of (n*(LOD/2) or 300 mrem and an upper 95% confidence interval of (n*LOD) or 600 mrem.

2.1.2.5 Multiple dosimetry monitoring

There are some individuals who, given their specific job function, might have worn multiple monitoring badges during a particular monitoring period. At some facilities these workers were classified as ROVER status. The central tendency of the missed dose for these individuals should be calculated using the same LOD/2 methodology, however, the number of zero measurements should be based on the number of routine monitoring intervals in a given year. The upper 95% dose of the log normal distribution should be based upon the total number of zero measurements multiplied by the LOD.

2.1.3 Occupational Medical Dose

2.1.3.1 Background

At many DOE facilities, physical examinations were required as a condition of employment. Some of these examinations included the use of diagnostic x-ray examinations. Because these were required, they are considered occupational dose for purposes of this program. Generally, these x-ray examinations result in a very small dose near the detection limit of film badge dosimetry. However in early years (< 1960) some facilities utilized photofluorgraphy equipment that could deliver substantial doses. As early as 1947, radiological control programs recognized that the use of this diagnostic procedure would not be appropriate for radiological workers, however some clinics continued to use the procedure (Parker, 1947). While the typical bone marrow dose from a standard chest x-ray is approximately 10 mrem, a standard photofluorography unit delivered a bone marrow dose of approximately 800 mrem (Cardarelli et al, 2001).

2.1.3.2 Method

The calculation of the Occupational Medical Dose is relatively simple. The most difficult component is determining the dose from the diagnostic procedure. The calculation of the dose should be converted to either ambient dose or deep dose equivalent. These diagnostic x-rays are generally low energy photons (<250 keV). If no information is known about the energy spectrum, they should be conservatively (claimant friendly) assumed to be in the 30-250 keV photon range. Medical records should contain the dates, type, and number of x-ray examinations. The general equation for calculating the Occupational Medical Dose is as follows:

$$D_{OE} = \sum_i nD_i$$

where

n = number of examinations in the calendar year

D_i = dose from diagnostic procedure i

When the dose from a diagnostic procedure are unknown, but the operating parameters of the x-ray machine are known (kVp, mA, and duration (msec), figure 2.4 below from NCRP 102 (1989) can be used to estimate the air kerma at 100 cm.

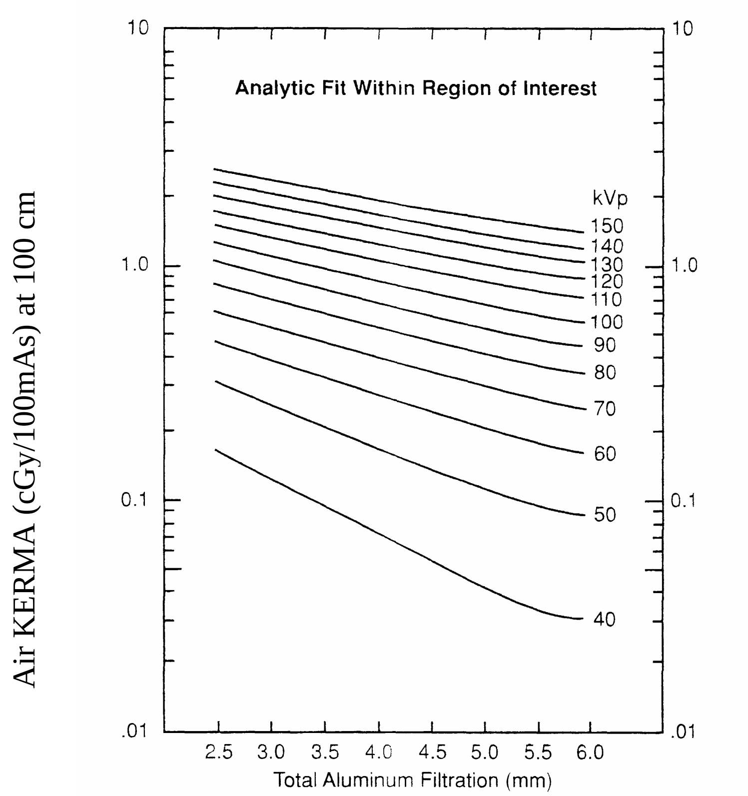

Figure 2.4 Air KERMA dose for 3 phase x-ray units. To obtain air KERMA for single phase units divide reading obtained from figure by a factor of 1.7. (NCRP 102, 1989)

2.1.3.3 Uncertainty

The uncertainty distribution about each diagnostic procedure is assumed to follow a normal distribution with D_{OE} being the mean dose. If known, the standard deviation of the procedure should be used. However this standard deviation is generally less than 20%.

2.1.3.4 Example

A worker received an occupational medical examination twice a year using a 3 phase x-ray machine operated at 100 kVp and 300 mA with 2.5 mm of aluminum for filtration, and the exposure duration was 32 msec. The distance between the x-ray machine and the claimant was 200 cm. The air kerma dose at 100 cm was calculated as shown below to be 1.15 mGy using data from figure 2.4, the 300 mA setting and an exposure duration of 32 msec.

$$D_i = \frac{1.2_{(100kVp, 2.5mm\ Al)}\ (cGy)}{100\ (mAs)} \times 300\ (mA) \times 0.032\ (s) = 0.115\ (cGy) @ 100\ \text{cm}$$

Applying a distance correction using basic health physics principles, the dose at 200 cm would be 0.029 cGy. Using an estimated uncertainty of 20%, the claimants annual air kerma dose would be 0.058 ± 0.012 cGy.

2.1.4 Environmental Dose

2.1.4.1 Background

Historically, radioactive stack emissions have substantially increased radiation levels around some facilities. Regulation of stack emissions has generally been designed to protect the general population or the environment with particular attention to dose rate levels near the site boundaries. Since the mid 1970s, stack emissions at most facilities have generally been low enough that these emissions have been negligible compared to occupational dose. However early stack emissions were not negligible compared to modern occupational limits and therefore will be considered, where appropriate, as part of the worker's exposure. Unlike the previous three modes of external exposure, which could be either chronic or acute, the environmental dose is always assumed to be chronic.

2.1.4.2 Method

At large DOE facilities, the stack releases were fairly well documented and ambient air dose rates were measured at monitoring stations throughout the facility. Since detailed employment history is generally available either through facility records or through the Computer Assisted Telephone Interview (CATI), this history can be used in conjunction with area measurements to estimate the dose contribution from plant emissions. Generally the dose will be very low (< 10 mrem/month), but during some time periods at certain facilities, the dose rate could be as high as a 100 mrem/month. A review of historical plant records should be conducted to make this dose determination on a case-by-case basis. Since energy distributions are not generally known from the dose rate measurements, the entire measurement is assumed to be high energy (>250 keV). The general equation to calculate the dose from plant emissions is given as:

$$D_E = n\dot{D}_m O_f$$

$n =$ number of months exposed
$\dot{D}_m =$ average monthly dose rate for year of interest
$O_f =$ Occupancy Factor (% of time on plant site)

2.1.4.3 Uncertainty

Ideally, the annual uncertainty should be calculated based on the standard deviation of monthly average dose rates or the standard deviation of all of the measurements. However in most instances this data will be difficult to obtain, thus some approximation of the uncertainty would be more reasonable. Based on the occupancy factor alone, it is highly unlikely that an environmental dose would ever exceed 500 mrem in a year. Thus this value can be used as a conservative (claimant friendly) upper bound (95%) with D_E being the geometric mean of a log normal distribution.

2.1.4.4 Example

Employment records indicate a claimant worked in the 200 Area at the Hanford facility from April – December in 1947. Environmental measurements indicate the average monthly dose rate was approximately 60 mR/month in the 200 Area. Since a worker would only be exposed while at the facility (generally: 40 hours/week out of the 168 hours/week) the occupancy factor O_f would be 0.238 for the 9 months of exposure. Therefore the environmental dose would be:

$$D_E = 9\ months \times 60 \frac{mR}{month} \times 0.238 = 129\ mR$$

The uncertainty distribution is depicted in Figure 2.5.

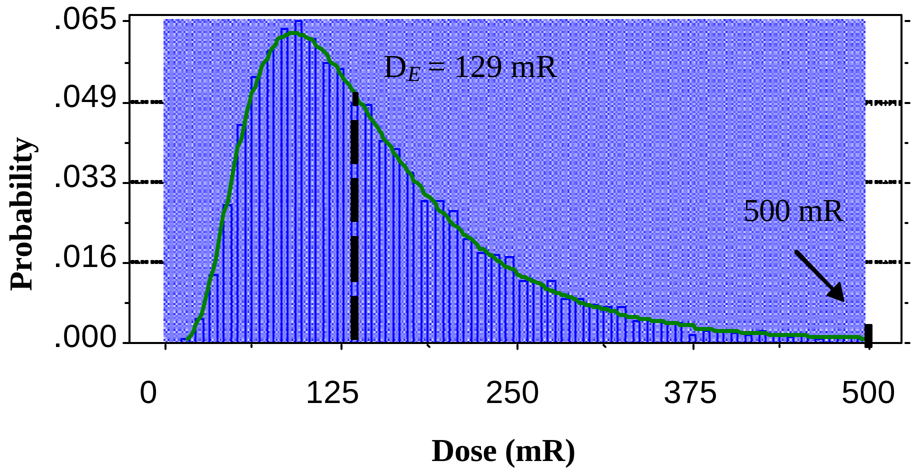

Figure 2.5 Lognormal Uncertainty Distribution of Environmental Dose with geometric mean of 129 mR and an upper 95% confidence interval of 500 mR.

2.2 Neutron Dose

Neutrons have not been used in occupational medicine and there are virtually no man-made environmental neutron exposures. A possible environmental exposure could occur during the operation of nuclear reactors, to produce plutonium, however by design special materials surrounded the reactor to reflect neutrons back into the core. When working in close proximity to a reactor, there is some neutron exposure, however these exposures are considered part of the dosimeter dose or the missed dose due to inadequate monitoring. As a result, for dose reconstruction under EEOICPA the dosimeter dose and the missed dose will be used to calculate an individual neutron dose.

As with photon exposures, the NIOSH-IREP program uses energy intervals to calculate the probability of causation from neutron exposures (Table 2.2). Neutrons, unlike photons, are high linear energy transfer (LET) radiation. As a result, the biological effectiveness of neutrons is believed to be greater per unit of absorbed dose than photons. To account for this, radiation weighting factors (w_R) are used to compute equivalent dose. In accordance with 42 CFR 82 (2002), ICRP 60 (1990) radiation weighting factors (w_R) will be used for dose reconstruction and reporting of dose.

Table 2.2: Neutron energy intervals and associated ICRP 60 weighting factor and some examples of exposures or facilities where the specific neutron energy might be encountered.

Neutron Energy (MeV)	ICRP 60 Radiation weighting factor, w_R	Typical Exposure Scenario
< 0.01	5	Low energy neutron exposures include thermal neutrons commonly found around nuclear reactors or moderated neutron sources. More prevalent around heavy water reactors.
0.01 – 0.10	10	Intermediate energy neutron exposures can also result from operation around nuclear reactors as high-energy neutrons are moderated to thermal energies.
0.10 – 2.00	20	Commonly called fission spectrum neutrons, this is the most typical energy range from operation of light water or graphite moderated reactors.
2.0 – 20.0	10	Reactions between alpha particles from materials such as plutonium or polonium and light materials such as beryllium can result in the production of neutrons. These reactions are commonly called (α,n) reactions. This neutron energy interval also includes 14 MeV neutrons from fusion reactions.
> 20.0	5	Exposures to neutrons greater than 20 MeV can result from work around accelerators.

2.2.1 Dosimeter Dose

2.2.1.1 Background

Prior to the early 1950s personal neutron monitoring was conducted using boron lined pocket ionization chambers at most facilities. In the early 1950s some facilities began using neutron track emulsion (NTA) film for measurements of fast neutrons, and in the 1960s, cadmium plates were used to distinguish between fast and slow neutrons. By the mid 1970s most facilities switched to TLDs, however several continued using NTA film.

The boron-lined pocket ionization chambers measured slow or thermal neutrons. Since the body easily thermalizes fast neutrons, some reflected neutrons would have been measured by these dosimeters. Unfortunately, these dosimeters were not calibrated for this effect; therefore there is a high degree of uncertainty when using these dosimeters to assess exposure to fast neutrons.

NTA film was far superior to the boron-lined pocket ionization chambers, however they suffered from an inability to accurately measure neutrons below about 500 keV (Griffith et al 1979). Due to the large variability associated with the boron lined pocket ionization chambers, an absorbing material such as cadmium or gadolinium was used to measure thermal neutrons. Thermal neutrons absorbed by the material emit low energy photons in a n,γ reaction. The additional darkening of the film from the low energy gamma rays of the n,γ reaction were then compared to the film behind a similar material with a low thermal neutron cross section. The difference was a relative measure of the thermal neutron exposure.

With the introduction of thermoluminescent dosimetry in the 1970s, neutron measurements improved with lower detection limits, however the relative uncertainty at higher doses remained generally about the same.

2.2.1.2 Method

As with photon monitoring, the dosimeter dose from neutrons is the summation of each of the dosimeters for a given year. Some analysis, however, is typically required to evaluate the neutron energy spectrum, the calibration and reported quantity, and the radiation quality factors used.

2.2.1.2.1 Neutron Energy

When no energy information can be found, the exposure scenarios discussed in Table 2.2 can be used to estimate the predominant energy. However, some reasonable assumptions are still required to estimate lower energy components, since neutron interactions with materials will decrease the energy.

2.2.1.2.2 Calibration and Reported Quantity

At some facilities, the calibration sources of the neutron dosimeters were changed thereby changing the response of the dosimeter. Some analysis is required to determine whether or not the radiological records were updated to reflect the change.

2.2.1.2.3 Quality Factor

In order to interpret site reported doses, some site-specific analysis of the quality factors used is required. Generally, since the 1950s, a quality factor of 10 has been applied to fast neutron exposures, however this has changed from 5 to 20 across facilities and time frames.

2.2.1.3 Uncertainty

The uncertainty associated with neutron monitoring is assumed to follow a normal distribution like the photon dosimeter dose. Several authors have reported general uncertainty to be about 20% to 30% (Oshino 1973, McDonald and Hadley (1985)). Several factors such as latent image fading, neutron spectrum energy, and reader repeatability affect the uncertainty of neutron dosimeter readings. Schimmerling and Sass (1968) reported latent image fading uncertainty to be 20% - 40%. Watson (1951) reported inter-reader variation due to difficulty in reading track information on NTA film to be approximately 24% at 63 mrem.

Schimmerling and Sass (1968) reported the standard deviation of two groups of irradiated dosimeters analyzed by commercial vendors from 1964 to 1966. Group A was irradiated on the first day of the badge wear period or cycle while group B was irradiated on the last day of the wear period just prior to reading. The group A badges generally displayed a systematic under-response due to latent image fading, while the group B badges displayed an over-response resulting in an overestimation of the irradiated dose. Within both of these groups, the uncertainty associated with reading the badges was provided. The reading uncertainty has been converted to the K(E) parameter in Figure 2.6.

McDonald and Hadley (1985) conducted one of the most comprehensive reviews of neutron dosimeter response uncertainty at 12 DOE sites. They evaluated the response of various neutron dosimeters which included the albedo TLDs and NTA film dosimeters. As noted previously, McDonald and Hadley (1985) reported the overall uncertainty was 10% to 25%. However, near the detection limit the uncertainty approached or exceeded 100%. Phase 1 of this study provided the most comprehensive review of routine monitoring at the DOE facilities. The relationship between uncertainty and dose is depicted in Figure 2.6, using the upper 95% dose limit methodology discussed in the photon section of this guide and the coefficient of variation data provided by McDonald and Hadley (1985).

Fix et al (1996) analyzed NTA film calibration data, which included reader uncertainty for the Hanford site from 1950 to 1961. Although these calibration doses were greater than 100 mrem, and all sources of uncertainty are not included, it is apparent that the observations reported by McDonald and Hadley in the 1980s should be reasonable approximations of the uncertainty over the monitoring time period from 1950-1990s. Note that this uncertainty factor does not account for any systematic bias that should be corrected for on a site-by-site basis as appropriate.

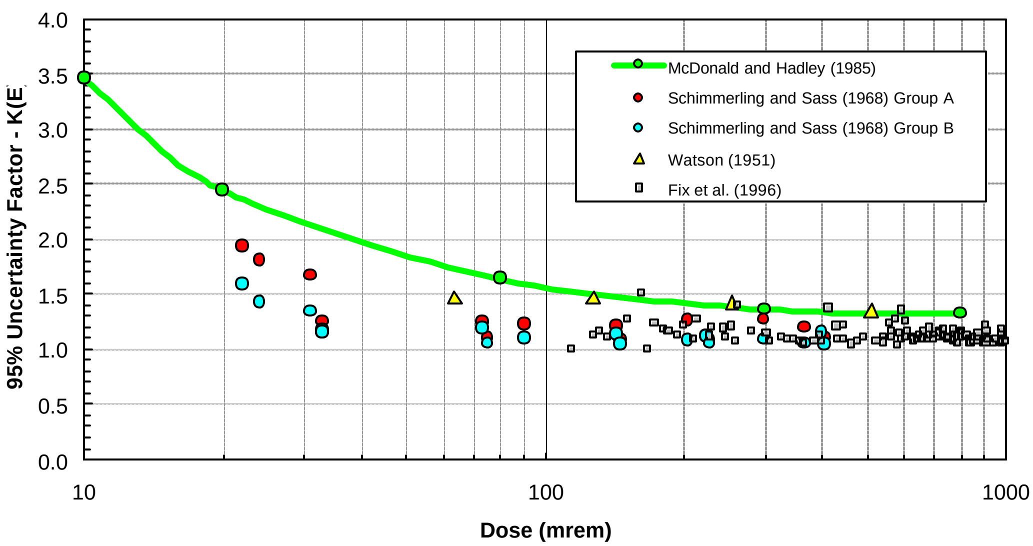

Figure 2.6 95% Uncertainty factor K(E) calculated from various data sets.

2.2.1.4 Example

Table 2.3 provides an example of neutron monitoring data collected from the Hanford site in 1961. The individual worked at the plutonium processing facility, which resulted in exposure to moderately high-energy neutrons around 4 MeV. A small fraction of these neutrons would be moderated through the plexiglass of the gloveboxes to intermediate energies and low or thermal energies. Heinzelmann and Nachtigall (1967) have described how the average energy of fast neutrons decreases with shield thickness.

The monitoring of thermal neutrons during this time period was conducted using a cadmium shielded film badge, thus the slow neutron measurements are of thermal neutrons only (0.025 ev). The threshold for fast neutrons using NTA film was approximately 500 keV. As a result, the dose due to neutrons between thermal energies (0.025 ev) and the NTA threshold of 500 keV has not been measured or reported. The dose from the intermediate energy neutrons should be treated as a missed dose.

Table 2.3 Neutron data for a worker at Hanford

Date Ending	Slow Neutron (mrem)	Fast Neutron (mrem)
1/13/1961	11	
1/27/61	11	49
2/10/61	8	
2/24/1961	8	
3/24/1961	8	
4/21/1961	7	
5/19/1961	11	
6/2/1961	8	
6/16/1961	6	107
6/30/1961	11	
7/14/1961	8	123
8/25/1961		96
9/22/1961	6	96
10/6/1961	11	115
11/3/1961	8	52
11/15/1961	11	
12/1/1961	8	84
12/15/1961		143
12/27/1961	11	

From Table 2.3 the slow neutron (Group I) dose is 152 mrem and the fast neutron (Group III) dose is 865 mrem. Using Figure 2.6, the uncertainty factor K(E) can be estimated to be approximately 1.52 and 1.28 respectively. This would result in the following distributions for Group I and Group III neutrons.

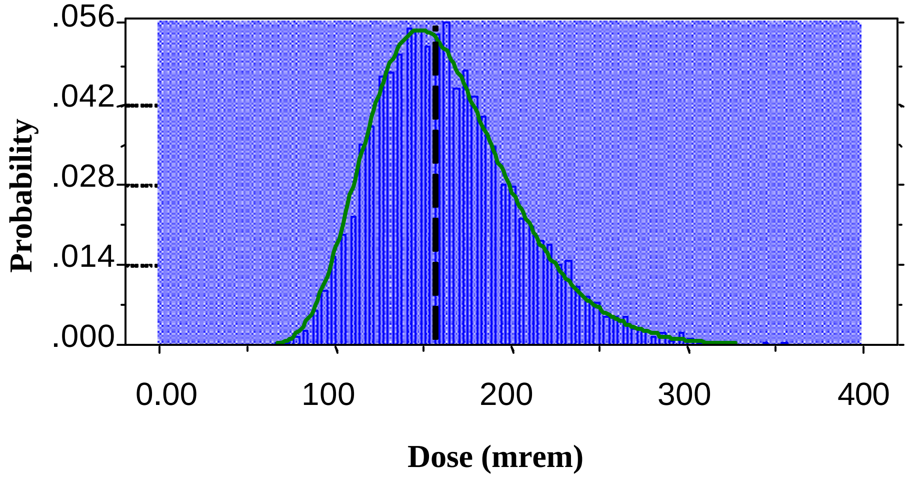

Figure 2.7 Group I neutron dose distribution with a mean of 152 mrem and an upper 95% confidence limit of 231 mrem.

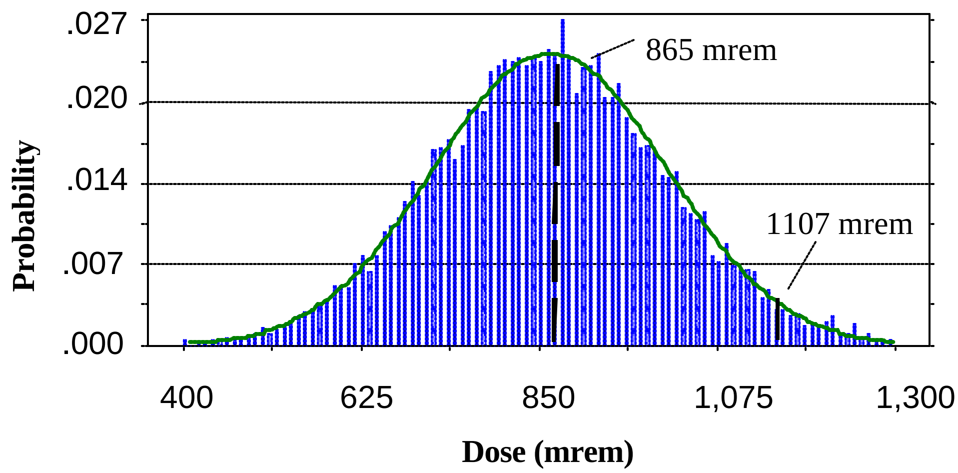

Figure 2.8 Group III neutron dose distribution with a mean of 865 mrem and an upper 95% confidence limit of 1107 mrem.

2.2.2 Missed Dose

2.2.2.1 Background

Neutron monitoring was not fully implemented, or was generally inadequate, until the late 1950s. By simple examination of the collective dose at the Hanford facility it is clear that there was virtually no recorded neutron dose before 1957, even though large-scale operations were ongoing since 1945 (Fix et al. 1996).

Watson (1959) discussed the neutron monitoring practices at the Hanford facility and reported the limit of detection/reporting limit for the neutron dosimeter in the early 1950s was 90 mrem. In some areas, these dosimeters were initially exchanged on a weekly basis, thus the maximum missed dose for a one-year period would be 4500 mrem, assuming a 50 week work year, or 90% or the current occupational limit of 5000 mrem. As a result, missed neutron doses have the potential to contribute significantly to the annual occupational dose, especially in early years of the DOE weapons complex.

At most facilities, neutron exposures were significantly smaller and were generally less than 20% of the photon exposures. Whether neutron exposures can be correlated with photon exposures remains an open question. Watson (1959) indicated that correlation between the neutron exposures and photon exposures at Hanford could not be quantified since badging cycles did not coincide and personnel were not required to wear their neutron badge at all times, as was the case for their photon badge.

There are two possible components of neutron-missed dose, 1) the missed dose from film badges recorded as zero, and 2) the potential for missed dose due to unmonitored neutron energies from early dosimetry methods.

2.2.2.2 Method

2.2.2.2.1 Monitoring Data - Below Limit of Detection

Generally neutron missed dose will be evaluated using the same method discussed for photons. The LOD/2 times the number of zero monitoring badges is the central estimate of a lognormal distribution and the upper 95% estimate is the LOD times the number of zero monitoring badges.

An exception to the method is needed for unreasonably high neutron missed doses. Generally the neutron dose is significantly less than the photon dose. Therefore when the neutron missed dose central estimate (nLOD/2) exceeds 75% of the photon dose (dosimeter dose + missed dose), the exposure should be treated as an unmonitored exposure and radiation survey data combined with stay times (frequency of exposure) should be used to estimate the missed dose. The reason for this deviation is that early monitoring of neutrons was sufficiently poor that the missed dose was virtually an unmonitored exposure. With accurate stay time information and numerous neutron measurements, a reasonable estimate of exposure can be derived for recorded exposures below the limit of detection.

2.2.2.2.2 Unmonitored Neutron Energy Interval

Some neutron monitoring programs were designed to measure either thermal neutrons or fast neutrons or both. Typically, the thermal neutrons were measured using the (n,γ) reaction with a material such as cadmium and the fast neutrons were measured using NTA film. However the capture cross-section for cadmium decreases from approximately 2500 barns (b) at thermal neutron energy (.025eV) to less than 1b at 24 keV. The response of NTA film is virtually negligible below 500 keV. Thus, relatively little attention has been given to energy intervals between thermal energies 0.025 eV (Group I) and the fast neutron NTA threshold of 500 keV (Group III).

Fundamentally, a worker cannot have an exposure to fast neutrons without an exposure to intermediate (Group II) or thermal neutrons (Group I), since the human body acts as a moderator. However, a worker can be exposed to only low energy neutrons if a moderator shielded the neutron source. Generally, Group II neutrons have gone unmonitored and unreported unless they have been accounted for in site specific algorithms. When site/facility specific neutron spectrums are known, respective doses in each missing group can be determined using the ratio of the dose within each energy interval. When no specific neutron spectrum information is known ***and*** both the thermal and fast components have been reported, the Group II dose can be estimated by interpolating the neutron fluence between the opposing groups. The conversion from dose to fluence is necessary since the dose conversion factor is different for thermal and fast neutrons. Once the interpolated fluence is known, the midpoint of the energy interval can be used to estimate the dose.

2.2.2.4 Uncertainty

The uncertainty associated with missed neutron dose should be evaluated as described in section 2.1.2 using the lognormal distribution.

2.2.2.5 Example – Limit of Detection

A worker was monitored for exposure to neutrons for six months. During this time, the monitoring exchange frequency was bi-weekly for the neutron exposures and the LOD was 40 mrem (thermal neutrons only). The worker had 10 out of 19 neutron badges recorded as zero and the total photon exposure over the monitoring period was 2085 mrem. The central neutron dose estimate is 200 mrem, which is less than 75% of the total gamma dose, therefore the LOD/2 method is considered valid. The upper 95% confidence interval would be 400 mrem as depicted in Figure 2.9.

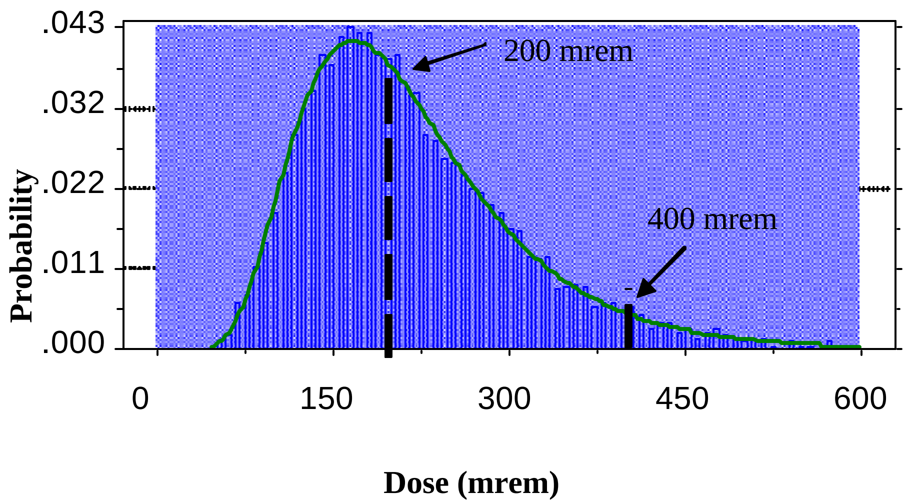

Figure 2.9 Missed Neutron Dose example, log normal distribution with geometric mean of 200 mrem and an upper 95% confidence interval of 400 mrem.

2.2.2.6 Example – Intermediate Energy Determination

From the dosimetry data listed in Table 2.3, the thermal neutron (Group I) dosimeter indicated an annual exposure of 152 mrem while the fast neutron dosimeter indicated an exposure of 865 mrem (Group III). Converting this ambient dose ($H^*(10)$) to fluence, the Group I fluence was 1.86×10^5 (n/cm^2s), and the Group III fluence was 2.34×10^4 (n/cm^2s). Interpolating between these, the Group II fluence would be 1.05×10^5 (n/cm^2s). Recalculating the ambient dose, the intermediate neutron (Group II) dose would be approximately 489 mrem .

2.3 Electron Exposures

In general, electron exposures or beta exposures are only significant for cases of skin cancer, however if high-energy (> 1.0 MeV) electrons are encountered, high exposures can be significant for breast and testicular cancer. Swinth et al (1986), in a review of beta exposures at DOE facilities, noted that the average beta energy was typically below 500 keV, however, at some facilities the average energy was near 1 MeV. For purposes of this guide, all discussions and examples are applicable to skin cancer only. Most workers were somewhat protected from electron exposures through the use of coveralls, gloves, or other anti contamination clothing. Low energy electrons usually do not have sufficient energy to penetrate outer clothing, however exposed skin on the hands and face can receive significant exposure.

For skin cancer, the dose should be estimated for the cancer site. If the cancer started on the hand, then the extremity dose would be more appropriate than the film badge worn on the lapel. Conversely, for skin cancer originating on the face, the lapel dosimeter would be more appropriate for determining the dose. Professional judgment should be used to determine which measurements are most appropriate for the skin cancer site.

2.3.1 Dosimeter Dose

2.3.1.1 Background

Currently, electron exposures are measured as a shallow dose (H'(0.07)) at a depth of 0.07 mm in tissue using a tissue equivalent TLD. Past DOE monitoring practices utilizing film do not provide a good conversion to shallow dose. The open window of the film dosimeters was typically calibrated using uranium, however at some facilities, an Sr-90/Y-90 source was used. These calibrations were typically in units of exposure (R).

2.3.1.2 Method

As with photon and neutron exposures, the dosimeter dose is the simple summation of each dosimeter for a given year. To properly determine the shallow dose, information about where the dosimeter was worn (i.e. inside or outside of clothing), electron energy spectra, and response of the film dosimeter is needed. Since extensive research would be required to properly convert each different dosimeter type to the current standard of shallow dose at 0.07 mm, the exposure is assumed to be equal to the shallow dose