Remediation Standards for Radionuclide Contamination in Connecticut

Connecticut Department of Environmental Protection
Division of Radiation
Bureau of Air Management

And

Connecticut Department of Public Health
Division of Environmental Epidemiology and Occupational Health

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Executive Summary

The Connecticut Departments of Environmental Protection (DEP) and Public Health (DPH) have developed a radiation remediation standard to be used for remediation purposes in all radiation contamination situations in Connecticut. The primary objective for developing such criteria is to protect human health and the environment from potential teratogenic and carcinogenic effects of ionizing radiation above background levels. It is based on the scientific consensus that radiation exposure should be kept as low as reasonably achievable.

DEP and DPH have reviewed the guidelines and standards developed by various national and international bodies, as well as other states and federal agencies. Based on a review of these standards and guidelines, DEP and DPH have selected the modeling in the National Academy of Sciences report, *Health Effects of Exposure to Low Levels of Ionizing Radiation, BEIR V.* This model is the most recent comprehensive analysis of health effects of exposure to low levels of ionizing radiation.

Connecticut DPH uses a risk range to set standards for cancer-causing contaminants found in air, water, and soils. The most protective level achievable is preferred, even when the contaminant naturally occurs at higher levels, or where measurement technology does not allow for detection of the contaminant at lower levels. The acceptable range of protection is one in a million excess cancer risk to one in ten thousand excess cancer risk.

Naturally occurring radiation has a high background in the environment, from soils, foods, and cosmic radiation. Thus, for radiation contamination, DEP and DPH have set the radiation remediation standard at the one in ten thousand excess cancer risk level. This results in a radiation remediation standard of 19 millirem/year total effective dose equivalent exposure above background levels, with the understanding that exposures must be kept As Low As Reasonably Achievable – the principle of ALARA.

Connecticut has derived a concept of ALARA that is applicable to the Connecticut standard of living and rate of inflation. The ALARA analysis that is required for radiation remediation in Connecticut uses a method that is: (1) unbiased, utilizing appropriate dose modeling to relate concentrations to dose; (2) usable as a planning tool for remediation; and (3) efficient, since the surveys conducted for other purposes could be used in the process.

The advantage of the ALARA approach outlined in this document is that it allows the user to estimate a concentration at which a remediation action will be cost-effective prior to starting a remediation and prior to planning the final status survey. Thus, it is a

useful planning tool that lets the user determine which remediation actions will be needed to meet the ALARA requirement.

1.0 Purpose and Background

The purpose of this document is to set forth the minimum criteria that should be used for the remediation of sites in Connecticut that are contaminated with radioactive materials, and to describe the rationale used for their development. The primary objective for developing such criteria is to protect human health and the environment from potential teratogenic and carcinogenic effects of ionizing radiation. It is based on the scientific consensus that radiation exposure should be kept as low as reasonably achievable.

The Connecticut Department of Environmental Protection (DEP) has established a remediation standard that would apply to the variety of radiation contamination situations that may occur in Connecticut in conjunction with the Connecticut Department of Public Health (DPH), the lead risk assessment agency in Connecticut. During the past decade such discussion on a standard has been done on a case-by-case basis.

For carcinogenic substances, DPH ordinarily uses a risk range of one in a million excess cancer cases (10^{-6}) to one in ten thousand (10^{-4}) excess cancer cases for incidence of cancer. This can be interpreted as: if an individual were to be exposed for a lifetime to a cancer-causing agent, there would not be more than one in a million chance (for 10^{-6}) to one in a ten thousand chance (for 10^{-4}) of developing cancer; or, if a million people were similarly exposed for a lifetime, no more than one (for 10^{-6}) to 100 (for 10^{-4}) cancers would be predicted to develop as a result of such exposure. DPH uses a risk level of 10^{-6} except where background levels of a substance are so high as to preclude its use, such as in the case of radiation or certain naturally-occurring metals in drinking water.

There are a number of criteria and standards which have been used by other states and federal agencies for various clearance purposes. DEP and DPH have examined the bases of and justification for these standards to provide information for the development of a Connecticut-specific standard. DEP is mandated by Connecticut General Statutes to use a standard at least as protective as the Nuclear Regulatory Commission (NRC). DEP and DPH have chosen to use the well-respected National Academy of Sciences report, *Health Effects of Exposure to Low Levels of Ionizing Radiation, BEIR V,* as the most appropriate basis for the Connecticut standard.

Consideration has also been given to the exposure that the public receives from naturally occurring ionizing radiation and other sources. Due to relatively high natural background, the guidance provided here is for decontamination to levels *above* the background in the specific situation that equate to an excess cancer risk of 10^{-4}. Determination of background levels for a site under consideration is to be conducted utilizing current technical methodology, as determined by the DEP.

DEP and DPH have consulted background documents and other sources from federal agencies, such as the Environmental Protection Agency (EPA), the Agency for Toxic Substances and Disease Registry (ATSDR), and the Nuclear Regulatory Commission (NRC); from national and international consensus bodies, such as the National Council on Radiation Protection and Measurements (NCRP), the International Commission on Radiation Protection (ICRP), and the National Academy of Sciences (NAS); and from other northeast states. DPH and DEP are aware of work in progress in reassessing the health effects of exposure to low levels of ionizing radiation, as set forth in the Biological Effects of Ionizing Radiation (BEIR VII) Phase I report. The present guidance does not provide independent modeling, but relies on the modeling in the BEIR V report.

Risk characterization will occur based on the scenario and radioactive materials involved, summing the risks from the particular radionuclides present. The Connecticut standard uses total effective dose equivalent (TEDE) for radiation dose, the sum of the external radiation dose and internal committed doses for inhalation or ingestion of contaminated materials. The risk assessment will apply to both residential and industrial exposures. Connecticut's standard is for unrestricted use of areas cleaned to the standard. Thus, if soils containing concentrations of radioactive material less than permitted by the remediation standard are removed and distributed as clean fill in any setting, including children's playgrounds, resultant doses will be at or below 19 mrem/yr TEDE.

1.1 Risk Assessment Principles

Risk assessment objectives differ from those of occupational radiation protection. In risk assessment, a broader range of human variability in populations may be expected compared with a healthy working population. This variability is seen in such factors as age ranges, health status, and modes of exposure. For instance, the overage marks of the critical group protected from excess risk of genetic or carcinogenic effects must take into account members of the general public, exposed over many years, and/or exposed through multiple pathways. These pathways could include exposure from external sources; ingestion of contaminated foodstuffs, soil, and water; inhalation; and dermal exposure. The CT standard is based on risk modeling using current understanding of radiation health effects, as derived from studies of persons exposed through atomic bombing, fallout from nuclear testing, accidents, diagnostic and therapeutic medical treatments, reference to animal studies and cell or tissue studies.

The dose to any particular person may differ due to the individual factors involved. These include the chemical and physical form of the radioactive material, individual body size, gender, and metabolism, and genetic factors. The risk from exposure during childhood is estimated to be about twice as large as the risk for adults, based on the BEIR V life table analyses (NAS, 1990).

There are apparent dose rate effects, in addition to effects due to the relative biological effectiveness (RBE) of the several types of radiation. In the derivation of the standard, a
Dose Rate Effectiveness Factor (DREF) has been applied. The average Single Best Estimate of the DREF was utilized, with the uncertainty in the calculation noted by presenting the range of DREFs from various animal studies utilizing 90% confidence limits. The standard presented here is an upper boundary condition, not a design criterion, since exposures to radiation should be kept as low as reasonably achievable (the ALARA principle).

1.2 Risk Assessment Assumptions.

The Connecticut DPH, Bureau of Community Health, Toxic Hazards Assessment Program conducts public health assessment and health risk assessments for a variety of contamination situations. The duration of exposure for location-specific scenarios, such as soil contamination, is thirty years. This value represents the national 90^{th} percentile for people living at one residence. Additionally, thirty years is the value used in the Connecticut DEP Remediation Standard Regulations (22a-133k through 22a-133k-3).

Connecticut DPH uses an upper bound risk estimate of 10^{-4} lifetime excess risk for development of cancer, the upper level of the risk range, when background levels are high relative to contamination. The modeling used by the National Academy of Sciences in BEIR V is based on a 100-year lifetime. The calculations for the Connecticut standard modified the BEIR V Model for the 30-year exposure hypothesized.

2.0 Derivation of the Standard based on BEIR V

2.1 BEIR V Modeling of Excess Cancer Risk

DEP and DPH utilize the data on excess cancer mortality presented in the BEIR V report to derive the standard. As shown in Table 2.1, for continuous lifetime exposure (100 years) to 100 mrem/year without a dose rate effectiveness factor (DREF) for nonleukemia, the BEIR V report lists the excess cancer mortality estimates as: Male, 520 fatalities (range: 410 - 980) per 100,000 exposed persons; Female, 600 fatalities (range: 500 - 930) for 100,000 persons exposed (NAS, 1990). The BEIR V report does not provide similar data for cancer incidence, except for breast cancer. DEP and DPH are utilizing the mortality data with a DREF for non-leukemia for this risk assessment.

TABLE 2.1 Excess Cancer Mortality Estimates and Their Statistical Uncertainty--Lifetime Risks per 100,000 Exposed Persons[a] *(NAS, 1990)*

	Male			**Female**		
	Total	**Nonleukemia[b]**	**Leukemia[c]**	**Total**	**Non-leukemia**	**Leukemia**
Continuous lifetime exposure to l mSv/y (100 mrem/yr)[d]	520	450	70	600	540	60
90 % confidence limits[d]	410 - 980	320 - 830	20 - 260	500 - 930	430 - 800	20 - 200

[a] Based on an equal dose to all organs and the BEIR V committee's preferred risk models—estimates rounded to nearest 10.

[b] Sum of respiratory, breast, digestive, and other cancers. [c] Estimates for leukemia contain an implicit dose rate reduction factor. [d] A dose rate effectiveness factor (DREF) has not been applied to the risk estimates for solid cancers.

The BEIR V Committee report noted that because of re-evaluation of estimates of exposures to the survivors of the Japanese bombings, the lifetime risk of cancer attributable to a given dose of gamma radiation now appears somewhat larger than previously estimated in the BEIR III report (NAS, 1990). The dose-dependent excess of mortality from all cancer other than leukemia shows no departure from linearity in the range below 400 rem (4 sievert), whereas the mortality data for leukemia are compatible with a linear-quadratic dose response relationship (linear at low doses, but incidence increasing exponentially requiring a quadratic function at higher doses). BEIR V developed separate dose-response relationships for leukemia and non-leukemia to address this observance. The leukemia dose-response relationship developed from human leukemia data uses a linear-quadratic representation without the need for a DREF. The non-leukemia dose-response relationship remained a linear no threshold representation, with the knowledge that a DREF should be applied to non-leukemia.

2.2 BEIR V Dose Rate Effectiveness Factors

For *low* Linear Energy Transfer (LET) radiation, accumulation of the same dose over weeks or months is expected to *reduce* the lifetime risk compared to the same dose received as acute exposure, due to such factors as the repair of sublethal damage. To account for this, BEIR V lists best estimates of Dose Rate Effectiveness Factors (DREFs) as 4 or 5 based on laboratory animal studies, since there is little human data. These are displayed in Table 2.2.

TABLE 2.2. Summary of Non-Leukemia Dose-Rate Effectiveness Factors For Low-LET Radiation (NAS 1990)

Source of Data	Observed Full Range Of Values	Limited for Narrow Range Of Values	Single Best Estimate
Laboratory animal studies			
Specific locus mutation	3 - 10	3 - 7	5
Reciprocal transloc.	5 - 10	5 - 7	5
Life shortening	3 - 10	3 - 5	4
Tumorigenesis	2 - 10	2 - 5	4

2.3 Annual Dose Resulting in 10^{-4} Excess Cancer Risk

DEP and DPH utilize an average single best estimate DREF of 4.5 for solid tumors in the remediation standard. An implicit DREF for leukemia is utilized in the BEIR V linearquadrative model and does not need further correction. Utilizing the data shown in Table 2.1 and a DREF of 4.5, the *best estimate* of the lifetime excess cancer risk level of 10^{-4} is an annual dose of 19 mrem/yr, with a 90% confidence range of 5 mrem/yr to 57 mrem/yr.
These calculations are displayed in Appendix A.

TABLE 2.3. Annual Dose Resulting in 10^{-4} Excess Cancer Risk, with Range of Dose Rate Effectiveness Factors

Description of Estimate*	Dose Rate Effectiveness Factor	Range of Estimates (90% Confidence Intervals) of Annual Dose with 10^{-4} Excess Cancer Risk (mrem/year)
Single Best Estimate, Average	**4.5**	**19**

Lower Range	2	5
Upper Range	10	57

For *high* LET radiation, such as for alpha-emitting radiation, the rate of exposure does not appear to make a difference in risk observed.

Thus, DEP and DPH have established the *health* basis of the radiation remediation standard at 19 mrem/year total effective dose equivalent, plus Connecticut ALARA.

2.4 Considerations of Uncertainty

There is a degree of uncertainty in extrapolating health effects resulting from low doses, and some attention has been given to attempting to verify these effects in population studies. The BEIR V Committee stated that "Studies of populations chronically exposed to low-level radiation, such as those residing in regions of elevated natural background radiation, have not shown consistent or conclusive evidence of an associated increase in the risk of cancer" (NAS, 1990). However, the BEIR V report further noted that "in areas of high natural background radiation, an increased frequency of chromosome aberrations has been noted repeatedly. The increases are consistent with those seen in radiation workers and in persons exposed at high dose levels, although the magnitudes of the increases are somewhat larger than predicted" (NAS, 1990). In commenting on high natural background radiation studies, the BEIR V Committee stated: "A cautious approach is warranted in the interpretation of geographically based mortality surveys. Although 'beneficial' effects of radiation have been alleged on the basis of reduced mortality in high background areas in the United States, analyses that include an adjustment for altitude indicate no 'beneficial' effects. ...This apparently 'beneficial' effect of radiation may, in fact, be an example of confounding, since conditions of reduced oxygen pressure stimulate a wide array of physiological adaptations, which could themselves be protective" (NAS, 1990).

There is an ongoing scientific effort to increase understanding of radiation-related biological processes. A BEIR VII Phase II Committee, as described in Appendix B, is conducting a comprehensive reanalysis of health effects of low-level radiation. It is anticipated that the CT standard can accommodate this reanalysis when it is published. Connecticut has considered a range of doses in setting its standard, includes ALARA, and considers the feasibility of radiation measurement in the presence of background radiation.

In developing the approach outlined in this document, Connecticut DEP and DPH have also considered the range of uncertainty in the risk estimates set forth by several other

national agencies and consensus groups. The range of uncertainty in the Connecticut preferred model, derived from BEIR V, from 5 – 57 mrem/year, is within the ranges of uncertainty in the estimates produced by these other agencies or groups, that are detailed in Appendix E. Another source of uncertainty is the choice of the health endpoint of interest used for deriving a standard. Connecticut DEP and DPH have considered the risks associated with other endpoints, such as teratogenesis or chronic diseases. Because deriving a standard based on the risks of carcinogenic effects is the most protective, DEP and DPH have based the standard on the carcinogenic dose, as presented in Appendix D.

3.0 ALARA in Connecticut: ALARA ANALYSIS

The radiological remediation criteria of 19 mRem/yr total effective dose equivalent to the average member of the critical group established by the State of Connecticut is not an annual exposure limit. It is an upper bound that cannot be exceeded. The annual exposure limit can possibly be taken even lower with the use of the ALARA (As Low As Reasonably Achievable) analysis provided in this document.

When a cognizant party submits their remediation plan to the Connecticut Department of Environmental Protection's (DEP) Division of Radiation, it must include an ALARA analysis. This plan must demonstrate whether it is feasible to further reduce the levels of residual radioactivity to levels below those necessary to meet the dose criteria. A remediation plan must be approved by the DEP for the property to be considered remediated and free of DEP radiological restrictions. This document explains how to prepare the ALARA analysis.

The ALARA analysis described in this document is derived from, but not wholly taken from, the ALARA analysis described in NUREG -1727 NMSS Decommissioning Standard Review Plan appendix D, ALARA Analysis. The information enclosed in Appendix C describes the methods acceptable to DEP staff for determining when it is feasible to further reduce the concentrations of residual radioactivity to below that necessary to meet the radiological remediation criteria. Although this guidance involves the same principles as an operational ALARA program, it does not apply to, nor does it replace guidance for operational ALARA programs. The economic parameters discussed in this analysis have been derived with regard to specific State of Connecticut socioeconomic factors. Assistance was provided from the University of Connecticut's Economics Division, Center of Economic Analysis and the State of Connecticut's Department of Economic and Community Development. These factors include the $3,000 per person-rem and the $ 3,800,000 value of a fatality equivalent to $3,000/person-rem.

"Reasonably achievable is judged by considering the state of technology and the economics of improvements in relation to all the benefits from these improvements. However, a comprehensive consideration of risks and benefits will include risks from non-radiological hazards. An action taken to reduce radiation risks should not result in a significantly larger risk from other hazards." NRC Regulatory Guide 8.8, Revision 3 (1978).

In general, a method for determining whether levels of residual radioactivity are ALARA would have the following characteristics: (1) simple in scope; (2) unbiased, utilizing appropriate dose modeling to relate concentrations to dose; (3) usable as a planning tool for remediation; and (4) efficient, since the surveys conducted for other purposes could be used in the process.

Dose limits typically define an adequate level of protection. When determining compliance with a dose limit that provides adequate protection, measurements with a

conservative bias is often sought. This assures that compliance with the limit established to provide adequate protection has been met. An ALARA analysis is an optimization technique that seeks the proper balance between costs and benefits below the dose limit. A balance requires that each factor be determined with as little bias as possible. If the analysis was intentionally biased in either direction, it would cause a misallocation of resources and would deprive society of the benefits from other uses of the resource. There are different ways that a remediation action can affect the future well being of society. A remediation action can avert future dose, which is a benefit. The remediation action can also cost money, which can be a detriment. According to modern research in economic theory, loss of the effective use of this capital will therefore deprive future generations of the return on the investment of this money, this is a detriment to society. Thus, if a great deal of money is spent for a remediation that would have a very small future benefit, it would be detrimental to future generations. This same concept can be utilized for physical risks, such as industrial and transportation accidents. A great deal of remediation can be perfumed to remove trace quantities of material which pose slight statistical risk but pose a much higher transportation and industrial accident risk. Therefore, societal benefits must be considered in an analysis.

3.1 The method is simple

The method for most applications should be simple. In an ALARA analysis of a remediation action, the primary benefit (i.e., the collective radiation dose that will actually be averted in the future) is uncertain because future land uses, the number of people that will actually occupy a site, and the types of exposure scenarios are all uncertain. These uncertainties mean that the future collective dose cannot be known with any precision. Because of the inherent limitation on the ability to precisely determine the future collective dose at a particular site, it is not useful to perform a complex analysis when a simple analysis may be appropriate. A facility may use more complex or sitespecific analyses if more appropriate for their specific situations. This does not imply that the computer modeling performed to determine a dose assessment is either poor or inadequate.

3.2 The method is not biased and uses appropriate dose modeling to relate concentrations to dose

The determination of ALARA should not be biased. This is different from demonstrating compliance with a dose limit. Unlike a demonstration of compliance, an ALARA analysis is an optimization technique that seeks the proper balance between costs and benefits below the dose limit required for compliance. To achieve a proper balance, each factor in the ALARA analysis should be determined with as little bias as possible. If the ALARA analysis was intentionally biased, it could cause a misallocation of resources and deprive society of the benefits from other uses of the resources. Thus, the ALARA analysis should provide an unbiased analysis of the remediation action that will avert

future dose (a benefit to society) at reasonable cost. The total cost of remediation is a potential detriment because it can deprive future generations of the return on the investment of this money. This is discussed later along with the methods that should be used in estimating benefits and detriments costs. This includes scenarios, models, and parameters for determining concentrations of activity at a site.

3.3 The method is usable as a planning tool for remediation

Before starting a remediation action, one must to determine the concentration of residual radioactivity required to meet the ALARA requirement. It would be inefficient if it could not be determined if the area would pass the ALARA test until after the remediation. Establishing ALARA post-remediation could result in it being less likely for a facility to remediate below the dose limit because of the additional start-up costs associated with doing additional remediation.

An ALARA analysis should be conducted during remediation planning, before the start of remediation, but after some or all of the site characterization is done.

The method described in this appendix should be used only to determine whether and where particular remediation actions should be taken to meet the ALARA requirement. The analysis described in this section and Appendix C is used only to justify *not* taking a remediation action. For example, if a facility plans to wash room surfaces either to the dose limit or as a good practice procedure, there is no need to analyze whether the remediation action of washing is necessary to meet the ALARA requirement.

3.4 The method is efficient, and as much as possible, uses the results of surveys conducted for other purposes

The demonstration that the ALARA requirements have been met should not require surveys beyond those already performed for other purposes, such as the characterization survey and the final status survey. If possible, surveys used for other purposes should be used to demonstrate compliance with the ALARA requirement.

3.5 ALARA Analysis

A simple method for demonstrating compliance with the ALARA requirement is described in this section and Appendix C. More complex or site-specific analysis may be used if it is determined to be more appropriate and authorized by the DEP. In general, complex analyses may not follow the concepts presented herein. Evaluation of more complex analyses will be handled on a case-by-case basis. Early involvement of the DEP is required if this is the chosen course of action.

It is very difficult or impossible to place a monetary value on an impact. However, a best effort should be made to assign a monetary value to an impact because there may be no other way to compare benefits vs. costs. If there are situations when a credible monetary value cannot be developed, a qualitative treatment may be the most appropriate approach. Qualitative analyses will be evaluated on their merit, on a case-by-case basis. The simplified method presented in this document is to estimate when a remediation action is cost-effective using generalized estimates. If the desired beneficial effects or benefits from the remedial action are greater than the undesirable effects or costs of the action, the remediation action being evaluated is cost-effective and shall be performed. Conversely, if the benefits are less than the costs, the levels of residual radioactivity that meet the dose criteria are already ALARA without taking additional remediation action. Examples of various benefits and costs are listed in Table One. The value of any benefit or cost can be negative.

Table 3.1: Possible Benefits and Costs Related to Decommissioning

Note to reader: a direct correlation between benefits and costs is not to be assumed. (NUREA 1727)

Possible Benefits	Possible Costs
Collective Dose Averted	Remediation Costs
Regulatory Costs Averted	Additional Occupational/Public Dose
Changes in Land Values	Occupational Non-Radiological Risks
Esthetics	Transportation Costs and Associated Risks
Reduction in Public Opposition	Environmental Impacts
Future Use	Loss of Economic Use of Site/Facility

Equations are derived in Appendix C for factors which could be considered in an ALARA analysis. These include: collective dose averted, regulatory and other costs avoided, changes in land values, and esthetics/reduction in public opposition, as well as calculation of costs and non-radiological risks.

3.6 Suggested Parameter Values

Sometimes it is very difficult or impossible to place a monetary value on an impact. A best effort should be made to assign a monetary value to the impact because there may be

no other way to compare benefits to costs. In these situations, a qualitative treatment may be the most appropriate. Qualitative analyses will be evaluated on their merits on a case-by-case basis. For performing these calculations, acceptable values for some of the parameters are shown in Table 3.2.

Examples of calculations using the Connecticut ALARA approach are given in Appendix C, along with a discussion of how these calculations could be used in making decisions about possible remediation options.

Table 3.2: Acceptable Values for Impacts

Parameter	Value	Reference
Workplace accident fatality rate, F_w	4.2×10^{-8}/hr	NUREG-1727
Transportation Fatal accident rate, F_T	Trucks: 3.8×10^{-8}/km	NUREG-1727
$/ person-Rem	$ 3,000	Connecticut Derived Value
Number of years of exposure, N	Buildings: 70 yr Soil: 1000 yr	NUREG-1727
Population Density, P_D	Building: 0.09 person/m^2 Land: 0.0004 person/m^2	NUREG-1727
Excavation, monitoring, packaging and handling soil	1.62 person-hours/m^3 of soil	NUREG-1727
Monetary Value of Fatality Equivalent to $3,000/person-rem	$ 3.8 Million dollars	Connecticut Derived Value
Waste shipment volume, V_{SHIP}	Truck: 13.6 m^3/shipment	NUREG-1727

4.0 Radiation Background and Measurement Issues

4.1 Radiation Background

Radiation exposure due to radioactive material contamination takes place amidst a background of exposure to ionizing radiation from cosmic and terrestrial radiation, fallout from nuclear weapons testing, internal exposures from ingestion of food, diagnostic and therapeutic medical procedures, and consumer products. These exposures are described in detail in a series of reports from the NCRP. The population weighted averages are summarized in Table 4.1 (NCRP 1987b):

Table 4.1. Annual Radiation Exposure to Residents of the U.S. Atlantic Plain by Source of Ionizing Radiation (NCRD 1987b)

Source of Ionizing Radiation Exposure	Annual Total Effective Dose Equivalents	
	(mrem/year)*	(msieverts/year)*
Cosmic radiation	23	0.23
Cosmogenic (^{14}Carbon in atmospheric processes)	1	0.01
Terrestrial (external gamma)	32	0.32
Inhaled (radon-related)	200	2.00
In the body	40	0.40
Consumer products, excluding tobacco	6 - 13	0.06 - 0.13
Medical procedures	50	0.50
Rounded Total	**360**	**3.60**
*1 millisievert (msievert) = 100 mrem		

4.1a Background Radiation Considerations

Everything and everyone on the planet is bathed in a sea of ionizing radiation known as background radiation. Anywhere on earth where one goes and picks up a handful of soil, it will contain billions and billions of unstable atoms that over time will undergo the decay process, giving off radiation and eventually becoming stable. While picking up this handful of soil, the body will be bombarded by thousands of gamma rays, and the air that a person breathes will contain natural radioactivity. A person's own body contains natural radioactive elements that accumulate in body tissues and organs depending upon their respective function. Background radiation comes from four major sources. First to be discussed is terrestrial radiation, which produces the largest radiation dose to people living in the Connecticut. The remaining components of background, which are cosmic, cosmogonic and man-made radiation sources, are relatively minor contributors to the dose from background compared to terrestrial radiation. Each of these sources is discussed in the next four sections of this report to give the reader a basic understanding of their origins, physical properties and relative contributions to the total background radiation dose rate.

Although background radiation is everywhere, its level varies. Many different factors lead to these variations in background radiation. The amount of cosmic radiation one is exposed to depends upon the degree of shielding provide by the earth's atmosphere, the higher the elevation the greater the exposure. Conversely the lower the elevation the lower the exposure. The amount of tetrogenic radiation (radiation from radionuclides in soil) is variable. These radionuclides include uranium-235 and 238, thorium-232, potassium-40, rubidium-40 and radium-226. It is important to note the isotopes of

uranium, thorium and radium that are part of man's natural background are also the same radionuclides that may be required to be remediated from past technologically enhanced manmade activities. This leads to difficulty in differentiating what is naturally abundant and what has been contributed by humans when remediating these sites. Not only are these radionuclides commonly found in soil, but are also naturally abundant in building materials such as brick, concrete and stone as discussed in the structure section of this report. This leads to the complicating issue of remediating radionuclides which are naturally found in background concentrations and are close to the desired concentration level of a remediation standard. Therefore two types of measurements are made. Those which have the contaminant of concern in background and those without the contaminant in background.

4.1b Terrestrial Radiation

The naturally occurring forms of radioactive elements that were incorporated into Earth during its formation and that are still present are referred to as terrestrial radionuclides. Virtually all materials found in nature have some degree of natural radioactivity. Rocks, soil, water, air, plants, and animal life all have varying concentrations of terrestrial radioactivity. The most significant of these are uranium-238 and thorium-232, which both decay in a long chain (or series) of various radionuclides, and potassium-40 and rubidium-87. These radionuclides and their decay products or progeny as they are commonly referred to, give off various forms of radiation. A non-inclusive list of these radionuclides can be found in Table 4.2.

The table is given to show the parent and decay product of radionuclides found in background. It also lists the major types of radiation given off in the decay of each radionuclide. Two of the more commonly known radioactive elements in the table are radium, which was discovered by Marie Curie and used extensively for luminous watch dials and medical treatments years ago, and radon, a gaseous decay product of radium that may be a residential concern. Other long-lived radionuclides that are found in background were not listed because their concentration is small and therefore less significant in terms of their contribution to background radiation dose.
A listing of the other radionuclides found in nature can be found in various references. Table 4.3 provides an example of the range of concentrations for naturally occurring radionuclides that can be found in some common materials:

Table 4.2 Terrestrial Radionuclides Contributing to Radiation Background

Nuclide	Half-Life	Major Radiation(s)
Parent-Uranium-238	4.47 billion years	alpha, x-rays

Thorium-234	24.1 days	beta, gamma, x-rays
Uranium-234	245 years	alpha, x-rays
Thorium-230	77,000 years	alpha, x-rays
Radium-226	1600 years	alpha, gamma
Radon-222	3.83 days	alpha
Lead-214	26.8 minutes	beta, gamma, x-rays
Bismuth-214	19.7 minutes	beta, gamma
Polonium-210	138 days	alpha
Lead-206	Stable	Stable
Parent-Thorium-232	14.1 billion years	alpha, x-rays
Radium-228	5.75 years	beta
Actinium-228	6.13 hours	beta, gamma, x-rays
Thorium-228	1.91 years	alpha, gamma, x-rays
Radium-224	3.66 days	alpha, gamma
Radon-220	55.6 seconds	alpha
Lead-212	10.64 hours	beta, gamma, x-rays
Bismuth-212	60.6 minutes	alpha, beta, gamma, x-rays
Lead-208	Stable	Stable
Parent-Potassium-40	1.28 billion years	beta, gamma
Argon-40	Stable	Stable
Parent-Rubidium-87	47 billion years	beta
Strontium-87	Stable	Stable

Table 4.3 Typical Ranges in Average Concentration of Background Radionuclides (pCi/gram)

Material	**Uranium-238**	**Thorium-232**	**Potassium-40**
Bauxite Ore	6.8	5.4	N/A
Coal	0.5	0.6	1.4
Crustal Rock	1	1.19	23
Phosphate Fertilizer	249 n/a n/a	Soil (U.S. avg.) 1 1 n/a	

4.1c Cosmic and Cosmogonic Radiation

Radiation is in many forms, from high speed heavy particles to high energy photons and muons. Cosmic radiation, commonly known as cosmic rays, consist of highly energetic particles, mostly the nuclei of the elements hydrogen and helium. Supernova explosions and other phenomena that occur throughout the universe are considered to be the source of cosmic rays. The upper atmosphere interacts with many of the cosmic radiations, and produces radioactive nuclides. They can have long half-lives, but the majority have shorter half-lives than the primordial nuclides. Here is a table with some common cosmogonic nuclides:

Table 4.4 Cosmogonic Nuclides

Nuclide	Symbol	Half-life	Source	Natural Activity
Carbon 14	14C	5730 yr	Cosmic-ray interactions	6 pCi/g (0.22 Bq/g) in organic material
Tritium 3	3T	12.3 yr	Cosmic-ray interactions with N and O; spallation from cosmic-rays	0.032 pCi/kg (1.2 x 10-3 Bq/kg)
Beryllium 7	7Be	53.28 days	Cosmic-ray interactions with N and O;	0.27 pCi/kg (0.01 Bq/kg)

Some other cosmogonic radionuclides are 10Be, 26Al, 36Cl, 80Kr, 14C, 32Si, 39Ar, 22Na, 35S, 37Ar, 33P, 32P, 38 Mg, 24Na, 38S, 31Si, 18F, 39Cl, 38Cl, 34mCl.

Production of charged particles on the Sun during solar flares can occasionally produce significant radiation doses on earth also.

4.1d Sources of Human Origin

Human activities have resulted in the production of various sources of radiation. Nuclear reactor and weapons have produced radionuclides through the fissioning of uranium and other heavy elements and the activation of various elements. Particle accelerators used in scientific research have produced smaller quantities. Although most of these radionuclides are short lived and quickly decay to form stable forms, a few have half lives of several to thousands of years. Technological advances such as nuclear reactors, particle accelerators and the development of nuclear weapons have lead to small increases in some background radiation.

4.2 Measurement Issues

At a site where radiation contamination is discovered, characterization of the particular radionuclides of concern must take place. This involves identification of the

radionuclides and their concentrations. One can then use them to calculate estimated carcinogenic mortality and morbidity associated with exposure to the radiation found.

4.2a Purpose and Scope

The purpose and scope of this section is to explain the current limitations on radiological instrumentation utilized in radiological detection. These limitations are due to the types of radiation emitted, type of structure being investigated such as soil and building material, background issues associated with different structures and locations being surveyed, type of measurement being performed, instrument detection sensitivities for the radiation being emitted, the instruments ability to achieve the desired investigation level, scaling factors, and human use/human error factors. These factors are the variables which lead to the calculation of the detectors minimum detectable concentration that is discussed later. The information utilized in this report was taken from current radiological guidance documents used in the decommissioning process and represent the best current state of technological knowledge.

4.2b Types of Radiation and Radionuclides Detected

The radionuclides of concern associated with radiological remediation activities may be emitting alpha, beta or gamma radiation or a combination of radiations. Radioactive material can be present as a solid, liquid or gas, and may be classified into different categories which include: source material, by-product material, special nuclear material or naturally occurring or technically enhanced material, also known as NARM. These radionuclides may occur naturally and be part of man's everyday environment. The radionuclides of concern that need to be detected and remediated should be determined during the historical review of the remediated facility. This information is also critical to determine the amount of radioactivity that must be remediated to achieve the release criteria. This quantity, the maximum concentration of radioactive material allowed, is known as the Derived Concentration Guidance Limit.

4.2c Radiological Release Validation Instrumentation

The types of radiological instrumentation commonly utilized by the State of Connecticut's Department of Environmental Protection's Division of Radiation will be described in this section. The significance of these instruments is that they are the types most likely to be utilized by the DEP for the purposes of radiological post-remediation validation. The instrumentation discussed is both hand-held and fixed laboratory instrumentation. The portable instrumentation used is battery operated, and somewhat rugged. In both cases the instrumentation requires the use of proper protocols including operational and quality control procedures. Not only are these following instrument types utilized by the DEP but they are also widely used by health physics professionals in radiological remediations.

Table 4.5 provides a list of types of portable instruments in common usage that may be utilized by DEP in assessing sites, along with the types of radiation for that they are most effective in measuring.

Table 4.5 Types of Portable Instruments with Their Primary Radiations for Which They are Most Effective

Types of Portable Instrument	Type of Radiation That Instrumentation is Effective in Detecting
Gas proportional detectors	Beta, alpha
Geiger-Muller detectors	Beta, gamma
Zinc sulfide scintillation detectors	Alpha
Sodium iodide scintillation detectors (NaI)	Gamma
Field Instrument for Detecting Low Energy Radiation (FIDLER)	Low energy beta and alpha
Pressurized Ion Counter	Gamma

Similarly, a number of laboratory instrumentation may be used in assessing the radioactivity of samples. Some of the most commonly used are listed in Table 4.6.

Table 4.6 Types of Laboratory Instrumentation Used for Analysis of Radiation Samples

Types of Laboratory Instrumentation	Type of Radiation for That Instrumentation is Efficient in Detecting
Silicon detectors	X-ray, Beta
Germanium detectors	X-ray, Gamma
Liquid scintillation	Alpha, Beta
Surface barrier detectors	Alpha
Gas flow proportional counters	Alpha, Beta
Sodium Iodide detectors	Gamma

A detailed analysis of the strengths and limitations of each type of field or laboratory instruments are discussed in Appendix F. This includes relevant issues affecting the minimum detectable concentrations in making radiation measurements, considering the effects associated with various surfaces and materials, and other factors.

5.0 Discussion and Recommendations

Connecticut DEP and DPH are proposing a health-based standard of 19 mrem/year TEDE above natural background for clearance at sites where radioactive contamination is found. In recommending this standard, DEP and DPH have also considered emerging issues in the science. DEP and DPH are also cognizant of issues of feasibility of measurement, and of issues related to the relatively high levels of natural background radiation.

Cancer, reproductive, and other chronic effects were all considered in the development of the radiation remediation standard. However, since cancer-causing effects occur at lower levels of exposure than other effects, the standard is based on carcinogenicity considerations. The preferred model that DEP and DPH are basing the radiation remediation standard in is in the NAS BEIR V report. DEP and DPH considered standards of federal agencies, states, or consensus bodies, that are detailed in Appendices D & E.

The radiation remediation standard considers dose assessment objectives for a general population, which differ from occupational radiation protection. Exposure to the general population must include a broader range of human variability in populations compared with a healthy working population. This variability is seen in such factors as age ranges, health status, and modes of exposure. These factors are taken into account in the modeling that will be used to assess the adequacy of clean-up at the various types of sites. Scenarios include the resident farmer, employee of a facility, truck farmer, etc.

Exposure to ionizing radiation should be kept to as low as reasonably achievable, ALARA. Connecticut's approach to ALARA is simple in scope, unbiased in its use of modeling, useful as a tool in planning, and efficient in using already existing data. It takes into consideration the cost of living in Connecticut and inflationary factors, and considers the impact of non-radiological risks in the remediation process.

The proposed standard is driven by the carcinogenic risk, which is greater than that for genetic effects. When comparison is made with estimates of carcinogenic risk, the Connecticut 19 mrem/year dose (TEDE) can be equated with a lifetime risk of cancer mortality of 10^{-4}. The 19 mrem/year standard like other carcinogens with relatively high background is at the upper limits of the carcinogenic risk range utilized by DPH for environmental contaminants, 10^{-6} to 10^{-4}. Considerations of measurement capability in the presence of background radiation enter into this decision. This standard is designed to apply to the range of radioactive contamination situations that may be encountered. Thus, Connecticut DEP and DPH believe that a level of 19 mrem/year Total Effective Dose Equivalent above background for radiation contamination represents the most appropriate and feasible standard.

Glossary of Terms

Acute
Occurring over a short time in comparison to the effect, usually a few minutes to days.

ALARA
Acronym for "As Low As Reasonably Achievable," means making every reasonable effort to maintain exposures to ionizing radiation as far below the dose limits as practical, taking into account the state of technology, the economics of improvements in relation to state of technology, the economics of improvements in relation to benefits to the public health and safety, other societal and socioeconomic considerations, and in relation to utilization of licensed materials in the public interest.

Alpha particle
A positively charged particle emitted from the nuclei during radioactive decay. It is identical to a helium nucleus with a mass number of 4 and an electrostatic charge of +2.

Background radiation
Radiation from cosmic sources, naturally occurring radioactive materials, and global fallout as it exists in the environment from the testing of nuclear explosive devices. It does not include radiation from source, byproduct, or special nuclear materials regulated by the Nuclear Regulatory Commission or state regulated sources containing NORM. The typically quoted average individual exposure from background radiation is 360 millirems per year.

Becquerel (Bq)
The unit of radioactive decay equal to 1 disintegration per second. 37 billion ($3.7x10^{10}$) becquerels = 1 curie (Ci).

Beta particle
A charged particle emitted from a nucleus during radioactive decay, with a mass and magnitude of charge equal to an electron.

Carcinogen
Any substance that may produce cancer.

CERCLA
The Comprehensive Environmental Response, Compensation, and Liability Act of 1980, also known as Superfund.

Chronic
Occurring over a long period of time.

Committed dose equivalent
This is the dose to some specific organ or tissue that is received from an intake of radioactive material by an individual during the 50-year period following the intake.

Committed effective dose equivalent
The committed dose equivalent for a given organ multiplied by a weighting factor.

Curie (Ci)
The basic unit used to describe the intensity of radioactivity in a sample of material. The curie is equal to 37 billion ($3.7X10^{10}$) disintegrations per second.

Deep Dose Equivalent

Dose, absorbed
The amount of energy deposited in any substance by ionizing radiation per unit mass of the substance. It is expressed in units of rads or grays.

Dose equivalent
The product of absorbed dose in tissue multiplied by a quality factor for the tissue, and other necessary modifying factors at the location of interest. It is expressed in units of rems or sieverts.

Dose Rate
The quantity of a dose delivered per unit time.

Dose Rate Effectiveness Factor (DREF)
A factor correcting the effect caused by a specific dose of radiation at high dose of dose rate to accurately represent the effects at low dose or low dose rates.

Epidemiology
The study of the occurrence and causes of health effects in human populations. An epidemiological study often compares two groups of people who are alike except for one factor, such as exposure to a chemical or the presence of a health effect. The investigators try to determine if any factor is associated with the health effect.

Exposure
Contact with a substance. Exposure may be short term (acute) or long term (chronic).

External radiation
Exposure to ionizing radiation when the radiation source is located outside the body.

Gamma radiation High-energy, electromagnetic radiation emitted from the nucleus.

Gray (Gy)
The international system (SI) unit of radiation absorbed dose expressed in terms of absorbed energy per unit mass of tissue. The gray is the unit of absorbed dose and replaces the rad. 1 gray = 1 Joule/kilogram and equals 100 rad.

Half-life, Radioactive
Time required for a radioactive substance to reduce its activity by decay to 50% its initial activity.

Linear Energy Transfer (LET)
Average amount of energy transferred to the material per unit track length.

Milli-One Thousandth

Mortality
The death rate: ratio of number of deaths to a given population.

Morbidity
Illness or disease. Morbidity rate is the number of illnesses or cases of disease in a population.

Mutagenesis
The induction of genetic mutation.

Mutagenicity
Causing genetic mutation.

Quality factor
The factor by which the absorbed dose (rad or gray) is multiplied by obtaining a quantity that expresses, on a common scale for all ionizing radiation, the biological damage (rem or sievert) to an exposed individual.

Rad
The unit for radiation absorbed dose. It is the amount of energy from any type of ionizing radiation (e.g., alpha, beta, gamma, neutrons, etc.) deposited in any medium (e.g., water, tissue, air). A dose of one rad means the absorption of 100 ergs per gram of absorbing tissue (100 rad = 1 gray).

Radionuclide
Atoms that emit radiation spontaneously.

Radioisotopes
Radioactive forms of individual elements.

Risk
In risk assessment, the probability that something will cause injury, combined with the potential severity of that injury.

REM
(Roentgen Equivalent Man) - a standard unit that measures the effects of ionizing radiation on humans.

Risk Assessment
Identification and quantification of the risk resulting from a specific use or occurrence of a chemical or physical agent, taking into account possible harmful effects on

individual people or society of using the chemical or physical agent in the amount and manner proposed and all the possible routes of exposure. Quantification ideally requires the establishment of dose-effect and dose-response relationships in likely target individuals and populations.

Sievert (Sv)
The international system (SI) unit for dose equivalent equal to 1 Joule/kilogram. 1 sievert = 100 rem.

Teratogen
Agent that, when administered prenatally (to the mother), induces permanent structural malformations or defects in the offspring.

Total Effective Dose Equivalent (TEDE)
Sum of the total external radiation (DDE) and internal committed effective dose equivalent (CEDE) to all organs and tissues.

Abbreviations used in this report

ALARA	As low as is reasonably achievable
ATSDR	Agency for Toxic Substances and Disease Registry
BEIR	Biologic Effect of Ionization Radiation
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CMR	Code of Massachusetts Regulations
CPM	Counts per minute
DCGL	Derived Concentration Guidance Level
DEP	Connecticut Department of Environmental Protection
DPM	Disintegrations Per Minute
DNA	Deoxyribonucleic acid
DPH	Connecticut Department of Public Health
EPA	Environmental Protection Agency
FGR	Federal Guidance Report
ICRP	International Council on Radiation Protection
LET	Linear Energy Transfer
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	Minimum Detectable Concentration
MDCR	Minimum Detectable Count Rate
MeV	Million electron volts
MREM	Millirem
NAS	National Academy of Sciences
NCRP	National Council on Radiation Protection
NESHAPS	National Emission Standards for Hazardous Air Pollutants Radionuclides
NIST	National Institute of Standards and Technology
NRC	Nuclear Regulatory Commission
OSWER	Office of Solid Waste and Emergency Response

RAD	Radiation absorbed dose
RBE	Relative biological effectiveness
RCRA	Resource Conservation and Recovery Act
RESRAD	(We need to define.)
TEDE	Total Effective Dose Equivalent

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Appendix A. Derivation of Dose Corresponding to Estimates of Excess Cancer Mortality, as Derived from NAS BEIR V, 1990, with Range of Dose Rate Effectiveness Factors and 90 % Confidence Intervals

A.1 DREF = 4.5

The risk for a 100 year exposure to 100 mrem per year is 1.75×10^{-3}. This is derived (p. 172) from continuous lifetime exposure, equal doses to all organs, excess cancer mortality. This value represents the average of the male non-leukemia + leukemia risk and the female non-leukemia + leukemia risk, taking into account a Dose Rate

Effectiveness Factor of 4.5 for non-leukemia. The fraction ½ is used to account for the averaging of the male and female risks.

$$\text{risk} = 1 \times \frac{1}{2}\left[\frac{450 \text{ cases}}{4.5} + 70 + \frac{540 \text{ cases}}{4.5} + 60\right]$$

$$\frac{}{100{,}000 \text{ persons for 100 years at 100 mares/yr}}$$

$$\text{risk} = 1.75 \times 10^{-3}$$

risk = 1.75 cases per 1,000 persons exposed for a 100-year lifetime at 100 mrem/yr

For a 30-year exposure, the risk = $1.75 \times 10^{-3} \times \frac{30}{100} = 5.25 \times 10^{-4}$

The risk estimation is then normalized for the dose that would result in a $1\,1\cdot 0^{-4}$risk. This is determined by solving for dose in the following: using the relationship that 100 mrem per year yields a risk of 5.25×10^{-4}, then what dose would yield a risk of $1\,1\cdot 0^{-4}$.

$$\frac{\text{dose}}{1\,10\cdot^{-4}\cdot\text{risk}} = \frac{100 \text{ mrem/yr}}{5.25 \times 10^{-4}}$$

Multiplying both sides of the equation by $1\,10\cdot^{-4}$ risk gives:

$$\text{dose} = \frac{100 \text{ mrem/yr} \times 1 \times 10^{-4}}{5.25 \times 10^{-4}}$$

$$\text{dose} = 19 \text{ mrem/yr}$$

A.2

DREF = 4.5, Lower Bound of Risk, 90% Confidence Interval

The 90% confidence limit ***lower bound*** of risk for a 100 year exposure to 100 mrem per year is 1.033×10^{-3}. This is derived (p. 172) from continuous lifetime exposure, equal doses to all organs, excess cancer mortality. This value represents the average of the male non-leukemia + leukemia risk and the female non-leukemia + leukemia risk, taking into account a Dose Rate Effectiveness Factor of 4.5 for non-leukemia.

$$\text{risk} = \frac{1}{2} \times \left(\frac{320 \text{ cases}}{4.5} + \frac{20}{4.5} + \frac{430 \text{ cases}}{4.5} + \frac{20}{4.5} \right)$$

100,000 persons for 100 years at 100 mrems/yr

$\text{risk} = 1.033 \times 10^{-3}$

risk = 1.033 cases per 1,000 persons exposed for a 100-year lifetime at 100 mrem/yr

For a 30-year exposure, the risk = $1.033 \times 10^{-3} \times \frac{30}{100} = 3.1 \times 10^{-4}$

The risk estimation is then normalized for the dose that would result in a $1\,1 \cdot 0^{-4}$ risk. This is determined by solving for dose in the following: using the relationship that 100 mrem per year yields a risk of 3.1×10^{-4}, then what dose would yield a risk of $1\,1 \cdot 0^{-4}$.

$$\frac{\text{dose}}{1\,10 \cdot^{-4} \cdot \text{risk}} = \frac{100 \text{ mrem/yr}}{3.1 \times 10^{-4}}$$

A.3

Multiplying both sides of the equation by $1\ 10^{-4}$ risk gives:

$$\text{dose} = \frac{100 \text{ mrem/yr} \times 1 \times 10^{-4}}{3.1 \times 10^{-4}}$$

$$\text{dose} = 32.3 \text{ mrem/yr}$$

DREF = 4.5, Upper Bound of Risk, 90% Confidence Interval

The 90% confidence limit ***upper bound*** of risk for a 100 year exposure to 100 mrem per year is 4.11×10^{-3}. This is derived (p. 172) from continuous lifetime exposure, equal doses to all organs, excess cancer mortality. This value represents the average of the male non-leukemia + leukemia risk and the female non-leukemia + leukemia risk, taking into account a Dose Rate Effectiveness Factor of 4.5 for non-leukemia.

$$\text{risk} = \frac{1}{2} \times \left[\frac{830 \text{ cases}}{4.5} + 260 + \frac{800 \text{ cases}}{4.5} + 200 \right]$$

$$\frac{\quad}{100{,}000 \text{ persons for 100 years at 100 mrems/yr}}$$

$$\text{risk} = 4.11 \times 10^{-3}$$

risk = 4.11 cases per 1,000 persons exposed for a 100-year lifetime at 100 mrem/yr

For a 30-year exposure, the risk = $4.11 \times 10^{-3} \times \frac{30}{100} = 1.23 \times 10^{-3}$

A.4

The risk estimation is then normalized for the dose that would result in a $1\,1\cdot 0^{-4}$risk. This is determined by solving for dose in the following: using the relationship that 100 mrem per year yields a risk of 1.23×10^{-3}, then what dose would yield a risk of $1\,1\cdot 0^{-4}$.

$$\frac{\text{dose}}{1\,10\cdot^{-4}\cdot\text{risk}} = \frac{100 \text{ mrem/yr}}{1.23 \times 10^{-3}}$$

Multiplying both sides of the equation by $1\,10\cdot^{-4}$ risk gives:

$$\text{dose} = \frac{100 \text{ mrem/yr} \times 1 \times 10^{-4}}{1.23 \times 10^{-3}}$$

$$\text{dose} = 81.3 \times 10^{-1} \text{ mrem/yr} = 8.1 \text{ mrem/yr}$$

DREF = 2, Lower Range of Dose for 10^{-4} Excess Cancer Risk

The risk for a 100 year exposure to 100 mrem per year is 3.125×10^{-3}. This is derived (p. 172) from continuous lifetime exposure, equal doses to all organs, excess cancer mortality. This value represents the average of the male non-leukemia + leukemia risk and the female non-leukemia + leukemia risk, taking into account a Dose Rate Effectiveness Factor of 2 for non-leukemia.

$$\text{risk} = \frac{1}{2} \times \left[\frac{450 \text{ cases}}{2} + 70 + \frac{540 \text{ cases}}{2} + 60 \right]$$

$$\frac{\left[\frac{450 \text{ cases}}{2} + 70 + \frac{540 \text{ cases}}{2} + 60 \right]}{100{,}000 \text{ persons for 100 years at 100 mrems/yr}}$$

A.5

risk = 3.125×10^{-3}

risk = 3.125 cases per 1,000 persons exposed for a 100-year lifetime at 100 mrem/yr

For a 30-year exposure, the risk = $3.125 \times 10^{-3} \times \frac{30}{100} = 9.375 \times 10^{-4}$

The risk estimation is then normalized for the dose that would result in a $1\,1 \cdot 0^{-4}$risk. This is determined by solving for dose in the following: using the relationship that 100 mrem per year yields a risk of 9.375×10^{-4}, then what dose would yield a risk of $1\,1 \cdot 0^{-4}$.

$$\frac{\text{dose}}{1\,10 \cdot^{-4} \cdot \text{risk}} = \frac{100 \text{ mrem/yr}}{9.375 \times 10^{-4}}$$

Multiplying both sides of the equation by $1\,10 \cdot^{-4}$ risk gives:

$$\text{dose} = \frac{100 \text{ mrem/yr} \times 1 \times 10^{-4}}{9.375 \times 10^{-4}}$$

dose = 11 mrem/yr

DREF = 2, Dose at Lower Bound of Risk for 10^{-4} Excess Cancer Risk, 90% Confidence Interval

The 90% confidence limit ***lower bound*** of risk for a 100 year exposure to 100 mrem per year is 2.075×10^{-3}. This is derived (p. 172) from continuous lifetime exposure, equal

doses to all organs, excess cancer mortality. This value represents the average of the male non-leukemia + leukemia risk and the female non-leukemia + leukemia risk, taking into account a Dose Rate Effectiveness Factor of 2 for non-leukemia.

$$\text{risk} = \frac{1}{2} \times \left[\frac{320 \text{ cases}}{2} + 20 + \frac{430 \text{ cases}}{2} + 20 \right]$$

$$\frac{}{100{,}000 \text{ persons for 100 years at 100 mrems/yr}}$$

$$\text{risk} = 2.075 \times 10^{-3}$$

risk = 2.075 cases per 1,000 persons exposed for a 100-year lifetime at 100 mrem/yr

For a 30-year exposure, the risk = $2.075 \times 10^{-3} \times \frac{30}{100} = 6.225 \times 10^{-4}$

The risk estimation is then normalized for the dose that would result in a $1\,1 \cdot 0^{-4}$ risk. This is determined by solving for dose in the following: using the relationship that 100 mrem per year yields a risk of 6.225×10^{-4}, then what dose would yield a risk of $1\,1 \cdot 0^{-4}$.

$$\frac{\text{dose}}{1\,10 \cdot^{-4} \cdot \text{risk}} = \frac{100 \text{ mrem/yr}}{6.225 \times 10^{-4}}$$

Multiplying both sides of the equation by $1\,10 \cdot^{-4}$ risk gives:

$$\text{dose} = \frac{100 \text{ mrem/yr} \times 1 \times 10^{-4}}{}$$

A.7

6.225×10^{-4}

$_{dose}$ = 16 mrem/yr

Appendix A. Derivation of Dose Corresponding to Estimates of Excess Cancer Mortality, as Derived from NAS BEIR V, 1990, with Range of Dose Rate Effectiveness Factors and 90 % Confidence Intervals A.1 DREF = 2, Lower Bound of Dose, Upper Bound of Risk for 10^{-4} Excess Cancer Risk, 90% Confidence Interval

The 90% confidence limit ***upper bound*** of risk for a 100 year exposure to 100 mrem per year is 6.375 x 10^{-3}. This is derived (p. 172) from continuous lifetime exposure, equal doses to all organs, excess cancer mortality. This value represents the average of the male non-leukemia + leukemia risk and the female non-leukemia + leukemia risk, taking into account a Dose Rate Effectiveness Factor of 2 for non-leukemia.

$$\text{risk} = \frac{1}{2} \times \left[\frac{830 \text{ cases}}{2} + 260 + \frac{800 \text{ cases}}{2} + 200 \right]$$

$$\frac{\left[\frac{830 \text{ cases}}{2} + 260 + \frac{800 \text{ cases}}{2} + 200 \right]}{100{,}000 \text{ persons for 100 years at 100 mrems/yr}}$$

risk = 6.375 x 10^{-3}

risk = 6.375 cases per 1,000 persons exposed for a 100-year lifetime at 100 mrem/yr

For a 30-year exposure, the risk = 6.375 x 10^{-3} x $\frac{30}{100}$ = 1.9 x 10^{-3}

The risk estimation is then normalized for the dose that would result in a $1\,1 \cdot 0^{-4}$risk. This is determined by solving for dose in the following: using the relationship that 100 mrem per year yields a risk of 1.9 x 10^{-3}, then what dose would yield a risk of $1\,1 \cdot 0^{-4}$.

$$\frac{\text{dose}}{1\,10 \cdot^{-4} \cdot \text{risk}} = \frac{100 \text{ mrem/yr}}{1.9 \times 10^{-3}}$$

Multiplying both sides of the equation by $1\,10\cdot^{-4}$ risk gives:

$$\text{dose} = \frac{100 \text{ mrem/yr} \times 1 \times 10^{-4}}{1.9 \times 10^{-3}}$$

$$\text{dose} = 52.6 \times 10^{-1} \text{ mrem/yr} = 5.3 \text{ mrem/yr}$$

A.7 DREF = 10, Upper Range of Dose for 10^{-4} Excess Cancer Risk

The risk for a 100 year exposure to 100 mrem per year is 1.15×10^{-3}. This is derived (p. 172) from continuous lifetime exposure, equal doses to all organs, excess cancer mortality. This value represents the average of the male non-leukemia + leukemia risk and the female non-leukemia + leukemia risk, taking into account a Dose Rate Effectiveness Factor of 10 for non-leukemia.

$$\text{risk} = \frac{1}{2} \times \frac{\frac{450 \text{ cases}}{10} + 70 + \frac{540 \text{ cases}}{10} + 60}{100{,}000 \text{ persons for 100 years at 100 mrems/yr}}$$

$$\text{risk} = 1.15 \times 10^{-3}$$

risk = 1.15 cases per 1,000 persons exposed for a 100-year lifetime at 100 mrem/yr

For a 30-year exposure, the risk = $1.15 \times 10^{-3} \times \frac{30}{100} = 3.45 \times 10^{-4}$

The risk estimation is then normalized for the dose that would result in a $1\,1\cdot 0^{-4}$risk. This is determined by solving for dose in the following: using the relationship that 100 mrem per year yields a risk of 3.45×10^{-4}, then what dose would yield a risk of $1\,1\cdot 0^{-4}$.

$$\frac{\text{dose}}{1\,10\cdot^{-4} \cdot \text{risk}} = \frac{100 \text{ mrem/yr}}{3.45 \times 10^{-4}}$$

Multiplying both sides of the equation by $1\ 10\cdot^{-4}$ risk gives:

$$\text{dose} = \frac{100 \text{ mrem/yr} \times 1 \times 10^{-4}}{3.45 \times 10^{-4}}$$

$$\text{dose} = 29.0 \text{ mrem/yr}$$

A.2 DREF = 10, Upper Bound of Dose for 10^{-4} Excess Cancer Risk, (*lower bound* of risk 90% Confidence Interval)

The 90% confidence limit ***lower bound*** of risk for a 100 year exposure to 100 mrem per year is 5.8×10^{-4}. This is derived (p. 172) from continuous lifetime exposure, equal doses to all organs, excess cancer mortality. This value represents the average of the male nonleukemia + leukemia risk and the female non-leukemia + leukemia risk, taking into account a Dose Rate Effectiveness Factor of 10 for non-leukemia.

$$\text{risk} = \frac{1}{2} \times \frac{\frac{320 \text{ cases}}{10} + 20 + \frac{430 \text{ cases}}{10} + 20}{100{,}000 \text{ persons for 100 years at 100 mrems/yr}}$$

$$\text{risk} = 5.8 \times 10^{-4}$$

risk = 0.58 cases per 1,000 persons exposed for a 100-year lifetime at 100 mrem/yr

For a 30-year exposure, the risk = $5.8 \times 10^{-4} \times \frac{30}{100} = 1.74 \times 10^{-4}$

The risk estimation is then normalized for the dose that would result in a $1\,1\cdot 0^{-4}$risk. This is determined by solving for dose in the following: using the relationship that 100 mrem per year yields a risk of 1.74×10^{-4}, then what dose would yield a risk of $1\,1\cdot 0^{-4}$.

$$\frac{\text{dose}}{1\,10\cdot^{-4}\cdot\text{risk}} = \frac{100 \text{ mrem/yr}}{1.74 \times 10^{-4}}$$

Multiplying both sides of the equation by $1\,10\cdot^{-4}$ risk gives:

$$\text{dose} = \frac{100 \text{ mrem/yr} \times 1 \times 10^{-4}}{1.74 \times 10^{-4}}$$

$$\text{dose} = 57 \text{ mrem/yr}$$

Appendix B. New Relevant Studies Re-evaluation of Current Risk Assessments

B.0 Comprehensive Reanalysis:

The standard presented in this report primarily relies on the analyses of the BEIR V Committee, with note of information presented in the BEIR VII Phase I report.

The most recent comprehensive analysis of health effects of exposure to low levels of ionizing radiation is the BEIR V report of the National Academy of Sciences, published in 1990. The BEIR VI report, 1998, considered alpha emitters and internally deposited radionuclides. A BEIR VII Committee recently (1998) published a preliminary scoping document (Phase I) which reviewed and evaluated the scientific literature pertinent to the biologic and health effects of low-level ionizing radiation, including relevant research in progress. The BEIR VII Committee concluded that information that has become available since 1990 makes this an opportune time to conduct a comprehensive reanalysis of health risks associated with low levels of ionizing radiations, a BEIR VII Phase II

effort. It is anticipated that the study would take about 36 months to complete. DPH and DEP take note of the areas for reanalysis described in the BEIR VII (I) report.

B.1 Epidemiology

The BEIR VII(I) Committee cited the large amount of additional epidemiologic data which have become available since the BEIR V report. They noted that new statistical methods are available to increase the analytic power of interpretation of those data. The extensive new epidemiologic data include categories of:

(1) Nonleukemia cancer mortality. Updates from the Japanese atomic-bomb survivor Life Span Study cohort through 1990, with particular attention to increases in cancer deaths in the population who were under the age of 20 years at the time of the atomic bombings. The preferred risk model was a linear excess-relative-risk model. The excess relative risk per sievert (100 rems) was approximately twice as high for women as for men.

(2). Mortality data updated for patients treated with x-rays.

(3) Mortality among radiation workers. A combined analysis of risk estimates can be compared with those obtained at higher doses from other series.

(4) Site-specific analyses. These include: leukemia; breast cancer; lung; gastrointestinal cancers; lymphatic and hematopoietic cancers other than leukemia; lung, salivary gland, skin, and central nervous system cancers; thyroid cancer; other cancers; and noncancer outcomes.

(5) New data on radiation-related risk in patients known to be genetically susceptible to cancer. These include: excess relative risk for bone sarcoma and soft-issue sarcoma in retinoblastoma patients, who are highly susceptible to radiation-related cancers; breast cancer in ataxia telengiectasis patients; and information on genetic susceptibility to radiation-related cancer in an International Commission on Radiological Protection report.

B.2 Modeling Methods

The BEIR VII (I) Committee also pointed to advancements in analytical methods, including adjustment for bias due to random errors in dosimetry, and systematic presentation of sources of uncertainty in various components of risk estimates and their combined influence.

B.3 Biologic Data

Biologic data are emerging on phenomena that could affect the shape of the doseresponse curve at low doses. Unique aspects of ionizing radiation damage to DNA stems from the

microdistribution of the damage, which can create multiply damaged sites, such as complex double strand breaks, which are difficult to repair, and difficult to detect with standard assays. Within the limits of detection of standard assays of DNA damage, induction of double strand breaks and other lesions in cellular DNA is generally found to depend linearly on radiation dose. The Committee concluded that in mechanistic studies the dose-effect relationships for the formation of chromosomal aberrations and the induction of mutations are curvilinear, with a strong indication of linearity at low doses. The shape of the dose-effect curve will need to be inferred from a deeper understanding of the mechanisms involved in their formation.

Some recent studies suggest that important biologic effects, including some chromosomal changes, can occur in an irradiated population in cells that have received no direct radiation exposure (so-called "bystander" effects). New evidence points to effects of low-level radiation on genomic instability, which could result in damage to cells many cell generations after exposure. Additional evidence suggests that the clusters of damage produced in the DNA at very low doses of radiation are refractory to DNA repair. In contrast, some lines of evidence suggest that adaptive responses might protect cells from later exposures. The Committee in its conclusions noted that some epidemiologic data have been cited as consistent with the existence of hormetic (i.e., beneficial) mechanisms. However, there have been no carefully controlled studies that negate the conservative view that even very low doses of radiation simply add to the burden of cellular damage and thereby increase the likelihood of deleterious mutagenesis.

B.4 Radiologic Principles and Risk Modeling

The BEIR VII(I) Committee described the major areas of disagreement between radiobiologic observations and current risk models, such as the shape of the doseresponse curve for gamma radiation. The majority of radiobiologic observations, whether from chromosome and cell-inactivation studies or findings in animals, are consistent with a curvilinear dependence on dose, a linear-quadratic dependence. However, at low to moderate doses (< 200 rads [2 gray]), there is no apparent deviation from linearity in the excess rates of solid cancer among atomic-bomb survivors, which is the primary source of risk estimates. Thus, dose proportionality is often used in current models. However, other approaches are possible. The International Commission on Radiological Protection (ICRP) has postulated curvilinearity in dose.

The Committee presented the outlines of an alternative approach which incorporates assumed dose rate effectiveness factors into a linear-quadratic numerical model, with the derivation of confidence regions for the resulting linear and quadratic coefficients. They also noted there are still uncertainties in the dosimetry for factory workers in Nagasaki, which could impact the models used for computations of gamma-risk and neutron-risk estimates considerably. They highlighted the need for more-detailed numerical analyses.

Appendix C: Derivation of Equations

C1a. Collective Dose Averted

In the simplest form of an ALARA analysis, the only benefit is the monetary value of the collective averted dose to future occupants of the site. The collective dose averted is defined as the sum of the doses received by the entire exposed population. The ALARA analysis compares the monetary value of the desirable effects (benefits) of a remediation action verses, the monetary benefit of averted dose, with the monetary value of the undesirable effects such as the costs of waste disposal. If the benefits of a remediation action would exceed the costs, then the remediation action should be taken to meet the ALARA requirement.

Simply stated: *If benefits >costs, the remediation action shall be taken.*

The primary benefit from a remediation action is the collective dose averted in the future. Assume:

1. You have a location with residual radioactivity at a concentration C
2. The concentration equivalent to 19 mRem/yr., the ($DCGL_W$), for the site has been determined. The derived concentration guideline level ($DCGL_w$) is the concentration of residual radioactivity that would result in a total effective dose equivalent to an average member of the critical group of 19 mr/yr.
3. The residual radioactivity at a site has been adequately characterized so that the effectiveness of a remediation action can be estimated in terms of the fraction, F of the residual radioactivity that the action will remove.
4. The peak dose rate occurs at time 0 and decreases thereafter by radiological decay.

Therefore, the annual dose D to the average member of the critical group from residual Radioactivity at a concentration C is:

Equation One:

$$D = (0.019 \text{ rem/year}) \left(\frac{C}{DCGLw} \right)$$

If a remediation action would remove a fraction F of the residual radioactivity present, then the annual averted dose $AD_{individual}$ to an individual is:

Equation Two:

$$AD_{individual}\,(rem/yr) = F\,(\,0.019\ rem/yr)\left\lceil \frac{C}{DCGLw} \right\rceil$$

The annual collective averted dose $AD_{collective}$ can be calculated by multiplying the individual averted dose, $AD_{individual}$, by the number of people expected to occupy the area A containing the residual radioactivity. The number of people in the area containing the residual radioactivity is the area, A, times the expected future population density, P_D (person/unit area) for the site.

Thus:

Equation Three:

$$AD_{collective} = F\ (0.019\ rem/yr) \left(\frac{C}{DCGLw}\right) (A)\ (P_D)$$

The annual monetary benefit rate at time 0, BR_0, due to the averted collective dose in dollars/year can be calculated by multiplying the annual collective averted dose $AD_{collective}$ by $ 3,000/person rem.

Equation Four:

$$BR_0 = \$\ 3{,}000\ (F)\ (0.019\ rem/year) \left(\frac{C}{DCGLw}\right) (A)\ (P_D)$$

The total monetary benefit of averted doses can be calculated by integrating the annual benefit over the exposure time in years considering both the present worth of future benefits and radiological decay. It is logical to consider the present worth of benefits and costs that occur in the future. A major thought to consider in the concept of future annual benefit is that money spent today is worth more than money spent in the future.

The equation for the present worth PW of a series of constant future annual benefit rates BR_o ($/yr) for N years at a monetary discount rate per year (r) using continuous compounding is:

Equation Five:

$$PW = BR_0 \left[\left(\frac{1}{r} - e^{-rN}\right)\right]$$

The continuous compounding form of the present worth equation is used because it permits an easy formulation that includes radiological decay. If the annual benefit rate BR is not constant but is decreasing from its original rate BR_0 due to radiological decay, the radiological decay rate acts like an additional discount rate that can be added to the monetary discount rate of decline so that the present worth factor PW becomes:
Equation Six:

$$PW = BR_0 \left[\frac{1}{-(re+\lambda)}\right]^{N}$$

As the number of years N gets longer, the equation may be approximated by the limit: $N \rightarrow \infty$:

Equation Seven:

$$PW = BR_0 \left[\frac{}{r+\lambda}\right]$$

For long-lived radionuclides, another approximation may be made from equation six for a fixed N. Here, the present worth of future benefits PW, will be proportional to the number of years. For r and λ, the approximation of the limitations, $(r + \lambda)\, N \rightarrow 0$, the equation has the form:

Equation Eight:

$$PW = (BR_0)\,(N)$$

In the scenario considering only the collective dose averted with finite r, λ, and N, the total benefit B_{Total} is the present worth of the annual benefit rates due to the averted dose. B_{Total} can be calculated by expressing all terms of equation five since $B_{Total} = B_{AD}$:

Equation Nine:

$$B_{AD} = PW = (\$\ 3{,}000)\,(F)\,(0.019\ rem/year)\left[\frac{}{DCGLC_W}\right](A)\,(P_D)\left[\frac{1}{-(re+\lambda)}\right]$$

Where:

$3000 = Acceptable value for collective dose averted in dollars per person-Rem.

F = fraction of the residual radioactivity removed by the remediation action. F may be considered to be the removable fraction for the remediation action being evaluated.

0.019 mrem/year = annual dose to an average member of the critical group from residual radioactivity at the Derived Concentration Guideline Level (DCGLw) concentration in Rem/yr.

C = average concentration of residual radioactivity in the area being evaluated in units of activity per unit area for buildings or activity per unit volume for soils.

$DCGL_w$ = derived concentration guideline equivalent to the average concentration of residual radioactivity that would give a dose of 19 mRem/yr to the average member of the critical group, in the same units as "Conc".

A = area being evaluated in square meters (m^2)

P_D = population density for the critical group scenario in people/m^2

r = monetary discount rate in units yr^{-1}

λ = radiological decay constant for the Radionuclide in units yr^{-1}

N = number of years over which the collective dose will be calculated.

An acceptable value for collective dose is $ 3,000 per person-Rem averted, discounted for dose averted in the future. This parameter was determined using the value given in NUREG 1727 and applying economic factors specific to the State of Connecticut.

For buildings, the collective averted dose from residual radioactivity should be based on some form of the building occupancy scenario. For land, the collective averted dose will generally be

based on the resident farmer scenario. These are two examples of computer model scenarios utilized to estimate future exposure to populations from residual radioactivity. In general, the ALARA analysis should use the same critical group scenario that is used for the compliance calculation.

The present worth of the benefit from the collective averted dose calculated by the previous equation assumes that the peak dose occurs in the first year. This is almost always true for the building occupancy scenario, but not always true for the residential scenario where the peak dose can occur in later years.

The $DCGL_w$ used shall be the same as the $DCGL_w$ used to show compliance with the 19mRem/yr-dose limit. The population density, $P_{D,}$ should be based on the dose scenario used to demonstrate compliance with the dose limit. Thus, for buildings, the $P_{D,}$ should be for the building occupancy scenario. For soil, the P_D, should be based on the residential scenario. The factor at the far right of the equation, which includes the exponential terms, accounts for both the present worth of the monetary value and radiological decay.
If more than one radionuclide is present, the total benefit from collective averted dose, is the sum of the collective averted dose for each radionuclide. When multiple radionuclides have a fixed concentration (i.e., secular equilibrium), residual radioactivity below the dose criteria is normally demonstrated by measuring one radionuclide and comparing its concentration to a $DCGL_w$ that has been calculated to account for the dose from the other radionuclides. In this case, the adjusted $DCGL_w$ may be used with the concentration of the radionuclide being measured. The other case is when the ratio of the radionuclide concentrations is not fixed and varies from location to location within a survey unit; this benefit is the sum of the collective averted dose from each radionuclide.

C1b. Regulatory and Other Costs Avoided

This benefit usually manifests itself in ALARA analysis of restricted release versus unrestricted release. By releasing the site with no restrictions, the facility will avoid the various costs associated with meeting costs associated with maintaining restricted release requirements. This can include future regulatory costs.

C1c. Changes in Land Values

The facility should account for any expected change in the value of the site or facility caused by different decommissiong options. This may be difficult to quantify.

C1d. Esthetics/Reduction in Public Opposition

This can be very difficult to quantify. The facility may wish to evaluate the effect of its decommissioning options with respect to the overall esthetics (including the decommissioning activities themselves) of the site and surrounding area. Another factor the facility may wish to consider is the potential reduction in opposition, if there is any, to the decommissioning activities/goal the facility is attempting to propose.

C1e. Calculation of Costs

The facility should assess the costs of the remediation actions being evaluated. When doing a simple evaluation the costs generally include the monetary costs of: 1) The remediation action being evaluated, 2) Transportation and disposal of the waste generated by the action, 3) Workplace accidents that occur because of the remediation action, 4) Traffic fatalities resulting

from transportation of the waste generated by the action, 5) Doses received by workers performing the remediation action, and 6) Doses to the public from excavation, transport, and disposal of the waste. Other costs that are appropriate for the specific case should also be included.

The total cost, $Cost_{T,}$ which is balanced against the benefits, has several components.

Equation Ten:

$$Cost_T = Cost_R + Cost_{WD} + Cost_{ACC} + Cost_{TF} + Cost_{WDOSE} + Cost_{PDOSE} + Cost_{other}$$

Where:

$Cost_R$ = monetary cost of the remediation action (may include "mobilization" costs):

$Cost_{WD}$ = monetary cost for transport and disposal of the waste generated by the action: monetary cost of worker accidents during the remediation

$Cost_{ACC}$ = action: monetary cost of traffic fatalities during transportation of the waste:

$Cost_{TF}$ = monetary cost of dose received by workers performing the remediation action and transporting waste to the disposal facility:

$Cost_{WDOSE}$ = monetary cost of the dose to the public from excavation, transport, and disposal of the waste: other costs as appropriate for the particular situation.

$Cost_{PDOSE}$ =

$Cost_{other}$ =

C1f. Remedial Action Costs

It can be estimated that the costs of a remediation action will be proportional to the area to be remediated. Defining the cost per unit area as the proportionality constant. These costs include: 1) The direct cost of the remediation action itself, 2) the cost of waste disposal including its shipping cost, 3) the monetary costs of workplace accidents during the remediation, 4) the monetary costs of transportation accidents during the shipping of waste, and 5) the monetary value of the dose that remediation workers receive. Calculation of the incremental remedial action costs include standard manpower and mechanical costs. The facility can also account for

any additional regulatory fees. *Survey costs related to evaluating compliance of the dose limit are not part of the ALARA analysis.*

C1g. Transportation and Disposal of the Waste

The cost of waste transport and disposal, $Cost_{WD}$, may be evaluated according to the following equation.

Equation Eleven:

$$Cost_{WD} = V_A \ (Cost_V)$$

Where:

V_A = volume of waste produced, remediated in units of m^3

$Cost_v$ = cost of waste disposal per unit volume, including transportation cost, in units of $\$/m^3$

C1h. Non-Radiological Risks

The cost of non-radiological workplace accidents, $Cost_{ACC}$, may be evaluated using the equation below.

Equation Twelve:

$$Cost_{ACC} = \$3{,}800{,}000 \ (F_w) \ (T_A)$$

Where:

$3,800,000 = monetary value of a fatality equivalent to $ 3000/person-rem

F_w = workplace fatality rate in fatalities/hour worked

T_A = worker time required for remediation in units of worker-hours

C1i. Transportation Risks

The cost of traffic fatalities incurred during the transportation of waste, $Cost_{TF}$, may be calculated according to the equation below.

Equation Thirteen:

$$Cost_{TF} = \$\ 3{,}800{,}000\ (V_A / V_{ship})\ (F_T)\ (D_T)$$

Where:

V_A = volume of waste produced in units of m^3

V_{ship} = volume of a truck shipment in m^3

F_T = fatality rate per kilometer traveled in units of fatalities/km

D_T = distance traveled in km

The actual parameters should depend on the site's planned method of waste transport. Some facilities may consider a mix of trucking and rail or barge transport to get the waste to the disposal site. In these cases, the cost would be equivalent to the total fatalities likely from the rail or barge transport and the limited trucking, not just the trucking, rail or barge alone.

C1j. Worker Dose Estimates

The cost of the remediation worker dose, $Cost_{wDose}$, can be calculated as shown in the following equation.

Equation Fourteen:

$$Cost_{wDose} = \$3{,}000\ (D_R)\ (T)$$

Where:

D_R = total effective dose equivalent rate to remediation workers in units of Rem/hr

T = time worked (site labor) to remediate the area in units of person-hour

C1k. Loss of Economic Use of Property

A cost that might fall into the other category, $Cost_{other}$, could include the fair market rental value for the site during the time the remediation work was being performed. This cost may be added to earlier calculated costs. These costs are usually associated with locations such as laboratories, hospital rooms, and industrial sites.

C1l. Environmental Impacts

Another cost that could fall into the other category would be a remediation action that may damage an ecologically valuable area or cause some other adverse environmental impact. These impacts should be included as costs of the remediation action.

Appendix C2. Residual Radioactivity Levels that are ALARA

The residual radioactivity level that is ALARA is the concentration, *C*, at which the benefit from removal equals the total cost. Setting the total cost, $Cost_T$, equal to the present worth of the collective dose averted. The ratio of the concentration, Conc, to the $DCGL_w$, can be determined as follows:

Equation Fifteen:

$$\frac{C}{DCGL_w} = \left[\frac{Cost_T}{\$3{,}000(P_D)(0.019)(F)(\ )A}\right]\left[\frac{r+\lambda}{1-e^{-(r+\lambda)N}}\right]$$

All the terms in the preceding equation have been defined previously.

Since P_D, N, and r are constants that have generic values for all locations on the site, the facility only needs to determine the total cost, $Cost_T$, and the effectiveness, F, for a specific remediation action. If the concentration at a location exceeds C, it will be cost effective to remediate the location by a method whose total cost is $Cost_T$. Note that the concentration, C, that is ALARA can be higher or lower (more or less stringent) than the $DCGL_W$, although the facility must meet the $DCGL_W$.

Examples of Calculations

C2a. Example 1: Washing Building Surfaces

This example considers a building with cesium-137 residual radioactivity (λ= 0.023/yr). The remediation action to be considered is washing a floor of 100 m^2 area. The facility estimates that this will cost $400 and will remove 20 percent (F= 0.2) of the residual radioactivity. For this example assume the following values for the parameters : P_D = 0.09 person/m^2, r = 0.07/yr, and N= 70 years. Using these values in equation fifteen:

$$\frac{C}{} = \left[\frac{\$400}{}\right]\left[\frac{0.07+0.023}{}\right.$$

$$\frac{C}{DCGL_W} \quad \frac{\$5{,}000}{(\$3{,}000)(0.2)(0.019)(0.09)(100 m^2)} \quad \frac{0.07+0.023}{1-e^{-(0.07+0.023)70}} = \left| \right|$$

$$\frac{C}{DCGL_W} = 0.36$$

To meet the ALARA requirement, the floor should be washed if the average concentration exceeds about 36 percent of the $DCGL_w$. This is more stringent than the radiological release limit of 19 mRem/year TEDE. This calculation shows that washing building surfaces is often necessary to meet the ALARA requirement. If the facility decided not to wash the building surfaces it would have to submit an evaluation that demands in the final status survey that all surfaces would have a concentration below 36 percent of the $DCGL_w$.

C2b. Example 2: Scabbling Concrete in a Building

This example is the same as above except that it evaluates use of a scabbling tool that removes the top 1/8 of concrete. The facility estimates the total cost of the scabbling will be $5,000 for the 100 m^2 floor and estimates that it will remove all the residual radioactivity so that *F=1*.

Using the values in the previous example for all terms except $Cost_t$ and F gives:

$$\frac{C}{DCGL_W} \quad \frac{\$5{,}000}{(\$3{,}000)(1)(0.019)(0.09)(100 m^2)} \quad \frac{0.07+0.023}{1-e^{-(0.07+0.023)70}} = \left| \right|$$

$$\frac{C}{DCGL_W} = 0.91$$

The facility could decide to scabble depending on the concentrations present. Instead of scabbling, the licensee could provide this analysis and demonstrate that the floor concentration is less than 0.91 $DCGL_W$.

C2c. Example 3: Removing Surface Soil

In this example, soil with an area of 1000 m^2 is found to contain radium-226 (λ = 0.000247/yr) residual radioactivity to a depth of 15 centimeters (cm). The facility estimates that the cost of removing the soil (F=1) will be $100,000. For the soil scenario, the generic resident former parameter is P_D = 0.0004 person/m^2 and the discount rate suggested by NUREG 1727 is r = 0.03/yr over a thousand year span. Using these values in equation fifteen gives:

$$\frac{C}{DCGL_W}\left(\frac{\$100{,}000}{(\$3{,}000)(1)(0.019)(0.0004)(1000)}\right)\left(\frac{0.03+0.023}{1-e^{-(0.03\ 0.023)1000+}}\right)=$$

$$\frac{C}{DCGL_W}=151$$

Thus, meeting the dose limit would be adequate by a considerable margin. The facility could use this evaluation to justify not removing soil with concentrations residual contamination of radium 226 below the $DCGL_w$. Removal of soil with concentrations of residual radium-226 about the $DCGL_w$ is still required to meet the remediation criteria of 19 mRem/year TEDE.

The advantage of the approach shown in these examples is that it allows the user to estimate a concentration at which a remediation action will be cost-effective prior to starting a remediation and prior to planning the final status survey. Thus, it is a useful planning tool that lets the user determine which remediation actions will be needed to meet the ALARA requirement.

Removal of loose residual radioactivity from buildings is almost always cost-effective except when very small quantities or radioactivity are involved. Therefore, loose residual radioactivity normally should be removed, and if it were removed, the analysis would not be needed.

Appendix C3. Determination of "Net Public or Environmental Harm"

The calculation to demonstrate net public or environmental harm is a special case of the general ALARA calculations described above that compare the benefits in dose reduction to the cost of doses, injuries and fatalities incurred. The calculation does not consider the monetary cost for performing further remediation, $Cost_R$, or the costs of waste disposal, $Cost_{WD}$. Thus, if the benefit from averted dose B_{AD} , is less than the sum of the costs of workplace accidents, $Cost_{ACC}$, the cost of transportation fatalities, $Cost_{TF}$, the costs of remediation worker dose, $Cost_{WDOSE}$, and the costs of any environmental degradation, $Cost_{ED}$, there is net public or environmental harm. Thus, there is net public or environmental harm if :

Equation Sixteen:

$$\text{Net harm if:}\quad B_{AD} < Cost_{ACC} + Cost_{TR} + Cost_{WDOSE} + Cost_{ED}$$

In some cases, it will be very difficult to assign a credible monetary value to environmental degradation. For example, environmental harm could be caused by an action such as remediation of a wetlands area. There may be no way to assign a

monetary value to this action. In these cases it is acceptable to use qualitative arguments that will be evaluated on a case-by-case basis.

Appendix D. Discussion of Alternative Approaches

Cancer, reproductive, and other chronic effects were all considered in the development of this clearance standard. However, since cancer-causing effects occur at lower levels of exposure than other effects, the standard is based on carcinogenicity considerations.

D.1 Teratogenic Risks.

There are specific periods in the development of the human fetus when it is most sensitive to the effects of ionizing radiation. Data for severe mental retardation and microcephaly in Japanese A-bomb survivors exposed at 8 - 15 weeks of gestational age indicates risk has been found to increase more steeply with dose than was expected at the time of the BEIR III report (ATSDR 1998b). The ICRP estimates that the data now reveal the magnitude of this risk to be approximately a 4% chance of occurrence per 10 rem (0.1 Sv). The data imply that there may be little, if any, threshold for the effect when the brain is in its most sensitive stage of development, but this is not certain. For a 15 mrem exposure during this critical period, EPA estimates a risk of 6×10^{-5} (EPA 1989). From week 16–25, the risk is slightly lower, 1% per 10 rem (thus, 1.5×10^{-5} for 15 mrem), but still appreciable compared with the generally used risk ranges of 10^{-6} to 10^{-4} for carcinogenic risk. Risks at other periods of fetal development may be slightly less, based on the Japanese survivor studies (representing an acute exposure) and other data (EPA 1989).

In general, a decrease in intelligence quotient (IQ) has been noted with increasing exposure to radiation during critical periods of fetal development. The Agency for Toxic Substances and Disease Registry (ATSDR) derives a Minimal Risk Level (MRL) of 400 mrem/9 month pregnancy based on the estimates of an IQ reduction of a range of 21–29 (mean: 25 points) due to an exposure of 100 rems (1 sievert) of ionizing radiation during fetal development. At 400 mrem, this equates to a risk of 1.2×10^{-3}, with some factor for human variability considered. ATSDR uses a postulated change in median population IQ test results of 0.3 as a shift that could be accepted as a non-adverse effect (ATSDR 1998b). Although the linear relationship developed for data from the Japanese fetalexposed population is strong, it has *not* been established that the linear relationship holds all the way to the lowest potential exposure levels.

There are ranges of serious congenital and genetic abnormalities that occur spontaneously in various populations. EPA states in the Background Document for NESHAPS that the genetic effects of the background exposure to ionizing radiation are thought to be at an equilibrium level of expression. Given the variation in these effects, EPA estimates that, on average, background radiation causes about 690 serious congenital and genetic abnormalities per 10^6 live births (EPA 1989).

D.2 Chronic (Non-Carcinogenic) Risks

ATSDR has derived an MRL for chronic exposure to external ionizing radiation. An MRL is usually derived for NON-cancer endpoints. Since ATSDR could not identify any individual studies which could be used for a chronic MRL which did not result in a cancer-producing endpoint, the agency applied an uncertainty factor of three, for human variability, to the background level of 360 mrem/year identified by NCRP as the United States average level of ionizing radiation. Thus, ATSDR derived an MRL of 100 mrem/year above background for chronic-duration external ionizing radiation exposure (ATSDR 1998b).

D.3 Carcinogenic Risks

The preferred model upon which DPH and DEP are basing the radiation clearance standard is the modeling in the NAS BEIR V report. The cancer risk estimates derived by other federal agencies or consensus bodies are displayed in Table D.1.

In Table D.1, the column entitled "Agency" describes the guidance or standard used by the particular agency, and the year of publication. The "Dose" column gives the dose in mrem/year that the agency uses. The column entitled "Exposure Duration" gives the number of years over which exposure is assumed to occur in the particular agency's calculations. The column entitled "Risk Estimate" gives the risk of cancer initiation or mortality in a population exposed to the dose listed in column two for the exposure duration in column three. The "Normalized Dose" column gives the dose corresponding to the 10^{-4} excess cancer risk level. The column entitled "Comments" provides explanatory notes, while the final column, entitled "Source-Assumptions," gives the references for the figures used in the dose and risk estimate columns.

Table D.1. Cancer Risk Estimates for Low Level Ionizing Radiation as Derived from Several Federal Agencies or Other Bodies

Agency	Dose	ExposureDuration	Risk-Estimate* For Dose [Column 2] & Exposure Duration [Column 3]	Normalized Dose for 10^{-4} Excess Cancer Risk	Comment	Source-Assumptions
EPA NESHAPS, 1989	10 mrem/ year	30 years	1.9×10^{-4}	5.3 mrem/yr	Excess cancer risk; uses ICRP models for physiology and metabolism, data available from BEIR III; considers metabolism of individual daughter products of radiation in body	Page 6-22, table 6-7. Combined incidence leukemia-bone cancer for all ages.
EPA OSWER, 1997	15 mrem/ year	30 years	3×10^{-4}	5 mrem/yr	Excess cancer risk	Page 4 of Attachment B. Effective Dose Equivalent
NCRP #116, 1993	100 mrem/ year	Lifetime: 50 years for adult exposures; 70 years for children	4.2×10^{-3}	5.6 mrem/yr	Adopted ICRP risk models: 6.0×10^{-4}/rem for fatal and non-fatal cancer together. Uses Dose Rate Effectiveness Factor of 2	Page 3, table 1-1. Committed equivalent dose and committed effective dose concept. Page 25.
NRC, 1997	100 mrem/ year	Lifetime: 30 years exposure	1.5×10^{-3}	6.7 mrem/yr	Linear non-threshold approach as used by ICRP and NCRP	Page 17. 25 mrem/year based on TEDE. 100 mrem from all sources of exposure.

Notes: EPA: 15 mrem/year for MAXIMALLY exposed individual; EPA Safe Drinking Water Act (SDWA) specifies 4 mrem/year radiation Maximum Contaminant Level (MCL) in groundwater. NRC 25 mrem/year, including groundwater, is for AVERAGE exposed individual, with max not to exceed 100 mrem/year from all sources of exposure, excluding medical.

Appendix E. A Review of Other Existing Approaches E.1 Other Northeast States

Other states in the Northeast use varying radiation clean-up standards, ranging from 10 millirem (mrem) per year above background to 25 mrem/year [plus ALARA]. These levels are not directly comparable, without specifying what they refer to, *i.e.*, internal doses only, or internal plus external radiation. The NRC uses a cost figure to determine what in practice is as low as reasonably achievable (ALARA) in given instances, based on occupational exposures. It is approximately $2,000 per person-rem of radiation dose avoided. Connecticut uses an ALARA figure which is keyed both to the cost of living in Connecticut and to inflation.

Table E.1 depicts the radiation clearance levels used by several Northeast states, with comments regarding the matching federal guidances. It is not sufficient merely to compare the numerical limits, but to understand what is encompassed by those limits and their applicability. The units used in this report are in the units previously in common use in the United States, *e.g.*, rads and rems, with conversions indicated to Standard International units, *e.g.*, grays and sieverts.

Table E.1. Radiation Guidance and Standards of Several Northeast States

State	Clean-up Standard	Comments
Maine	25 mrem*/year + ALARA	Uses NRC standard
Massachusetts	10 mrem/year (buildings only)	Regulation for property transfers with license termination; same as EPA air emission standard under National Emission Standards for Hazardous Air Pollutants (NESHAPS)
New Jersey	15 mrem/year (soils only)	Proposed regulation; based on analysis of variation of background radiation in New Jersey, minus radon
New York	10 mrem/year (soils only)	Regulation; same as EPA air emission standard under NESHAPS
* 1 mrem = 1 millirem = 1/1000 rem = 0.001 rem = 0.00001 sievert		

The Northeast States utilize or recommend standards ranging from 10 mrem/year (New York and Massachusetts) to 15 mrem/year (New Jersey) to 25 mrem/year (Maine). Massachusetts uses 10 mrem/year over background when radiological licenses are terminated and properties transferred (Massachusetts CMR) and New York uses 10 mrem/year in evaluating cleanup plans for soils contaminated with radioactive materials (New York, 1993). The proposed New Jersey standard of 15 mrem/year is based on analysis of variation in external and internal background terrestrial radiation, excluding radon (New Jersey, 1999a, 1999b). Maine has no separate standards of its own, but utilizes the NRC standard.

E.2 Relevant Consensus and Federal Guidance

A number of radiation guidelines or standards are in use nationally for different purposes. These include the EPA, the NRC, and the NCRP. The carcinogenic risk estimates associated with these standards are displayed in Table D.1.

Table E.2 displays the standards or guidance given by national consensus bodies or federal agencies in the United States. These range from 10 mrem/year to 100 mrem/year, for continuous exposure. In addition, other countries have adopted criteria. Canada in 1987 adopted a maximum individual *risk* objective of 10^{-6} per year. The other countries adopted individual *dose* objectives: France adopted a limit of < 25 mrem/year in 1993, Germany adopted < 30 mrem/year in 1989, the Nordic Countries adopted < 10 mrem/year in 1989, Spain stated a policy of < 10 mrem/year in 1987, and Switzerland adopted < 10 mrem/year in 1980. The standards or recommendations all refer to radiation contamination *above* naturally occurring background in an area.

The EPA has established standards or guidance for radiation contamination at sites over which it has jurisdiction. These include radionuclides covered by the National Emission Standards for Hazardous Air Pollutants (NESHAPS), and sites regulated under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), with guidance given by the Office of Solid Waste and Emergency Response (OSWER).

Table E.2. National Consensus Recommendations and Federal Agency Guidance

Agency	**Standards-Recommendations**	**Comments**	**Source and Assumptions**
Environmental Protection Agency, 1989	10 mrem/year for air emissions	National Emission Standards for Hazardous Air Pollutants (NESHAPS) Radionuclides	OSWER 9200.4-18 Attachment B. & NESHAP 40 CFR Part 61

Environmental Protection Agency, 1997, 1998b	15 mrem/year for maximally exposed person for CERCLA sites (OSWER); EPA Safe Drinking Water Act specifies limit equivalent to 4 mrem/year for drinking water	Assumes 30 year residential exposure. Uses total cancer incidence risk conversion factor = 6.2 X 10^{-7}/mrem, which is 3 X 10^{-4} for 15 mrem annual exposure for 30 years.	OSWER page 5. Effective Dose Equivalent (internal doses only)
Environmental Protection Agency, 1993, 1998a	Risk range, approximately: (10^{-6} - 10^{-4})	Federal Guidance Report: Radionuclide-specific lifetime radiogenic cancer risk coefficients for the U.S. population, based on agedependent intake, dosimetry, and risk models.	
National Council on Radiation Protection #116, 1993 (replaced NCRP #91)	100 mrem/year for continuous exposure; 500 mrem/year for occasional exposure; (page 35), existing facilities only 1 mrem/year = negligible risk; (page 52)	Recommended remedial action (500 mrem) at level 10 times greater than average annual effective dose equivalent due to external exposure from natural background (50 mrem), exclusive of radon.	Page 35)[upper bound]. Effective Dose Equivalent, Chapter 6.
Nuclear Regulatory Commission, 1997	25 mrem/year + ALARA for average person exposed, with instances up to 100 mrem/year for person from all sources, excluding medical; includes groundwater	Cognizant of NCRP Report # 91. This equals the NRC standard for allowable exposure to the public due to operating nuclear power facilities.	NCRP Report # 91 Page 17. 25 mrem/year based on TEDE. Federal Register Vol. 62, No. 139, Monday, July 21, 1997: page 39062.

The NRC has standards for operating nuclear facilities and for facilities to return to unrestricted uses after license termination. The NCRP has issued recommendations regarding exposure to the public from radiation sources. The NAS, through its committees on Biological Effects of Ionizing Radiation (BEIR) has issued a series of reports assessing the health risks from ionizing radiation. On the international level, the International Council on Radiation Protection (ICRP) has done the same, and these have been referenced in some of the federal documents mentioned above.

The agencies differ in their consideration of *fatal* versus *non-fatal* cancers. EPA bases its risk modeling on the incidence of cancers, whether or not they lead to fatalities. The ICRP and the NRC consider the risk of *fatal* cancers, not of overall cancer incidence. DPH and DEP normally utilize those risk models that consider *incidence* of cancer, not just *mortality* from cancer, when these are available.

E.2a EPA NESHAPS Air Emissions

In its 1989 Background Document to the NESHAPS rule, EPA describes its methodology for estimating carcinogenic risks of low Linear Energy Transfer (LET) (i.e., beta particle and gamma photon) radiation and high LET (alpha particle and neutron) radiation. The EPA central estimate of average lifetime risk from low LET radiation, 392 fatal cancers per million person-rad (or 620 total cancers per million person-rad) is based largely on

results from the NAS BEIR III Committee study of 1980, incorporating the most conservative model assumptions utilized by the Committee, i.e., a linear dose response and age-specific relative risks projected over a lifetime for solid tumors (EPA, 1989). It is important to note that the data relied on by the BEIR III Committee, and hence the EPA NESHAPS document, was limited by the extent of data available at that time, some of which was developed in the 1950s and 1960s. For reasons also described, including the re-evaluation of neutron doses to survivors of Hiroshima and Nagasaki bombings, EPA further stated that estimates of average lifetime risk based on the linear relative risk model must be revised upwards - to roughly 1,200 fatal cancers per million person-rad. Pending other analyses, they considered this an upper bound. They also regarded a lower bound estimate to be 120 fatal cancers per million person-rad. They based the lower bound on laboratory data which showed reduced sensitivity to radiation in some animal species compared with humans, who are much more heterogeneous (EPA, 1989).

EPA sought to emphasize that this estimate cannot be regarded as "conservative" in the sense of providing any significant margin of safety with respect to public health protection. The uncertainty in risks for specific cancer sites may be substantially larger than the uncertainty in the whole-body risk, and the magnitude of risks for individual sites may be larger than the average, or central estimate.

In the case of high-LET radiation, a linear dose response is commonly observed in both human and animal studies, and this linear response is not reduced at low rates. In fact, some data on human lung cancer indicate that the carcinogenic response per unit dose of alpha radiation is maximal at low doses. In the case of low-LET radiation at low dose rates, which could be expected to be encountered in environmental contamination situations, some agencies use models that employ a dose rate effectiveness factor to account for differences. In calculating the risk from a given absorbed dose of alpha particle irradiation, EPA used a smaller quality factor of relative biological effectiveness for alpha radiation (8) than the ICRP used (20), but did not use a dose rate effectiveness factor. Thus, although the models were different, the estimated risks were similar.

E.2b Environmental Protection Agency (EPA) Office of Solid Waste and Emergency Response (OSWER) for CERCLA Site Cleanup

The EPA Office of Solid Waste and Emergency Response (OSWER) cites the EPA NESHAPS Background Document as the basis of its risk estimates. Thus, it adopts the average risk from low-level exposure to ionizing radiation overall for *incidence* of cancer of 6.2×10^{-4} per rem (6.2×10^{-7} per mrem) (EPA, 1987). For 30 years exposure to 15 mrem/year, this yields an estimated risk of 3×10^{-4}. The EPA Issues Paper on Radiation

Site Cleanup Regulations (EPA, 1993) describes the range of 10^{-6} to 10^{-4} as a convenient and practical level for radiation site cleanup regulations, especially if CERCLA provides at least part of the statutory authority for the rulemaking. The RCRA corrective action program already has adopted the CERCLA risk range, so this approach would be familiar to the RCRA-regulated community that handles mixed waste.

E.2c Environmental Protection Agency: Federal Guidance Report No. 13 - Part 1, Health Risks From Low-Level Environmental Exposure to Radionuclides

The EPA has published a series of Federal guidance documents to provide technical information to agencies to assist them in implementation of radiation protection programs. The most recent of these, the Federal Guidance Report (FGR) #13 published in September 1999 (EPA, 1999), addresses risks to health from exposure to specific radionuclides. It uses state-of-the-art methods and models for estimating the risks to health from internal or external exposure. These take into account the age and genderspecific aspects of radiation risk. It provides tabulations of risk estimates, or "risk coefficients", for cancer attributable to exposure to any of approximately 100 important radionuclides through various environmental media. These risk coefficients apply to populations that approximate the age, gender, and mortality experience characterized by the 1989-91 U.S. decennial life tables.

The risk tabulations in this FGR #13 report for exposure to a given radionuclide is expressed as the probability of radiogenic cancer *mortality* or *morbidity* per unit activity inhaled or ingested, for internal exposure, or per unit time-integrated activity concentration in air or soil, for external exposure. An important feature is that these risk coefficients may be applied to *either* chronic *or* acute exposure to environmental radionuclides. EPA considers that the risk coefficients may be interpreted in either of two ways: (1) as average risk per unit exposure for persons exposed throughout life to a constant activity concentration of a radionuclide in an environmental medium; or (2) as average risk per unit exposure for persons acutely exposed to the radionuclide through the environmental medium, as long as the exposure involved is properly characterized as low acute dose or low dose rate. It is emphasized that analyses based on these coefficients should be limited to estimation of total or average risks in large populations. They should not be used for application to specific individuals or to age or gender subgroups, for example, children.

It is necessary first to characterize the individual radionuclides and their daughter products that may be present on a site. Then the approaches used in the FGR #13 may be used to estimate a cancer risk level, which can be compared to the maximum risk ranges used in risk assessments. Connecticut DPH typically uses a risk level of 10^{-6} for drinking water contaminants, unless high background levels of naturally-occurring substances make this unfeasible. For CERCLA sites, as noted above, EPA uses an approach more

consistent with standard risk assessment methods, and derives a maximum on the order of 10^{-4}.

E.2d National Council on Radiation Protection (NCRP)

In 1993 the National Council on Radiation Protection (NCRP) provided its most recent estimates of radiation risks, which, in the interests of a uniform international approach, incorporated the recommendations and concepts provided in the International Commission on Radiation Protection (ICRP) Publication 60 of 1991. This publication superceded its 1987 risk estimates for both cancer and genetic effects which had been generally consistent with those of the 1980 BEIR III report (NCRP 1987a). New considerations included the addition of further data from the studies of the Japanese survivors, additional data on breast cancer, and a greater appreciation of the probable validity of relative risk projection models.

NCPR Report #116 provided estimates of risk for cancer both for workers and for members of the public. Their estimate for *workers* (averaged over both sexes and for all ages) is 4.0×10^{-4} per rem [10^{-2} Sievert] for *fatal* cancer and 0.8×10^{-4} per rem for *nonfatal* cancer. For members of the general public, which were not addressed in the earlier document, their corresponding estimates of risk are: 5.0×10^{-4} per rem for *fatal* cancers and 1.0×10^{-4} per rem for *non-fatal* cancer. The NCRP model effectively endorses the ICRP model in which a dose-rate effectiveness factor of two is used. Their estimate of genetic risk for the general public is 1.3×10^{-4} per rem for severe genetic effects, which is less than their estimate of cancer risk.

NCRP Report #116 use the concept of committed dose. Radiation doses received from radionuclides deposited in organs and tissues will be distributed temporally depending upon the effective half-life of the radionuclide. The Report notes: "For radionuclides with approximate effective half-lives ranging up to about three months, the committed quantities are approximately equal to the annual quantities for the year of intake. For radionuclides with an effective half-life, exceeding three months, the committed equivalent dose and the committed effective dose are greater than the equivalent or effective dose received in the year of intake because they reflect the dose that will be delivered in the future as well as that delivered during the year of intake." Although they noted that the committed equivalent dose and the committed effective dose from the lifelong intake of radionuclides of very long effective half-life will overestimate, by a factor of 2 or more, the lifetime equivalent dose or effective dose, they cautioned that these concepts *are* appropriate for all routine radiation-protection purposes, such as, assessing compliance with the annual effective dose limits and for planning and design. The annual effective dose limit to which they referred is the sum of the external effective dose and the committed effective dose from internal emitters, i.e., the total effective dose equivalent.

NCRP Report #116 also affirmed the concept of negligible incremental risk level, which is a policy judgment. They continued to use the value of 1 mrem/year (above background) as a negligible incremental risk value (equivalent to a projected risk of about 5×10^{-7} per year for fatal cancers). This value was also adopted as a starting point for discussion by the Committee of the National Academy of Sciences which addressed the technical bases for Yucca Mountain standards.

E.2e Nuclear Regulatory Commission (NRC)

In 1994 the NRC published a proposed rule for public comment which set forth 15 mrem/year (0.15 mSv/year) as an acceptable criterion for unrestricted use of a facility upon license termination. The final NRC rule in 1997, however, promulgated, a criterion of 25 mrem/year, with a 100 mrem/year (1 mSievert/year) cap to a degree that is ALARA if institutional controls should fail. A licensee could propose exceeding this cap in unusual site-specific circumstances if, in addition to the normal provisions of restricted use and other controls, the licensee also met the criterion of reducing contamination to no greater than the 500 mrem/year (5 mSv/year) value.

The NRC noted that the risk coefficients used by EPA in assessing the risk from contaminated sites are based on an assumed 30-year lifetime exposure, since land use patterns are such that it is unlikely that an individual will continue to live or work in the same area for more than 30 years. It further noted that the total effective dose equivalent (TEDE) which an individual could receive is usually controlled by relatively short-lived nuclides of half-lives of 30 years or less for which the effect of radioactive decay will, over time, reduce the risk significantly (e.g., at reactors, where much of the contamination is from Co-60 with a half-life of 5.3 years) (NRC, 1997).

The NRC discussed its change from the proposed 15 mrem/year to 25 mrem/year in the Final Rule for Radiological Criteria for License Termination, referencing some of the comments it had received on the proposed rule (NRC, 1997). Although some commenters noted that the risk of fatal cancers from 15 mrem/year is too high in comparison with the risk range 1×10^{-4} to 1×10^{-6} used by EPA in CERCLA regulations, NRC took note of other commenters who stated that precedents from earlier NRC rulemakings support a level of risk significantly greater than that and more appropriately in a range of 1×10^{-2} to 1×10^{-3}. The level of lifetime risk corresponding to the 100 mrem/year public dose limit, which is NRC's basic standard for public safety, is about 1.5×10^{-3} (NRC, 1997). NRC also referenced the recommendations of NCRP and ICRP of 100 mrem/year as annual public dose limits, and thus discussed the standard of 25 mrem/year as affording a degree of conservatism (NRC, 1997).

E.2f NAS, Technical Bases for Yucca Mountain Standards

The Energy Policy Act of 1992 (P.L. 102-486) directed EPA to promulgate standards to ensure protection of public health from high-level radioactive wastes in a deep geologic repository that might be built under Yucca Mountain in Nevada. Congress also asked the NAS to advise EPA on technical issues, which resulted in their 1995 report.

NAS utilized the risk number published by the ICRP for fatal cancers, of 5×10^{-2} per Sv (or 5×10^{-4} per rem), which is consistent with other agencies.

NAS also discussed another issue inherent in a high-level wastes repository, that of intergenerational equity. Their perspective encompassed looking at the time of maximal risk, given the mix of radionuclides to be stored. In some instances, the maximal risk could occur more than 10,000 years after the wastes were stored. Thus, in the NAS proposal, they specified that the basis for the standard should be the peak risk, whenever it occurs.

Appendix F: Instrumentation Considerations Associated With Radiological Release Criteria

F.1 Portable Instrumentation

F1.a Gas Proportional Detectors

Gas Proportional Detectors are utilized for detecting both alpha and beta radiations. This can be performed either independently or simultaneously by changing voltages in the instrument. The effective probe size of the detectors (region of the detector capable of detecting radiation) can be 100 cm^2 or greater and can be either be used as a small hand held instrument or on a larger scale be put on wheels and used to count large surface areas in a short period of time. The capability of having an effective probe surface area of 100 cm2 is important because some activity measurements are reported in activity per 100 cm2 and physical to effective probe calculations thus become unnecessary. This type of detection instrument has high detection efficiencies and its ability to detect both beta and alpha radiations make this instrument highly versatile. Unfortunately this instrument's detector requires P-10 counting gas, a mixture of 90% argon and 10% methane which limits its portability capabilities. This type of instrument would be used on relatively flat surfaces like floors and walls. This type of detector is typically attached to an instrument which records counts of activity in counts per minute. Typical alpha efficiencies for unattenuated alpha sources are about 15-20% and for beta sources unattenuated up to 35%. These instruments are typically utilized for scans and fixed readings. Common uses of these instruments include use for release of smooth surfaces such as floors, walls or work surfaces.

F1b. Geiger-Muller Detectors

When used for the purposes of site remediation surveys this type of instrument utilizes a probe which is widely referred to as a "pancake" type detector and is generally used for detecting beta and gamma radiations and to a lesser extent alpha radiation. This instrument has an effective probe area of only 15.5 cm2. This type of detector also utilizes a counting gas, a mixture of argon, helium, neon and a halogen -quenching gas, but is sealed at the factory and does not require being recharged. This makes this instrument extremely versatile as a hand held field instrument. Due to the relatively small size of the detector and inability to detect lower energy beta particles (such as from H3) the detection efficiency of this detector is not as great as others. Geiger-Muller detectors are attached to scalar/rate meters which record the output pulse. These instruments are typically utilized for scans and for fixed readings. This is another instrument commonly used for smooth surfaces such as floors, walls or work surfaces.

F1c. Zinc Sulfide Scintillation Detectors

This type of survey instrument is used for the detection of alpha contamination. It operates using the process of scintillation. The window of these instruments, made of Mylar, is thin enough to allow the penetration of the alpha particle without significantly shielding it. The incident alpha particle produces a light pulse which is amplified by a photo multiplier tube and passed on to the survey meter. It operates in this manner, as opposed to the Geiger-Muller and gas proportional detectors, without a counting gas. This instrument has a range of probe effective areas ranging from 50 to 100 cm2. This instrument is typically used for smooth surfaces such as floors, walls or work surfaces.

F1d. Sodium Iodide Scintillation Detectors

These instruments are primarily used for the detection of gamma emitting radiation. As can be inferred from their name this is another instrument which uses the process of scintillation to detect radiation. This type of detector however uses thallium activated sodium iodide used to interact with the incoming radiation, in this case gamma radiation, to produce the light pulse amplified by the photo multiplier tube. This instruments response to radiation is dependent upon the energy of the incidental gamma photon, thus causing it to over respond to lower energy radiations and under respond to higher energy level radiations. Due to this behavior this instrument can not be solely used for the process recording true radiation dose or exposure limits, but this can be accomplished if the correct energy correction factor is determined and applied. The instruments over response however does it make it useful in detecting lower energy radiations that otherwise would be more difficult to detect. This type of detector is overall extremely sensitive and measures radiation levels in the range or micro Roentgens per hour (uR/hr). This is the primary instrument used for surface scans of surfaces such as floors and large outdoor areas.

F.2 Laboratory Instrumentation

Laboratory instrumentation for the purpose of radiological remediation includes instruments to detect small quantities of radionuclides in media such as soil or water, which cannot be detected by hand held instrumentation. Additionally, the use of this type of instrumentation augments the analysis performed by the hand held instrument. Radioisotopic analysis, either gamma ray or alpha particle make up the majority of laboratory analysis. Other widely used laboratory analysis is the use of alpha/beta gas flow proportional counters used for detection of removable radioactivity for counting wipe or smears and in counting gross measurements in other media such as soil or water.

F2a. High Pure Germanium Detectors

Germanium detectors are semiconductor diodes having a p-I-n structure in which the intrinsic (I) region is sensitive to ionizing radiation, particularly x rays and gamma rays. Under reverse bias, an electric field extends across the intrinsic or depleted region. When photons interact with the material within the depleted volume of a detector, charge carriers (holes and electrons) are produced and are swept by the electric field to the P and N electrodes. This charge, which is in proportion to the energy deposited in the detector by the incoming photon, is converted into a voltage pulse by an integral charge sensitive preamplifier.

F2b. Liquid Scintillation Detector

Liquid scintillation counting (LSC) is a technique applicable to the detection of radiations with very low penetrations, e.g. - and low energy - particles (3H and 14C - the -particles from 3H have an average energy of 6keV and are capable of penetrating a few m in solid or liquid). These radiations are incapable of penetrating the walls of most detectors and if the source is in the form of a solid, then only very thin films will allow counting with a minimum of self-absorption.

A liquid scintillation sample consists of three components: radioactive material, a solvent and a fluor (or scintillator) so that the emitted radiations are in intimate contact with the detector. Energy is forfeited by the radiation and transferred to the fluor molecule which emits a photon detectable by photo multiplier tubes (PMT).

F2c. Alpha Spectroscopy Detectors

The detectors used in Alpha Spectroscopy are semiconductors. These devices operate by transferring the energy of an incident charged particle to the valence shell electrons of atoms within the material. The electrons which receive the energy are excited, and if the energy is greater than the band gap of the electron (~1 eV) it is moved to the conduction band.

Semiconductor detectors are based on a p-n junction (i.e. a diode) which means that a dead layer exists between the poles of a circuit into which it is placed. When a voltage is applied across the detector in the reverse direction (in other words the diode is placed in a circuit backwards), no current will flow until something happens to change the energy level of the dead band. Charges induced by the excitation of electrons in the material (i.e., from an alpha particle) are collected in and cross the dead layer. This pulse of conductivity results in a pulse of current in the circuit to which the detector is connected. The size of the pulse (conveniently enough) is proportional to the amount of energy deposited by the charged particle which hit the detector.

The radius of an alpha particle is ~8-9E-13 cm (for N=150 to 240), with a mass of 4.031882 amu and an energy of 3 to 5 MeV (typical, with some as high as 10). Alpha particles are thus fat and energetic (as particles go), and are prone to collide with anything in their path. To minimize the number of particles which undergo partially elastic collisions on their way to the detector, alpha spectrometers are operated under vacuum.

There are two types of Alpha detectors commonly in use for spectroscopy. Both of these consist of a circular wafer of silicon mounted in a cylindrical metal housing with a snap ring front bezel and a threaded rear contact.

The first type of detector is a silicon surface barrier detector (SSB). These detectors have a thin layer of gold vacuum evaporated onto the front surface, and an aluminum layer vacuum evaporated onto the rear for an electrical contact.

The second type of detector is the ion implanted detector which has an ion diffused front surface instead of a gold flashed surface. The ion implanted detector has the advantages of being "rugedized" (cleanable), and a higher threshold temperature of resolution degradation. The first of these advantages gives the ion implanted detector a longer useful life because increased background due to recoil contamination can be removed. The second advantage means that the resolution observed in a normal laboratory setting (where the detector sits in a hot NIM bin) is much better than an SSB.

Bias voltages for both of these types of detectors are typically in the range of 30 - 100 VDC.

This is an instrument utilized for identifying and quantifying the activity of alpha emitting radioanuclides. These typical alpha emitting radionuclides include the isotopes of Uranium, thorium, plutonium, polonium, americium and radium. Typically samples of water or soil are analyzed with this instrument. Use of this instrument requires that the samples be chemically prepared prior to analysis, which leads to the relatively high cost of this form of analysis. Alpha spectroscopy costs on average $250-$400 per analysis, leading it to become cost prohibitive on a large scale.

F2d. Gas-Flow Proportional Counter

This is a laboratory instrument used for determining gross alpha or beta activity of water, soil, air filters, or smears. When used for the gross analysis of soil and water, this instrument can be used to determine if further analysis is required. This manner of use highly cost effective. The other major purpose of this instrument is for the analysis of removable contamination on smears, a piece of filter paper rubbed over a surface and counted in an instrument, such as this to detect removable surface activity. These types of detectors have the capability to count tens to hundreds of samples automatically, and has the capability print or when connected with a computer save data. Like the portable

gas-flow proportional counters, the laboratory version has the capability to simultaneously count for alpha and beta radiations.

F2e. Additional Instrumentation

Additional survey instrumentation commonly used for decommissioning surveys includes, portable gamma spectrometers, pressurized ion chambers, liquid scintillation counters, FIDLERs, and other new emerging technological devices.

Portable gamma spectrocopy gives the user the ability to identify and quantify radioactive material in the field, *in situ*. There are two major difficulties with this however. The portable germanium multichannel analyzer requires a dewer of liquid nitrogen to accompany the instrument and the other type of portable multichannel analyzer the sodium iodide type, has a relatively poor resolution. Although these instruments are great tools in detection, due to many quality control issues these are not used for final status surveys.

Pressurized ion chambers or PICs can be used to measure "real time" direct gamma-ray levels and record exposure rates. This field instrument is highly accurate for measuring gamma exposure rates in air and for correcting for the energy dependence of other instruments and is generally used in conjunction with soil sampling to judge the success of remediation efforts.

Liquid scintillation counters are laboratory instruments which count concentrations of low energy beta emitters such as Carbon-14 and Tritium (H-3) in water and soil.

The Field Instrument for Detecting Low Energy Radiation (FIDLER), where low enegry X or gamma rays are used to detect the presence of Plutonium or Americum-241. This is a scintillation type instrument. A problem with this instrument is the fragility of the detector window.

One of the new emerging technologies is a floor monitor developed by Shonka Reseach Associates, Inc. that uses position sensitive proportional counter-based radiation detectors on a wheeled cart based device. The position-sensitive proportional counter allows one detector to act as the equivalent of hundreds of individual detectors, allowing the collection of vast amounts of data as the detector moves over the surface being counted. Additionally, all data collected is relayed to a computer which logs survey locations and count rate data on a real time basis.

This is not an inclusive listing of instrumentation which can be used for radiological remediations nor does it represent a list of instruments which are endorsed by the Connecticut DEP.

F.3 The Minimum Detectable Concentration Concept of Radiation Survey Instruments in Field Use

In 1995 the Oak Ridge Institute for Science and Education, Brookhaven National Laboratory and the Nuclear Regulatory Commission performed a study to determine the capabilities of radiation detection instrumentation utilized for the decommissioning process. A literature search was performed on the detection sensitivity capabilities of portable survey instruments. In general, the information on the minimum detectable concentrations (MDC) in the literature were for optimum capabilities under conditions of low-background, smooth clean surfaces and experienced survey personnel. Additional studies were determined to be necessary to developed comprehensive information, relative to instrument performance under actual field conditions.

The major emphasis of the study was the measure of detection sensitivity for field survey instruments for both direct and scanning modes of operation. The parameters that were studied for their effects on the detection sensitivity of field instruments included variables that determine the instruments MDC. These include the size of the probe surface area, Radionuclide energy, window density thickness, source to detector geometry. Also included were variables which that could effect the detection sensitivity of the instruments such as various surfaces types and coatings, included painted, scabbled or wet surfaces.

Scan MDCs were evaluated for both building surfaces and land areas. The innovative approach use to determine scan MDCs coupled the detector and contamination characteristics with human factors.

F3a. Methodology

As previously stated, during radiological surveys in support of decommissioning, field instruments are generally used to scan the surface areas for elevated direct radiation, and to make direct measurements of total surface activity at particular locations. Although the surface scans and direct measurements can be performed with the same instruments, the two procedures have very different MDCs. Scanning can have a mush higher MDC than a direct measurement. Depending on scanning speed, distance of the probe to the surface and other instrument factors. The scanning MDC is also affected by human use factors.

Studies were performed primarily at Oak Ridge Institute for Science and Education (ORISE) facilities in Oak Ridge, Tennessee. A measurement hood, constructed of Plexiglas, provided a controlled environment in which to obtain measurements with minimal disturbances from ambient airflow. The hood was equipped with a barometer and thermometer to measure ambient pressure and temperature within the chamber. Measurements were performed within the hood using a detector-source jig to ensure that the detector-to -source geometry was reproducible for all parameters studied. Various

field conditions were simulated under well-controlled and reproducible conditions. Special sources were constructed and characterized in ORISE laboratories to meet specific objectives of the study. On the basis of the empirical results obtained from the studies information was derived that would indicate instrument response as a function of source energy, geometry, background radiation level, and other parameters, including source-to-detector distance, window density thickness, and density thickness of overlaying material.

Detection limits for field survey instrumentation are an important criterion in the selection of appropriate instrumentation and measurement procedures. Instruments and measurement procedures must be capable of detecting residual activity at the regulatory release criteria known as the derived concentration guidance limit (DCGL). One may demonstrate compliance with decommissioning criteria by performing surface activity measurements and directly comparing the results to the surface activity DCGLs. However, prior to any measurements being performed, the survey instrument and measurement procedures to be used must be show to possess sufficient detection capability relative to the surface activity DCGLs. The detection limit of the survey instrument must be less than the appropriate surface activity DCGL.

The measurement of residual radioactivity during surveys in support of decommissioning often involves measurement of residual radioactivity at near background levels. Thus, the minimum amount of radioactivity that may be detected by a given survey instrument and measurement procedure must be determined. In general, the minimum detectable concentration (MDC) is the minimum activity concentration on a surface or within a material volume, that an instrument is expected to detect with a 95% confidence. It is important to note, however that this activity concentration or MDC is determined *a priori*, that is, before survey measurements have been taken.

As generally defined the detection limit which may be a count or count rate is independent of field conditions such as scabbled, wet or dusty surfaces. That is the detection limit is based on the number of counts and does not necessarily equate to measured activity under field conditions. These field conditions do, however affect the instrument's detection sensitivity or MDC. Therefore, the terms MDC and detection limit should not be used interchangeably.

In the study that was performed by ORISE, the MDC corresponded to the smallest activity concentration measurement that was practically achievable with a given instrument and type of measurement procedure. That is, the MDC depends not only on the particular instrument characteristics (instrument efficiency, background, integration time) but also on the factors involved in the survey measurement process, which include surface type, source to detector geometry and source efficiency (backscatter and self-absorption).

F3b. MDC Fundamental Concepts

It is important at this point to discuss concepts of the MDC. The MDC concepts discussed derive from statistical hypothesis testing, in which a decision is made on the presence of activity, specifically, a choice is made between the null hypothesis (H_o) and the alternative hypothesis (H_a). The null hypothesis is generally stated as no net activity present in the sample or observed counts are not greater than background. The alternative hypothesis states that the observed counts are greater than background, and thus, that net activity is present. This derives the following statement:

H_o: No net activity is present in the sample, and

H_a: Net activity is present in the sample.

A first step in understanding the MDC concept is to consider an appropriate blank (background) distribution from the medium to be evaluated. Currie (1968) defines the blank as the signal resulting from a sample which is identical, in principle, to the sample of interest, except that the residual activity is absent. This determination must be made under the same geometry and counting conditions as used for the sample. In the context of remediation surveys and example of this medium may be an unaffected concrete surface that is considered representative of the surface to be measured in the remediaiton. It should be noted that the terms blank and background are interchangeable.

In the statistical framework, one must consider the distribution of counts obtained from measurements of the blank, which may be characterized by a population mean ($\ddot{y}_B$) and standard deviation ($\ddot{y}_B$). Now consider the measurement of a sample that is known to be free of residual activity. This zero-activity (background) sample has a mean count (C_B) and standard deviation (S_B). The net count (and, subsequently, residual activity) may be determined by subtracting the blank count from the sample counts. This results in a zero-mean count frequency distribution that is approximately normally distributed. The standard deviation of this distribution, $\ddot{y}_o$, is obtained by propagating the individual errors (standard deviations) associated with both the blank ($\ddot{y}_B$) and the zero-activity samples (S_B).

As written in the formula:

ÿO = ÿ ÿ B2 + S B2

A critical level may then be determined from this distribution and used as a decision tool to decide when activity is present. The critical level, L_C, is the net count in a zero-mean count distribution having a probability, denoted by ÿ, of being exceeded. It is common practice to set ÿ equal to 0.05 and to accept a 5% probability of incorrectly concluding that activity is present when it is not. That is, if the observed net count is less than the critical level, the surveyor correctly concludes that no net activity is present. When the net count exceeds L_C, the null hypothesis is rejected in favor of its alternative, and the surveyor falsely concludes that net activity is present in the blank sample. It should also be noted that the critical level, L_C, is equivalent to given probability of committing what is referred to as a Type I error (false positive detection). The expression for L_C is generally given as:

LC = ÿÿ ÿÿ

where $K_{ÿ}$ ÿÿ is the value of the standard normal deviate corresponding to a one-tailed probability level of 1-ÿ. As stated previously, the usual choice for ÿ is 0.05, and the corresponding value for $K_{ÿ}$ is 1.645. For an appropriate blank counted under the same conditions as the sample, the assumption may be made that the standard deviations of the blank and zero-activity sample are equal. Therefore the critical level may be expressed as:

$LC = 1.645\ ÿ\ 2\ \ S_B = 2.33\ S_B$

The L_C value determined above is in terms of net counts, and as such, the L_C value should be added to the background count if comparisons are to be made to the directly observable instrument gross count.

The detection limit, L_d is defined to be the number of mean net counts obtained from samples for which the observed net counts are almost always certain to exceed the critical level. It is important to recognize the L_D is the mean of a net count distribution. The detection limit is positioned far enough above zero so that there is a probability, denoted by ÿ**,** that the L_D will result in a signal less than L_C . It is common practice to set ÿ equal to 0.05 and to accept a 5% probability of incorrectly concluding that no activity is present, when it is indeed present, the concept known as a Type II error. More simply stated, the surveyor has already agreed to conclude that no net activity is present for an observed count that is less than the critical level, however, an amount of residual activity that would yield a mean net count of L_D is expected to produce a net count less than the critical level 5% of the time. This is equivalent to missing residual activity when it is present.

The expression for L_D is generally given as:

$$LD = CS + K\ddot{y} \quad \ddot{y}D$$

where Kÿ is the value of the standard normal deviate corresponding to a one-tailed probability level of 1-ÿ for detecting the presence of net activity, and $\ddot{y}_D$ is the standard deviation of the net sample count (C_S) when C_S equals L_D. The detection limit may be written as follows, recognizing that C_S equals LP :

LD = Cs + (CB - ÿB)

As stated previously, the usual choice for ÿ is 0.05, and the corresponding value for ÿÿ is 1.645. If the assumption is made that ÿD is approximately equal to the standard deviation of the background, than for the case of paired observations of the background and sample ÿÿ2 equals 2S*B*2. Following algebraic manipulation, the detection limit may be expressed as:

$$LD = 2.71 + 4.65\ S_B$$

The assumption that the standard deviation of the count (ÿD) is approximately equal to that of the background greatly simplifies the expression for L_D, and is usually valid for total counts greater than 70 for each sample and blank count (Brodsky 1992). Brodsky has also examined this expression and determined that in the limit of very low background counts, S*B* would be zero and the constant 2.71 should be 3, based on Poisson count distribution. Therefore the expression for detection limit becomes:

$$LD = 3 + 4.65\, S_B$$

The detection limit calculated above may be stated as the net count having a 95% probability of being detected when a sample contains activity at L_D and with a maximum 5% probability of falsely interpreting sample activity as activity due to background (false negative or Type II errors).

The MDC of a sample follows directly from the detection limit concepts. It is a level of radioactivity, either on a surface or within a volume of material, that is practically achievable by an overall measurement process. The expression for MDC may be given as follows:

$$MDC = \frac{(3 + 4.65\, SB)}{KT}$$

Where *K* is a proportionality constant that relates the detector response to the activity level in a sample for a given set of measurement conditions and *T* is the counting time. This factor typically encompasses the detector efficiency, self-absorption factors, and probe area correction.

Another expression of the MDC equation was derived assuming equivalent or paired observations of the sample and blank, in contrast to the MDC expression that results when taking credit for repetitive observations of the blank. There is some debate concerning the appropriateness of taking credit for repetitive observations of the blank, considering the uncertainties associated with using a well-known blank for many samples when there can be instrument instabilities or change in the measurement process that may be undetected by the surveyor (Brodsky & Gallaghar 1991). Therefore, it is desirable to obtain repetitive measurements of background, simple to provide a better estimate of surface activity. Thus, the background is typically well known for purposes other than reducing the corresponding MDC, such as to improve the accuracy of the background value. The expression for MDC that will be used throughout this report will be given as follows:

$$MDC = 3 + 4.65\ddot{y}\sqrt{\frac{C_B}{KT}}$$

Where C_B is the background count in time, *T*, for paired observations of the sample and blank. For example, if ten 1-minute repetitive observations of background were performed, C_B would be equal to the average of the ten observation and *T* is equal to 1 minute. The quantities encompassed by the proportionality constant, *K*, such as the detection efficiency and probe geometry, should also be average, ÿwell knownÿ values for the instrument. For making assessment of MDC for surface activity measurements, the MDC is given in units of disintegrations per minute per 100 square centimeters (dpm/100 cm2).

One difficulty with this MDC expression is that all uncertainty is attributed to Poission counting errors, which can result in an overestimate of the detection capabilities of a measurement process. The proportionality constant, *K*, embodies measurement parameters that have associated uncertainties that may be significant as compared to the Poisson counting errors.

F3c. Variables Affecting Instrument Minimum Detectable Concentrations

Before the MDC for a particular instrument can be determined, it is necessary to introduce the expression for total surface activity per unit area of the calibration source. Currently, surface contamination is assessed by converting the instrument response to surface activity using one overall total efficiency. This is not a problem provided that the calibration source exhibits characteristics similar to the surface contamination-including radiation energy, back scatter effects, source geometry, and self-absorption. In practice this is hardly ever the case: more likely, total efficiencies are determined with a clean, stainless steel source, and then those efficiencies are used to measure contamination on a dust-covered concrete surface.

An instrument efficiency is defined as the ratio between the net count rate of the instrument and the surface emission rate of a source for a specified geometry. The surface emission rate is defined as the number of particles of a given type of radiation above a given energy emerging from the front face of the source per unit time. The

surface emission rate is the 2ÿ particle fluence that embodies both the absorption and scattering process that affect the radiation emitted from the source.

The instrument efficiency is determined during calibration by obtaining a static count with the detector over a calibration source that has a traceable activity or surface emission rate or both. In many cases, it is the source surface emission rate that is measured by the manufacturer and certified as National Institute of Standards and Technology (NIST) traceable. The source activity is than calculated from the surface emission rate based on assumed back scatter and self absorption properties of the source. The theoretical maximum value of instrument efficiency is one.

The source efficiency, is defined as the ratio between the number of particles of a given type emerging from the front face of a source and the number of particles of the same type created or released within the source per unit time. The source efficiency takes into account the increased particle emission due to back scatter effects, as well as the decreased particle emission due to self-absorption loss. Source efficiencies may either be determined experimentally or simply selected from guidance documents.

F3d. Radionuclide Sources for Calibration

For accurate measurements of total surface activity, it is essential that field instruments be calibrated appropriately. The MDC of an instrument depends on a variety of parameters, one of which involves the selection of calibration sources. Calibration sources should be selected that emit a radiation with energies similar to those expected to be encountered in the field. As an example, both the uranium and thorium series emit a complex decay scheme of alpha, beta and gamma radiations. Calibration to a single Radio nuclide must carefully be assessed to ensure that is representative of the detectionÿs response to these decay series.

F3e. Source to Detector Distance

The distance between a source and the detector is another factor that may effect the instrument efficiency and, thus the MDC. As can be expected the greatest reduction in detector response per increased distance from a source will be obtained from alpha and low-energy beta emitting radionuclides such as Ni-63 and C-14. The practicality of this must be realized by the deviations in instrument response that results when the source to detector distance is slight (such as during an instrument calibration) as opposed to the detector to surface spacing maintained during field measurements of surface activity. It is not uncommon for detectors to be held approximately twice the distance from a surface such as scabbled concrete (2 cm spacing), as from the distance it was from its calibration source (1 cm spacing).

F3f. Source Geometry Factors

The source geometry must be considered in determining the instrument MDC. The detector's response may be influenced, in part by the contaminants' distribution on the surface being surveyed. If the contamination to be surveyed is considered to be widely distributed, then the detector should be calibrated to a distributed or extended source. Conversely, if the contamination is characterized by localized spots of surface contamination that may closely approximate a point source, then the calibration source should be similar to a point source geometry. It would be conservative however, to use the instrument efficiency obtained from a distributed source geometry for all surface activity measurement locations.

F3g. Ambient Background Count Rate

The effects of ambient background (in particular, relatively high ambient background) on the calculated MDC and measured activity concentration are quite measurable. In general as the ambient background increases, and the ratio of the calculated MDC to the actual activity concentration present approaches unity, the uncertainty in the measured activity increases. Due to variations in ambient background great caution must be exercised when making measurements that are close to the MDC, because substantial uncertainties may be associated with the measurements.

F.4 Variables Affecting Minimum Detectable Concentrations in the Field

Surface activity levels are assessed by converting detector response, through the use of a calibration factor, to radioactivity. Once the detector has been calibrated and an instrument efficiency established, several factors must still be carefully considered when using that instrument in the field. One of these factors involves the background count rate for the particular surface. The surveyor needs to know how and to what degree these different field conditions can affect the sensitivity of the instrument. A particular field condition may significancy affect the usefulness of a particular instrument such as wet surfaces for alpha measurements or scabbled surfaces for low energy beta measurements.

As previously stated, one of the more significant implicit assumptions commonly made during instrument calibration and subsequent use of the instrument in the field is that the composition and geometry of contamination in the field is the same as that of the calibration source. This may very well not be the case, considering that may calibration sources are fabricated from materials different from those that comprise the surfaces of interest in the field, activity plated on a metallic disc. This difference usually manifests itself in the varying of backscatter characteristics of the calibration and field surface materials.

F.4a Background Count Rates for Various Materials

Several different types of surface materials may be encountered in a facility undergoing decommissioning. Among the typical surface materials that may be encountered include: brick, ceramic block, ceramic tile, concrete block, unpainted drywall, vinyl floor tile, linoleum, steel, pine wood treated with commercially available water sealant product, and untreated pine. Since the detectors MDC varies directly with the background count rate, the lowest MDCs will be obtained in materials like linoleum, carbon steel, concrete block, drywall and wood. Conversely the higher backgrounds and higher MDCs will be in material like brick, ceramic materials, and soil. These materials having higher backgrounds and higher MDCs because of the natural radioactive material which they are made of. It is important to assess the alpha background of various surface materials due to the wide range in MDC values which will be represented. This is in contrast to the beta MDCs which are fairly consistent for all materials with the notable exception of brick and ceramics. In application it is important that the surveyor establish specific material backgrounds that are representative of the surface types and field conditions.

F.4b Effects of Surface Conditions on Detection Sensitivity

The conversion of the surface emission rate to the activity of the contamination source is often a complicated task that may result in significant uncertainty if there are deviations from the assumed source geometry. For example, consider the measurement error associated with an alpha surface activity measurement on a rough surface, such as scabbled concrete, where substantial attenuation reduces the count rate as compared to the calibration performed on the smooth surface of a National Institute of Standards and Technology traceable source.

Instrument response can be affected by energy response to the source, backscatter from media, and self-absorption of radiation in the surface. It is possible that relatively low efficiencies obtained for concrete surfaces is due to the penetration of radioactive contamination into the surface material and the resultant self-absorption. This porosity effect is also evident in untreated wood. Both backscatter and self-absorption effects may produce considerable error in the reported surface activity levels if the field surface is composed of material significantly different in atomic number of the calibration source used to calibrate the instrument. Therefore it is important to consider the effects that result when the calibration source has backscatter and self-adsorption characteristics different from the field surfaces measured.

F.4c Human Performance and Scanning Sensitivity

Scanning is performed during radiological surveys in support of decommissioning to identify the presence of any locations of elevated direct radiation. The probability of detecting residual contamination in the field is affected not only by the sensitivity of the survey instrumentation when used in the scanning mode of operations, but also by the

surveyor's ability. The surveyor must decide whether the signals represent only the background activity, or whether they represent residual contamination in excess of background.

The minimum detectable concentration of a scan survey (scan MDC) depends on the intrinsic characteristics of the detector (efficiency, window area, etc.), the nature (type and energy of emissions) and relative distribution of the potential contamination, point versus distributed source and depth of contamination, scan rate and other characteristics of the surveyor. Factors which may affect the surveyor's performance include, the surveyor's *a priori* expectation of the likelihood of contamination present. For example if the surveyor believes that the potential for contamination is very low, as in an unaffected area, a relatively large signal may be required for the surveyor to conclude that contamination is present.

An overview of the process used to determine scan MDC is given: Signal detection theory provides a framework for the task of deciding whether the audible output of the survey meter during scanning was due to background or signal plus background levels. An index of sensitivity (*d*), that represents the distance between the means of the background and background plus signal, in units of their common standard deviation, can be calculated for various decision errors-Type I (ÿ) and Type II error (ÿ).

As an example, for a correct detection or true positive rate of 95% (1-ÿ) and a false positive rate (ÿ) of 5%, *d* is 3.29. The index of sensitivity (d) is independent of human factors, and therefore, the ability of an ideal observer, may be used to determine the minimum (d) that can be achieved for particular decision errors. The ideal observer makes optimal use of the available information to maximize the percent correct responses, providing an effective upper bound against which to compare actual surveyors. Computer simulations and field experimentation can then be performed to evaluate the surveyor efficiency (p) relative to the ideal observer. The resulting expression for the ideal observer's minimum detectable count rate (MDCR), in counts per minute, can be written as follows:

$$MDCR = d * \ddot{y}b_i * (60/I) = Is * (60/I)$$

where

MDCR= minimum detectable (net) count rate in counts per minute, can be written

b_i = background counts I the observation interval.

Is = minimum detectable number of net source counts in the observation interval, and

I = observational interval (in seconds), based on the scan speed and areal extent of the contamination.

Scan MDCs are determined from the MDCR by applying conversion factors to obtain results in terms of measurable surface activities and soil concentrations. As an example, the scan MDC for a structure surface can be expressed as

Scan MDC = MDCR

ÿP ÿi ÿs Probe area

100 cm2

A common rule for scanning sensitivity is base on the surveyor being able to detect three times the background level for low count rates. The specification of detectable levels is complicated by the difficulty of defining detectable as applied to the performance of the surveyor. For example, guidance on scanning capabilities is given in ANSI Standard 13.12, *Control of Radioactive Surface Contamination on Materials, Equipment and Facilities To Be Released for Uncontrolled Use*. This document states that the scanning speed shall be slow enough to ensure that a small-diameter source is detected with a 67% probability. However, the specification of scan MDC requires the ability to detect false positives as well. In theory, any correct detection rate can be achieved for any source intensity if the number of false positives permitted is unlimited.

A few attempts to quantify scanning sensitivity experimentally have been reported. Scanning MDCs have been evaluated for both alpha and beta instrumentation under varying background conditions using a semi-empirical approach (Goles et al.1999). MDCs were defined as that activity that would be detected 67% of the time under standard survey condition. The instruments evaluates were for alpha detection , a 50 cm2 portable alpha monitor, a 100-cm2 large-area scintillation monitor, and a 100 cm2 gas proportional counter; for beta/gamma detection , a pancake GM probe, a 100 cm2 large-area scintillation monitor, and a 100-cm2 gas proportional counter. The test procedure involved maintaining a scan rate of 5 cm/s, with a scan height held as 0.64 cm. Alpha sources were 2.54-cm-diameter electroplated sources; beta/gamma sources consisted of point source geometries and uniformly dispersed geometries. The MDC for alpha activity was defined as the amount of activity that produces on count as the detector passes over the surface (alpha background was considered to be zero) and the MDC for beta/gamma activity was determined for different background activities (e.g., 50, 250, and 500 cpm), based on whether it could be detected 67% of the time. For the most part, the researchers concluded that detectors were more sensitive to point sources than to area sources. The reported scanning sensitivities for the GM detectors demonstrated that activities producing net instrument responses of 305, 310, and 450 could report the data within the 67% range however, Goles cautioned that the "*data are highly idealized and that the performance of these instruments will differ under field conditions."*

F.4d Signal Detection Theory

Signal detection theory provides a means for characterizing the performance of surveyors performing scans. The theory relies on the statistical decision technique and applies to the detection of signals in background noise by surveyors. Personnel conducting radiological surveys for residual contamination at decommissioning sites must interpret the audible output of a portable survey instrument to determine when the signal ("clicks") exceeds the background level by a margin sufficient to conclude that contamination is present. It is difficult to detect low levels of contamination because both the signal and the background vary widely.

In abstract terms the task of personnel conducting radiological scan surveys can be briefly characterized as follows. The condition of the surface being scanned is represented to the surveyors by samples from random processes (Poison distributed counts). Furthermore, the samples are limited in size for practical reasons (scan speed). On the basis of the samples, the surveyors must decide whether they have sampled the distribution of activity associated with contaminated surface or an uncontaminated surface (background only). The concepts and methods of signal detection theory are well suited to the analysis of performance on such tasks, and require the specification of the acceptable Type I and Type II error rates. The information available to the observer can arise from either noise alone or from signal-plus-noise and can be represented by two (typically overlapping) probability density distribution. The task of the observer is to indicate whether an increase in survey instrument output arose from a "noise alone" or a "noise plus signal" event. To make this decision, a criterion must be established at some point along the continuum. Once the criterion point is set, any measurement greater than the criterion will be interpreted as a contamination event.

F.4e Human Factors and Scanning

According to statistical decision theory, the *a priori* probabilities of the events and the values and costs associated with the outcomes will influence the placement of the criterion, which is a human factors effect. Thus the detection of a signal in a noise background is determined not only by the magnitude of the signal relative to the background *(d)*, but also by the willingness of the surveyor to report that a signal is present.

The surveyor's decision is itself influenced by a variety of factors, including the relative costs of misses and false positives, and the surveyor's assumptions regarding the likelihood of contamination being present. The principal implication of the signal detection theory perspective from scanning performance is that, in view of the nature of the task one must consider false positive rates as well as correct detection rates in order to meaningfully characterize human performance. The rewards or penalties associated with various outcomes influence the subject's responses. In the context of scanning surveys, these factors may affect performance significantly. Surveyors are typically motivated to detect all instances of possible contamination (to maximize the correct detection rate). However, there are costs associated with incorrectly identifying areas as being

contaminated (e.g., making follow-up static measurements or collecting and analyzing samples). The placement of the criterion reflects a balance between these two influences. Observer's estimates of the likelihood/frequency of signals will also influence their willingness to decide that a signal is present. Other things being equal, then, a surveyor will adopt a less-strict criterion when examining areas in which contamination may be expected. Similarly, surveyor's criteria may be more strict when examining areas in which they don't expect contamination to be present.

In practice, surveyors do not make decisions on the basis of a single indication. Rather, upon noting an increased number of counts, they pause briefly and than decide whether to move on or take further measurements. Thus, scanning consists of tow components: continuous monitoring and stationary sampling. In the first component, characterized by continuous movement of the probe, the surveyor has only a brief "look" at potential sources, determined by the scan speed. The surveyor's criterion (i.e., willingness to decide that a signal is present) at this stage is likely to be liberal, in that the surveyor should respond positively on scant evidence, since the only "cost" of a false positive is a little time. The second component occurs only after a positive response was made at the first stage. It is marked by the surveyor interrupting his scanning and holding the probe stationary for a period of time, while comparing the instrument output signal during that time to the background counting rate. Owing to the longer, observation interval, sensitivity is relatively high. For this decision the criterion should be more strict, since the cost of a "yes" decision is to spend considerably more time taking a static measurement or sample.

F.4f The Ideal Observer Paradigm

If the nature of the distributions underlying a detection decision can be specified, it is possible to examine the performance expected of an ideal observer, i.e., one that makes optimal use of available information to achieve a specified goal, such as to maximize the percent correct responses. This is important because it allows the basic relationships among important parameters like background rate and length of observation to be anticipated, and it provides a standard of performance (an upper bound) against which to compare performance of actual surveyors.

The audio output of a survey instrument represents randomly occurring events. It will be assumed that the surveyor is a "counting" observer (one that makes a decision about the presence or absence of contamination based on the number of counts occurring in a given period of time. This number will have a Poisson distribution , and the mean of the distribution will be greater in the presence of contamination than when only background activity is present. When the intensity of activity associated with contamination is low, as it often is during final status surveys, these distributions will overlap. The ideal observer decides that contamination is present if the number of counts is greater than x, where the criterion value x is chosen to maximize percent correct.

Together, the results of the simulation studies indicate the extent to which human limitations and the nature of the scanning task reduce the efficiency of the surveyor relative to an ideal observer. The ideal observer attempting to detect 180 cpm (gross) in a background of 60 cpm (i.e., a source three times background), in a 4-second observation interval, will be capable of correctly detecting the source roughly 91% of the time with about 5% false alarms (determined from tabulated values of the cumulative Poisson distribution). This corresponds to a *d* value of roughly 3. In the defined interval rating task, using the same background and source values, a typical surveyor detected about 90% of the sources with a false positive rate of 14% for a *d* value of about 2.4. In the undefined interval procedure, under the same conditions the performance of the same surveyor yielded a *d* value of 1.8. This demonstrates that one: even under ideal circumstances humans do not behave as perfect counting devices (they are less efficient than the ideal observer), and two: in scanning, where observation intervals are not defined, the efficiency of the surveyor (relative to the ideal observer) declines further.

F.4g Estimation of Scan Minimum Detectable Count Rate (MDCRs)

The changes in detectability as a function of background level and observation interval (as determined in simulation studies using adaptive level adjustment) were consistent with theoretical predictions, i.e., the number of source counts required to yield a constant level of performance was proportional to the square root of the number of background counts in the observation. Therefore, if performance is known to be acceptable for a given background/source condition and observation interval, it is possible to estimate source levels expected to support similar performance under other conditions.

F.4h Scan MDCs for Structure Surfaces and Land Areas

The survey design for determining the number of data points for areas of elevated activity (as in the MARSSIM guidance) depends on the scan MDC for the selected instrumentation. In general, alpha or beta scans are performed on structure surfaces to satisfy the elevated activity measurements survey design, while gamma scans are performed for land areas.

The scan MDC is determined from the minimum detectable count rate (MDCR) by applying conversion factors that account for detector and surface characteristics and surveyor efficiency. As previously discussed the MDCR accounts for the background level performance criteria *(d),* and observation interval. The observation interval during scanning is the actual time that the detector can respond to the contamination source-it depends on the scan speed, detector size in direction of scan and size of the hot spot. The greater the contamination source effective area, and slower the scan rate, the greater the observation interval. Because the actual areal dimensions of potential hot spots in the field cannot be known *a priori*, it becomes necessary to postulate a certain hot spot area (e.g., perhaps 50 to 200 cm^2), and then to select a scan rate that provides a reasonable

observation interval. Finally, the scan MDC for structure surface may be calculated as follows:

Scan MDC = *MDCR*

ÿÿ ÿi ÿs 100 cm2Probe area

where

ÿi = the instrument efficiency, and

ÿs= the surface efficiency

As an example, the scan MDC (in dpm/100 cm2) for Tc-99 on a concrete surface may be determined for a background level of 300 cpm and a 2-second observation interval using a hand-held gas proportional detector (126 cm2 probe area). Using a surveyor efficiency of 0.5, and assuming instrument and surface efficiencies of 0.36 and 0.54, respectively, the scan is calculated as follows:

Scan MDC = 130

ÿ0.5 (0.36) (0.54) (1.26) = 750 dpm/100cm2

The scan MDC above may be compared to the static MDC (one minute count) for the same detector of approximately 340 dpm/100 cm2 . The above scan MDC can also be calculated using a faster scan rate, such that yields only a 1-second observation interval. Assuming other parameters in the example remain the same the yielded scan MDC = 1,070 dpm/100 cm2. The scan MDC may also be calculated for a higher background level (400 cpm) and a 1-second observation interval which yields a MDC of 1,230 dpm/100 cm2.

Now consider an example to determine the scan MDC for a GM detector (20 cm2) that is used to scan a concrete wall potentially contaminated with Tc-99-in a background of 60 cpm and with a 2-second observation interval. Assuming instrument and surface efficiencies of 0.19 and 0.52 respectively, the scan MDC is calculated as follows:

Scan MDC = 60

ÿ0.5 (0.19) (0.52) (0.20) = 4,300 dpm/100 cm2

F.4i Scan MDCs for Land Areas

The scan MDC (in pCi/g) for land areas is based on the area of the hot spot, depth of the hot spot, and the radionuclide (energy and yield of gamma emissions). It is generally assumed that NaI scintillation detectors are used for scanning land areas.

An approach used to determine scan MDCÿs for land areas follows. The NaI scintillation detector background level and scan rate (observation interval) are postulated, and the MDCR for the ideal observer, for the ideal observer, for a given level of performance, is obtained. A surveyor efficiency is selected, and then it is necessary to related the surveyor MDCR to a Radionuclide concentration in soil (in pCi/g). This requires two steps, first the relationship between the detectorÿs net count rate to net exposure rate (cpm/ÿR/h) is established and second the relationship between the Radionuclide contamination and exposure rate is determined.

For a particular gamma energy, the relationship of NaI scintillation detector count rate and exposure rate may be determined analytically (in cpm per ÿR/h). The approach to determine the gamma fluence rate necessary to yield a fixed exposure rate (1ÿ R/h) as a function of gamma energy. The NaI scintillation detector response (cpm) was then related to the fluence rate at specific energies, considering the detectors efficiency (probability of interaction) at each energy. It is then possible to obtain NaI scintillation detector versus exposure rate for varying gamma energies. It is possible to calculate the relative detector response for a specific gamma energy by multiplying the relative fluence to exposure rate by the probability of interaction. As an example with a Ludlum Model 44-10 probe Ludlum provides a value of 900 cpm per µR/h for this detector for Cs-137. Once the relationship between the NaI scintillation detector response (cpm) and the exposure rate is known the surveyors MDCR (in cpm) of the NaI scintillation detector can be related to the minimum detectable radionmuclide concentration (the scan MDC) by modeling a specific hot spot.

Modeling, using Microshield (a computer isotope specific dose modeling code) of an area of activity can be used to determine the net exposure rate produced by a Radionuclide concentration at a distance 10 cm above the source. This position is selected because it relates to the average height of the NaI scintillation detector above the ground during scanning. The following factors are considered in the modeling: ÿ Radionuclide of interest (considering all gamma emitters for decay chains) ÿ concentration of Radionuclide of interest ÿ area dimensions of hot spot

- ÿ depth of hot spot
- ÿ location of dose point (NaI scintillation detector height above the surface) ÿ density of soil

As an example to determine the scan MDC for 3% enriched uranium using a 1.5" X 1.25" NaI scintillation detector with a background of 4,000 cpm. With the desired level of performance of 95% correct detections and 60% false positive rate, results in a *d* of 1.38.

The scan rate of 0.05 m/s provides an observation interval of 1-sec (based on a hot spot of about 56 cm). The surveyor MDCR may be calculated assuming a surveyor efficiency (p) of 0.5 as follows:

(1) b_i = (4,000 cpm) * (1 sec) * (1 min/60 sec) = 66.7 counts

(2) MDCR = (1.38)* (ÿ66.7) * (60 sec/ 1 min) = 680 cpm

(3) Surveyor MDCR = 680/ ÿ 0.5 = 960 cpm

Utilizing the Microshield information to determine the count rate to exposure rate ratio in cpm/µR/hr for 3% enriched Uranium yields 2,010 cpm/µR/hr. The minimum detectable exposure rate can be calculated using the cpm/µR/hr value, as follows:

Minimum detectable exposure rate =

$$\frac{960 \text{ cpm}}{2{,}010 \text{ cpm}/\mu\text{R/hr}} = 0.478 \ \mu\text{R/hr}$$

A list of NaI scintillation detector scan MDCÿs for common radiological contaminant is provided below with the use of a 1.25" X 1.5" NaI detector:

Radionuclide	**Scan MDC (pCi/g)**
Am-241	44.6

Radionuclide	**Scan MDC (pCi/g)**
Co-60	5.8
Cs-137	10.4
Th-230	3,000
Ra-226 (In equilibrium with progeny)	4.5
Th-232 (Sum of all radionuclides in equilibrium)	28.3

Depleted Uranium (0.34% U-235)	80.5
3% Enriched Uranium	137
20% Enriched Uranium	152
50% Enriched Uranium	168
75% Enriched Uranium	188

F.4j Scan Data and Annual Exposure

The purpose and eventual outcome of measurements performed during radiological remediations is to form a conclusion that regions surveyed have meet the established release criteria. Although fixed measurements and samples are taken, the scan measurement performed is the most critical step in determination of achievement of the release goal. Scanning is performed from 10%-100% of regions being released dependent upon the degree of suspected contamination. Regions with a higher likelihood of having contamination levels approaching the release criteria or greater will receive a greater percentage of scan. It is therefore critical that the instrumentation have the detection ability to achieve the release criteria.

The release criterion is defined as either the activity in pCi per gram in soil or in dpm/100 cm2 for structures and building surfaces. Computer modeling is utilized to determine the residual concentration of activity which meets this release criteria known as the Derived Concentration Guidance Level (DCGL). Two computer codes have been developed to model parameters to determine radiation dose from residual radioactivity from radiological site remediations. They are RESRAD developed by Argonne National Laboratory for the Department of Energy and the other is the D and D code developed for the Nuclear Regulatory Commission by Sandia National Laboratory. These are complex computer codes which take many factors into effect to determine the effective dose equivalent (EDE) from external radiation plus the committed effective dose equivalent (CEDE) from internal exposure to determine the total effective dose equivalent (TEDE).

Utilizing the quantities for the scan MDCs from table 2.1 and the default parameters from the RESRAD computer code exposures can be derived. These annual exposures range from 24 mRem per year to at the first year for Cs-137 to greater than one Rem at the one thousandth year for Th-232. It should be noted that those computations were for single

isotopes only. The combination of Co-60 and Cs-137 derived 103 mRem at the first year.

F.4k Conclusion

The concepts of types of different instrumentation, background radiation, surface effects, and human performance factors have been discussed as they relate to instrumentation limitations associated with radiological release criteria. It has been shown possible to determine the capabilities of instruments utilized for scanning in radiological remediation surveys. This has shown the limitation of instrumentation and the minimum detectable concentrations of certain radionuclides when used with a NaI detector such as utilized by the Connecticut DEP's Division of Radiation. The Division of Radiation believes these data are relevant to the establishment of a radiological remediation standard. The establishment of a radiological release standard which is not technically achievable serves no purpose. The factors outlined in this report need to be considered when determining a radiological release criteria for a particular site.