A Research Report for Westinghouse Hanford Company

**A Manual for Applying the
Allowable Residual Contamination
Level Method for Decommissioning
Facilities on the Hanford Site**

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August 1988

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A MANUAL FOR APPLYING THE ALLOWABLE
RESIDUAL CONTAMINATION LEVEL METHOD
FOR DECOMMISSIONING FACILITIES ON
THE HANFORD SITE

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SUMMARY

The Allowable Residual Contamination Level (ARCL) method, as presented by Kennedy and Napier (1983a,b) was developed to assist the U.S. Department of Energy's (DOE) Richland Operations Office decommission the retired nuclear production reactors and associated facilities situated on the Hanford Site, which is located in south-central Washington state. The method is intended to help evaluate decommissioning alternatives by using a site-specific radiation scenario/exposure-pathway analysis to determine the acceptable levels of residual radioactive contaminants that can remain in a nuclear facility that is to be decommissioned in situ. The maximum allowable residual levels of radioactive material is based on an annual dose limit to potentially exposed individuals.

This report describes the modifications that have been made to enhance the original ARCL method to make it more applicable to site-specific analyses. This version considers the mixture of radionuclides present at the time of site characterization, the elapsed time after decommissioning when exposure to people could occur, and includes a calculation of the upper confidence limit of the potential dose based on sampling statistics that are developed during the site characterization efforts. The upper confidence limit of potential exposure can now be used for comparison against applicable radiation dose limits (i.e., 25 mrem/yr at Hanford). The level of confidence can be selected by the user. A wide range of exposure scenarios were evaluated; the rationale used to select the most limiting scenarios is explained.

The radiation dose factors used for the inhalation and ingestion pathways were also updated to correspond with the radiation dosimetry methods utilized in the International Commission of Radiological Protection Publications 26 and 30 (ICRP 1977; 1979a,b, 1980, 1981, 1982a,b). This simplifies the calculations, since "effective whole body" doses are now calculated, and also brings the dosimetry methods used in the ARCL method in conformance with the rationale used by DOE in developing the 25 mrem/yr limit at Hanford.

Step-by-step procedures are included in this report for the convenience of the user. Example calculations are also included in the Appendix.

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1.0 INTRODUCTION

The Allowable Residual Contamination Level (ARCL) method was developed in 1983 (Kennedy and Napier 1983a; 1983b) to assist the U.S. Department of Energy's (DOE) Richland Operations Office (RL) in decommissioning the retired nuclear production reactors and associated facilities situated on the Hanford Site, located in south-central Washington state (Figure 1.1). The decommissioning operations involve developing the detailed cost, engineering, and safety data necessary to ensure that the final disposition of the retired facilities is performed in an environmentally acceptable manner. Decommissioning a facility in situ (i.e., burying an entire or part of a facility in place) is often an economically attractive alternative. However, a major consideration is determining the amount or level of residual radioactive contamination that can be allowed to remain at a site or facility. In the consideration of decommissioning alternatives, the ARCL method assists by using a site-specific radiation scenario/exposure-pathway analysis, based on an annual radiation dose limit, to determine the acceptable levels of residual radioactive contaminants that can remain in a nuclear facility that is decommissioned in situ.

To permit full consideration of decommissioning alternatives, three specific modes of future use of the land and facilities (i.e., restricted, controlled, and unrestricted) were included in the scope of the method described by Kennedy and Napier (1983a,b). Specific radiation exposure scenarios for individuals who may use the land or facilities after decommissioning were defined and modeled for each mode of future use. For soil contamination and for radioactive material that may be buried in place under stabilized cover materials, both unconfined (surface to a depth of 1 m) and confined (subsurface at depths below 1 m) conditions were modeled by the radiation exposure scenarios.

After its development in 1983, the ARCL method was adopted as the official analysis method for evaluating decommissioning alternatives of

FIGURE 1.1. Location of the Hanford Site

facilities on the Hanford Site. (a) During development of the Hanford 100 Area Long-Range Decommissioning Plan (Adams et al. 1984) and following initial uses of the ARCL method, questions were raised about the application of the method to site-specific conditions and sampling results for the in situ decommissioning option. To answer these questions, therefore, the original ARCL method was modified, as described in this report, to account for the statistical uncertainties in the radiological sampling and site characterization program, and to make the method easier to apply to field

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- (a) Letter from Michael J. Lawrence, Manager of the Richland Operations Office of the U.S. Department of Energy, to Hanford Site Contractors, July 3, 1984.

situations. In particular, the ARCL method was modified to consider scenario/exposure-pathway enhancements for residual radioactive materials in soil, sampling statistics, facility- or site-specific modifications to the method, and adoption of a statistically based decision rule. Step-by-step procedures are also included in this report for the convenience of the user. These procedures may be used in implementing the ARCL method, or other procedures may be developed by the user to fit particular needs.

1.1 EXISTING DECOMMISSIONING STANDARDS

In 1981, Mueller, Kennedy, and Soldat conducted a review of soil contamination and existing decommissioning guidance. They concluded that it was difficult to compare the existing soil standards because each was intended for a different situation and because different units or bases were used. Most of the soil contamination information appeared to be consistent with the philosophy of maintaining exposures at levels "as low as reasonably achievable" (ALARA). The U.S. Nuclear Regulatory Commission (NRC) has provided guidance for decommissioning commercial reactors in Regulatory Guide 1.86 (U.S. AEC 1974). However, it has drawbacks in that the guidance values provided in the Guide are not directly comparable to a single dose limit and they are difficult to detect with field instrumentation. Guidance is also provided by NRC (1976) for the release of decontaminated facilities and equipment from by-product, source, or special nuclear material manufacturing. Other criteria for operating and/or decommissioning nuclear facilities have been adopted by NRC (Federal Register 1981) and the U.S. Environmental Protection Agency (EPA) (40 CFR 190; 40 CFR 192).

The Richland Operations Office has adopted a public radiation protection standard of 25 mrem/yr, as estimated using the ARCL method, as the upper limit of dose that a member of the public could potentially receive as a result of decommissioning operations at Hanford.^(a) The basic DOE dose limits for protection of the public are being revised to limit exposures of

(a) Letter from Michael J. Lawrence, Manager of the Richland Operations Office of the U.S. Department of Energy, to the Hanford Site Contractors, July 3, 1984.

5 years or less duration to 500 mrem/yr, and exposures of longer duration to 100 mrem/yr (DOE 1988). [The method for estimating public radiation exposure has also been revised to consider the effective dose equivalent (EDE) as the official dose-limit unit. The EDE is defined in the International Commission on Radiological Protection (ICRP) Publication No. 26 (ICRP 1977) and is calculated using the dosimetry models described in ICRP Publication No. 30 (ICRP 1979a,b; 1980, 1981, 1982a,b)]. In setting this annual dose limit, the Richland Operations Office recognized the importance of the ALARA concept and the potential for individuals who could receive exposures from multiple sources on the Hanford Site. The 25 mrem/yr dose limit considers both the potential for multiple sources as well as the differences between unrestricted release and continued control of the Hanford Site.

1.2 HISTORY OF THE ARCL METHOD

The concepts behind the ARCL method have been under development at Pacific Northwest Laboratory (PNL) for over 12 years. Its first application was as part of a conceptual decommissioning study conducted for the NRC (Schneider and Jenkins 1977). The method was modified as the NRC conceptual decommissioning studies progressed to include a variety of nuclear fuel cycle facilities. These facilities ranged from fuel fabrication plants, to pressurized and boiling water reactors, to low-level waste disposal sites, and independent spent-fuel storage facilities. Example applications of the ARCL method from this early NRC effort to reactors that directly relate to this study are contained in reports by Smith, Konzek, and Kennedy (1978), Oak et al. (1980), and Konzek et al. (1981).

The ARCL method was modified in 1983 for application specific to the Hanford Site (Kennedy and Napier 1983a,b). These documents include an explanation of the procedures to be used and sample worksheets to assist in performing the calculations for a wide variety of facility conditions and mixtures of radionuclides. The purpose of that effort was to provide a tool for the evaluation of decommissioning alternatives for Hanford 100-Area facilities. The results were based on a scenario/exposure-pathway analysis and compliance with an annual dose limit assigned for each of three specific

modes of future use of the land and facilities. The three modes of future use that were considered were: restricted, controlled, and unrestricted.

Since the documents were published 1983, many assumptions concerning the application of the ARCL method have been standardized. The ARCL calculations may now be performed assuming a period of institutional control after the facilities are decommissioned to permit stabilization of the site and to allow for radioactive decay of the residual radionuclides. All of the facilities are assumed to be demolished after decontamination so that the "end-use" condition of the facilities and sites is one of contaminated rubble or soil under a layer of clean dirt or some other barrier. In the process of considering decommissioning alternatives using the ARCL method, several questions about applying the method to facility- and site-specific conditions were raised. This document is intended to help answer these questions and to provide enhancements of the original documentation.

1.3 OVERVIEW OF THE MODIFICATIONS TO THE METHOD

The original objective of the ARCL method was to help determine if radioactively contaminated sites require further decontamination or remedial action prior to release (Kennedy and Napier 1983a,b). In accomplishing this objective, the original ARCL method consisted of a screening method to select among decommissioning alternatives. The results of the analysis were also intended to be used to indicate the general magnitude of any remedial actions required prior to release.

Modifications have been made to the original method to account for the sampling statistics associated with the site characterization program and scenario enhancements to make the method more applicable to site-specific analyses. The result is the creation of a revised method that relies more on performance assessment methods than screening methods. The revised ARCL calculations depend on the mixture of radionuclides initially present, the time at which exposure is physically possible, the physical characteristics of each contaminated site, the radiation dose limit (i.e., 25 mrem/yr), the sampling statistics associated with the site characterization program, and on the scenarios of possible human exposure to radiation (including both probable levels of exposure and the upper bounds of potential exposure). The

basic approach to the modified ARCL method is presented in Section 2, and detailed procedures are presented in Section 5.

Finally, the radiation dose conversion factors used for calculating the doses from the inhalation and ingestion pathways have been updated. The dose factors are now presented in terms of "effective whole body dose," using the radiation dosimetry methods described in ICRP Publications 26 and 30 (ICRP 1977; 1979a,b, 1980, 1981, 1982a,b), along with certain parameters from ICRP Publication 48 (ICRP 1986). This updating simplifies the calculation of these dose factors and brings them into conformance with the rationale used by DOE in developing the 25 mrem/yr dose limit.

2.0 ALLOWABLE RESIDUAL CONTAMINATION LEVEL METHOD

The primary objective of the Allowable Residual Contamination Level (ARCL) method is to determine if radioactively contaminated sites require further decontamination or remedial action prior to release. A secondary objective is to permit a determination of what remedial actions could be effective. The ARCL method does not choose the most appropriate disposal alternative, nor does it automatically provide the best means of hazard mitigation. The analysis of remedial actions is simply an extended analysis of a site with modified physical characteristics. The general approach for performing ARCL calculations, including expanded consideration of sampling statistics in the site characterization program, is presented in this section.

2.1 SUMMARY OF THE METHOD

The general activities required for a complete ARCL application are illustrated in Figure 2.1. Each of these activities is further described in Sections 5.1 through 5.7 of this report. Each of the seven basic activities is comprised of from one to five tasks; each task, in turn, is composed of from one to eight procedures. A total of 25 separate procedures are detailed in a step-by-step manner. To aid in many of the procedures, worksheets are illustrated and explained in Section 5.

Preparation of a comprehensive site radionuclide inventory is the necessary first activity. This activity includes sampling and monitoring procedures that are outside the scope of this manual, but mention is made of where these tasks fit into the entire process.

The radionuclide inventory is adjusted in the second activity for application to each of the three controlling exposure scenarios. This adjustment can include corrections for radioactive decay if the proposed action is not immediate decontamination and release of the facility.

The calculation of individual radiation doses can proceed concurrently for the three major exposure scenarios, as illustrated in Figure 2.1. Calculation of the doses can include consideration of waste form availability, areal distribution, depth, decay, and mass of recyclable metals.

FIGURE 2.1. Required Activities During an ARCL Calculation

The dose estimates from the Residential/Home-Garden and Agricultural scenarios are then summed and the sum compared to the resource recycle result. The most restrictive dose is then compared to the mandated dose limit (i.e., 25 mrem/yr) to determine if the proposed decommissioning alternative is acceptable. In all calculations, the sampling variability and covariability of the measured radionuclide concentrations may be propagated, and the final answer may be phrased in terms of the Upper Confidence Limit.

Careful completion of each of the seven activities will provide a completely documented analysis of the final site disposal alternative, which should prove to be of great benefit to the Hanford decommissioning program.

2.2 SCENARIO DESCRIPTIONS

The use of the ARCL method for decommissioning the facilities on the Hanford Site is based on an evaluation of the potential radiation exposures resulting from specific "end-use" conditions of residual soil contamination. Radiation exposures are based upon representative scenarios and the measured mixtures of radionuclides present. These scenarios are designed to estimate the exposure conditions for individuals who could live on a decommissioned site after loss of institutional controls after decommissioning. The exposure scenarios include the radiation exposure pathways that are most likely for individuals who may actively contact residual radiative materials. The basic scenarios considered in the ARCL method for residual contamination associated with demolished facilities are: 1) the Resource Recycle Scenario, 2) the Residential/Home-Garden Scenario, and 3) the Agricultural Scenario. Full descriptions of these scenarios are given in the following sections.

2.2.1 Resource Recycle Scenario

This exposure scenario is designed to represent the potential activities of an individual engaged in recycling materials recovered during salvage operation in the buried rubble left after demolition of a facility. Because loss of institutional controls is postulated after decommissioning, it is assumed that an individual is free to salvage anything that may be of economic value at the site. The previous studies in this series (Kennedy and Napier 1983a,b) defined two scenarios for this situation: 1) the resource salvage scenario and 2) the resource recycle scenario. For standard assumed

conditions, they showed that the resource recycle scenario resulted in higher doses to an individual than did the resource salvage scenario. Therefore, for the purpose of this revised analysis, only the resource recycle scenario is considered further.

Data presented in the Draft Environmental Impact Statement Concerning Proposed Rulemaking Exemption From Licensing Requirements for Smelted Alloys Containing Residual Technetium-99 and Low-Enriched Uranium (U.S. NRC 1980), indicate that the operations with the greatest potential dose rate (i.e., annual dose) to an individual occur during smelting and manufacture of consumer products. During these operations, the worker in a smelter or foundry is exposed to piles of metal scrap, metal ingots, and accumulated finished products. He is additionally exposed to metal fumes and particulates. Radiation dose factors for these operations have been prepared by O'Donnell et al. (1978) for a study of dose to man from recycle of metals reclaimed from decommissioned nuclear power plants. From the information in this reference, it appears that the individual with the greatest potential for exposure is one working in a metal scrap yard.

The dose to a worker during material recycle is directly dependent on the quantity of material assumed to be recovered. The individual in the resource salvage scenario is assumed to work 2000 h/yr. The market price for scrap iron is assumed to be about \$0.09/kg, so for the individual to make a reasonable income, he would need to recover nearly 200 Mg/yr of scrap iron (about 1 Mg/day). This quantity of material is assumed to be melted and made into consumer products. A factory worker is assumed to work in a scrap yard, as described in O'Donnell et al. (1978), and to be exposed to the inhalation threshold limit value (TLV) of metal particulates (5 mg/m³) for a period long enough to process 200 Mg of recovered material.

2.2.2 Residential/Home-Garden Scenario

This scenario is designed to represent the unrestricted use of a decommissioned site by an individual who resides there and engages in home gardening activities for 50 years. When the individual moves onto the site, he is assumed to construct a basement with a volume of 1000 m³ (dimensions of 20 x 10 x 5 m). For purposes of determining if the inventory of radioactive materials will result in an annual dose rate less than 25 mrem/yr, the

basement is located in an area where the greatest (or worst) radionuclide inventory associated with the particular facility being evaluated would occur. All of the contamination present in this volume is assumed to be uniformly mixed with the clean soil in the volume and the resulting material is spread to a depth of 15 cm over about two-thirds of a hectare. This part of the scenario is similar to the intruder-construction scenario developed by the NRC in the Draft Environmental Impact Statement in support of 10 CFR Part 61 (U.S. NRC 1981). After the house is constructed, the individual is assumed to spend 12 h/d outdoors on the site, during which time he is exposed to direct penetrating radiation from the soil. The individual is also assumed to inhale resuspended contamination in the surface soil for 12 h/d during his 50 years of exposure, with an air concentration calculated using a mass loading factor of 10^{-4} g/m³ air (Napier et al. 1984).

Radionuclides contained in activated metals are not immediately available for resuspension or plant-root contact. For small pieces of metal with little intrinsic value (i.e., those that would not be considered in the resource recycle scenario), a reduction factor (RF) can be defined as the ratio of the mass of metal that has corroded at a certain time to the original mass. For plates or tubes of material, this can be written as

$$RF = 2t \text{ (corrosion rate, cm/yr)} / (\text{thickness, cm}) \quad (2.1)$$

where t is the time period of interest. The corrosion rate used is the uniform corrosion rate, which can be thought of as a total rate, expressed in cm/yr of surface. Many materials experience pitting corrosion, expressed in terms of weight loss per year. For the purposes of this analysis, pitting corrosion rates can be converted to equivalent uniform rates because only the total mass involved is of interest.

Uniform corrosion rates of 0.01 cm/yr and 0.001 cm/yr are used for carbon and stainless steels, respectively (Oztunali and Roles 1986), and 0.001 cm/yr for aluminum by analogy to stainless steel (MacKenzie et al. 1985).

Finally, the individual is assumed to grow 25% of his fruit and vegetable diet in a backyard garden that is located in the soil contaminated from the excavation.

2.2.3 Agricultural Scenario

The agricultural scenario is defined to account for the potential exposure to an individual from ingesting agricultural products whose roots penetrate into buried radioactive materials. The contaminated material could be either deeply buried (at a depth of 5 m or greater) or buried closer to the surface in the near vicinity of facilities under consideration through the residential/home-garden (basement) scenario. This scenario is similar to the root penetration scenario previously defined in the original ARCL method (Kennedy and Napier 1983a,b) for confined materials buried 5 m or greater from the surface. The only pathway of radiation exposure in this scenario is the ingestion of agricultural products containing radionuclides. The maximum exposed individual is assumed to obtain 25% of his fruit and vegetable diet from this food source. Because of the limited size of the contaminated areas, animal products have been excluded from this scenario. In this application, the depth of the buried materials is used to obtain a root penetration factor. This factor accounts for the fraction of the plant roots that penetrate below a given depth. This factor is then used as a multiplier to effectively reduce the root uptake of radionuclides from the soil and thus reduce the concentration of radionuclides in the plants. Figure 2.2 shows the crop root penetration factor as a function of depth as obtained from Napier (1982).

This scenario is based on the assumption that the site is of large enough size (maximum of 1 ha) to account for 25% of the fruit and vegetable diet. To account for small sites of radioactive contamination, an area correction factor for this scenario is next defined. To produce one quarter of the fruit and vegetable diet, it is estimated that the site would have to occupy an area of at least 0.1 ha. This value is based on the following estimates for the general area required to produce food products, assuming rather intensive gardening practices: 1) about 50 m² would be required to raise leafy and other above-ground vegetables, 2) about 200 m² would be required for root vegetables and grains, and 3) somewhat over 200 m² would be

LEGEND

1. 100% of plant roots are assumed to be at or below the soil surface.
2. 70% of alfalfa roots are in the top 1 meter (Washington State University 1970).
3. Soil line adapted from results for cumulative percent frequency of rooting depth for 1,012 vascular plants as reported by Foxx et al. (1985).
4. Extrapolation of Point 3 (dashed line).
5. Arbitrary 10-meter cutoff.

FIGURE 2.2. Crop Root Penetration in the Absence of Barriers

required for fruit trees (Kennedy et al. 1986). Of course, these estimates would vary depending upon the agricultural methods used and the climatic conditions encountered. It is suggested by these general estimates that 0.1 ha would be sufficient for producing 25% of the plant food diet for a family of

four. Thus, the fraction of the plant food diet that is assumed to be grown in contaminated soil is assumed to be proportional to the contaminated area for sites smaller than 0.1 ha, as shown in Figure 2.3.

2.3 DOSE FACTORS FOR RADIATION EXPOSURE PATHWAYS

The ARCL method relies on an analysis of maximum annual radiation doses resulting from the limiting radiation exposure scenario associated with the "end-use" of the residual contamination. Radiation dose conversion factors for each exposure pathway are given in Table 2.1. Dose factors for the resource/recycle scenario are derived from information given by O'Donnell et al. (1978).

FIGURE 2.3. Allowable Residual Contamination Level
Scenario Modification Factors

**TABLE 2.1. Exposure Pathway Dose Conversion Factors
in Units of Effective Dose Equivalent**

Radio- nuclide	Resource-Recycle (rem/yr per pCi/m ² Surface Area)	Contaminated Soil (rem/yr per pCi/g soil)	
		Inhalation of Resuspended Soil and External Exposure from Ground/Surface Shine	Crop Ingestion-- Internal Exposure Only
³ H	--	3.6E-11	4.7E-5
¹⁴ C	3.1E-15	6.1E-9	4.6E-5
³⁶ Cl	2.7E-14	7.0E-7	2.3E-3
⁵⁷ Co	2.7E-11	2.0E-4	1.1E-6
⁶⁰ Co	4.2E-10	6.4E-3	1.5E-5
⁵⁵ Fe	7.1E-15	8.8E-8	3.8E-8
⁵⁹ Fe	1.5E-10	2.9E-3	3.5E-7
⁵⁹ Ni	3.8E-14	1.1E-7	5.7E-7
⁶³ Ni	3.8E-14	1.0E-9	1.5E-6
⁹⁰ Sr+D	4.0E-12	1.6E-5	4.1E-3
⁹³ Mo	2.7E-11	3.0E-7	2.9E-5
⁹⁴ Nb	2.4E-10	3.9E-3	1.1E-5
⁹⁹ Tc	3.3E-15	4.5E-8	5.1E-4
¹²⁴ Sb	2.7E-10	5.2E-3	1.5E-5
¹²⁵ Sb+D	6.7E-11	1.2E-3	5.1E-6
¹³⁴ Cs	2.4E-10	4.4E-3	2.1E-5
¹³⁵ Cs	1.3E-13	1.3E-8	2.1E-6
¹³⁷ Cs+D	1.0E-10	1.5E-3	1.5E-5
¹⁴⁴ Ce+D	8.7E-11	1.4E-4	1.6E-6
¹⁵² Eu	1.8E-10	3.0E-3	2.6E-6
¹⁵⁴ Eu	1.9E-10	3.1E-3	3.7E-6
²²⁶ Ra	6.7E-11	5.2E-3	6.4E-2
²³² Th	3.9E-11	7.4E-3	2.4E-3
²³⁵ U+D	2.0E-11	2.6E-4	1.1E-5
²³⁸ U+D	1.2E-11	1.0E-4	1.8E-7
²³⁷ Np+D	1.5E-9	8.6E-4	2.1E-3
²³⁸ Pu	9.3E-10	1.2E-4	1.4E-5
²³⁹ Pu	1.0E-9	1.3E-4	1.6E-5
²⁴¹ Am	8.9E-10	2.0E-4	1.5E-4

The PNL-developed computer program GENII (Napier et al. 1988) was used to calculate dose conversion factors for continuous exposures. For inhaled particles, a particle size of 1 μm activity-median aerodynamic diameter (AMAD) is assumed. This particle size is within the respirable size distribution and is a "standard" assumption used in inhalation dose calculations when detailed information on the particle size distribution is not available. Dose factors for external exposures to contaminated soil are obtained from the ISOSHL D (Engel, Greenborg, and Hendrickson 1966; Simmons et al. 1967) computer program. Table 2.1 shows the combined inhalation and external exposure dose conversion factors for surface soil contamination obtained from the GENII computer program in units of rem/yr per pCi/g. These factors include consideration of 12-h/d exposure. A listing of the exposure pathway dose conversion factors for the ingestion pathway is also given in Table 2.1. These represent the dose factors for an infinite plane slab of contamination 15 cm deep, which leads to contamination of 25% of the individual's diet.

2.4 SCENARIO-SPECIFIC DOSE CONVERSION FACTORS

By applying the exposure conditions defined for the radiation exposure scenarios, scenario-specific dose conversion factors for the combined pathways of each scenario are next calculated. These scenario-specific dose conversion factors are calculated using the pathway dose factors shown in Table 2.1 and are listed in Table 2.2. For the recycle pathway, the external dose conversion factors are in units of rem/yr per pCi/m² of contaminated surface, adjusted to a recycle rate of 200 Mg/yr, as described in Section 2.2.1. These factors are taken from O'Donnell et al. (1978) and are shown by radionuclide in Table 2.1. The dose factors for the recycle scenario are in units of rem/yr per pCi in 200 Mg of contaminated metal, the residential/ home-garden scenario dose factors are in units of rem/yr per pCi in the excavated soil spread over about 0.67 ha, and the agricultural scenario dose factors are in units of rem/yr per pCi in soil distributed over 1 ha. Scenario dose conversion factors are supplied for a listing of 28 radionuclides of potential importance to decommissioning operations on the Hanford Site.

TABLE 2.2. ARCL Scenario-Specific Dose Conversion Factors in Effective Dose Equivalent rem/yr per pCi/Stratum

Radio-nuclide	Scenario (rem/yr per pCi/Stratum)			
	Resource-Recycle	Residential Home-Garden		Agricultural Ingestion(b)
	External, Inhalation & Ingestion	External & Inhalation	Ingestion(a)	
³ H	--	2.4E-20	3.1E-14	2.1E-14
¹⁴ C	1.6E-19	4.1E-18	3.1E-14	2.1E-14
³⁶ Cl	1.4E-17	4.7E-16	1.5E-12	1.0E-12
⁵⁷ Co	1.4E-14	1.3E-13	7.3E-16	4.9E-16
⁶⁰ Co	2.1E-13	4.3E-12	1.0E-14	6.7E-15
⁵⁵ Fe	3.6E-18	5.9E-17	2.5E-17	1.7E-17
⁵⁹ Fe	7.5E-14	1.9E-12	2.3E-16	1.5E-16
⁵⁹ Ni	1.9E-17	7.3E-17	3.8E-16	2.5E-16
⁶³ Ni	1.9E-17	6.7E-19	1.0E-15	6.7E-16
⁹⁰ Sr+D	2.0E-15	1.1E-14	2.7E-12	1.8E-12
⁹³ Mo	1.4E-14	2.0E-16	1.9E-14	1.3E-14
⁹⁴ Nb	1.2E-13	2.6E-12	7.3E-15	4.9E-15
⁹⁹ Tc	1.7E-18	3.0E-17	3.4E-13	2.3E-13
¹²⁴ Sb	1.4E-13	3.5E-12	1.0E-14	6.7E-15
¹²⁵ Sb+D	3.4E-14	8.0E-13	3.4E-15	2.3E-15
¹³⁴ Cs	1.2E-13	2.9E-12	1.4E-14	9.3E-15
¹³⁵ Cs	6.5E-17	8.7E-18	1.4E-15	9.3E-16
¹³⁷ Cs+D	5.0E-14	1.0E-12	1.0E-14	6.7E-15
¹⁴⁴ Ce+D	4.4E-15	9.3E-14	1.1E-15	7.1E-16
¹⁵² Eu	9.0E-14	2.0E-12	1.7E-15	1.2E-15
¹⁵⁴ Eu	9.5E-14	2.1E-12	2.5E-15	1.6E-15
²²⁶ Ra	3.4E-14	3.5E-12	4.3E-11	2.8E-11
²³² Th	2.0E-14	4.9E-12	1.6E-12	1.1E-12
²³⁵ U+D	1.0E-14	1.7E-13	7.3E-15	4.9E-15
²³⁸ U+D	6.0E-15	6.7E-14	1.2E-16	8.0E-17
²³⁷ Np+D	7.5E-13	5.7E-13	1.4E-12	9.3E-13
²³⁸ Pu	4.7E-13	8.0E-14	9.3E-15	6.2E-15
²³⁹ Pu	5.0E-13	8.7E-14	1.1E-14	7.1E-15
²⁴¹ Am	4.5E-13	1.3E-13	1.0E-13	6.7E-14

(a) Based on the activity of the respective radionuclides in the top 15 cm of soil within an area of 0.67 hectare; no correction for depth should be applied.

(b) Based on the activity of the respective radionuclides in the top 15 cm of soil over an area of 1.0 hectare; can be adjusted for depth and area of contamination.

3.0 EXPOSURE SCENARIOS CONSIDERED BUT DISMISSED

The Allowable Residual Contamination Level (ARCL) method for decommissioning nuclear facilities depends on the comparison of an annual radiation dose limit to a calculated annual dose commitment. The calculated dose used in the comparison is the maximum from a suite of scenarios designed to provide estimates of dose to a maximally exposed individual. In the development of the original ARCL technique (Kennedy and Napier 1983a,b), the scenarios chosen for detailed analysis were those involving human intrusion into and/or interaction with the residual radioactive material. Certain other scenarios were discarded after initial analyses indicated that they were not limiting (i.e., that they did not result in the highest dose rate). This section provides additional detail on the scenarios that were discarded and compares dose factors for them to dose factors for the scenarios that were retained.

3.1 ADDITIONAL INTRUDER SCENARIO PATHWAY DESCRIPTIONS

The potential routes through which people may be exposed to radionuclides or radiation are called "exposure pathways." The general pathways can be thought of as external exposure, inhalation, and ingestion. Doses from external exposure result from direct radiation from air, water, soil, and contaminated structures. Doses from inhalation can result from breathing aerosols released from facilities or from materials resuspended from soils. Ingestion doses are from consumption of water, fish, waterfowl, game, food crops, animal products, or direct consumption of small amounts of material transferred from contaminated surfaces to the hands. The ARCL method used for individual sites is based on the sum of exposures through all the selected pathways in a radiation exposure scenario analysis.

The key to the ARCL method is an analysis of the maximum annual radiation dose an individual could receive, which is calculated by summing the doses from many exposure pathways. The pathways are chosen depending on the potential ways an individual could be exposed to residual radionuclides for each condition that could exist after decommissioning. The collection of

appropriate pathways is called an "exposure scenario." The ability of the user of the method to choose the exposure scenario is what gives the ARCL method the flexibility to handle many types of sites, inventories, and locations.

For release of facilities for unrestricted use, the following general types of potential exposure scenarios were investigated:

- transient
- permanent resident
- offsite populations
- well drilling, excavation
- contact with soil, inhalation of resuspended material
- drinking of well water
- transport in the environment by plants and animals (biotic transport)
- backyard garden
- inadvertent intruder
- intentional intruder
- resource recovery
- recovered resource use.

As described in Section 2, the detailed composite scenarios of resource recycle, residential home gardening, and agriculture in contaminated soil are used as the limiting scenarios for unrestricted release of soil and facilities. Those scenarios are compared here to biotic transport and groundwater transport scenarios for several conditions. This comparison is provided to illustrate the degree to which the intruder scenarios are the most important. Also given are potential doses to individuals and populations out of the immediate vicinity of the wastes. These doses are included to illustrate the strength of the conclusion that the dose to the onsite individual is indeed the limiting dose. No attempt is made to consider the additive effects of more than one scenario occurring at a time. There are three reasons for

this: 1) several of the scenarios are mutually exclusive (i.e., biotic transport by native plants versus agriculture); 2) the scenarios consider effects that occur at different times (i.e., intrusion versus long-term transport); and 3) the scenarios are designed to affect different individuals involved in different activities (i.e., residents, farmers, offsite members of the public). Furthermore, since 100% of the radionuclide inventories are assumed to be involved with each scenario, addition would result in "double counting" of the impacts.

3.1.1 Biotic Transport Scenario

Transport of buried radioactive material to the soil surface by indigenous plants and animals is a very slow process, but continued over long periods, it may result in substantial exposure to humans from unprotected waste sites. At Hanford, biotic transport has resulted in "nuisance" contamination from past practices. The overall processes of waste-form degradation, followed by plant or animal uptake, are relatively poorly understood, but are continuing to be researched. A preliminary model of biotic transport processes has recently been developed (McKenzie et al. 1982a,b; Kennedy, Cadwell, and McKenzie 1985). This model indicates that, for low-level radioactive waste disposal sites without barriers to prevent intrusion by plants and animals, the quantity of radionuclides transported to the soil surface can be significant compared to other pathways. A simplified version of the McKenzie model was implemented in the code GENII (Napier et al. 1988) to develop appropriate dose conversion factors.

Following release of site control, a plant and animal community representative of arid, western sites is assumed to establish itself over the wastes. The description of the biotic community is taken from McKenzie et al. (1982b). To illustrate the variability of biotic transport with depth of clean cover, both 1-m- and 5-m-thick overburdens are modeled.

The human exposure assumptions used in the calculations are the same as those used in the residential/home garden scenario. After a period of biotic transport, an individual is assumed to move on to the land and grow a garden in the soil. The individual residing on the site contaminated under the biotic transport scenario is exposed through three pathways: inhalation of resuspended dusts, ingestion of crops grown in the soil, and direct

exposure. Initially, the radionuclides preferentially moved by biotic transport are those readily taken up by plants. At longer times, the doses tend to come from the less mobile, but longer-lived nuclides, such as ^{239}Pu , through resuspension. Therefore, the values used in the comparisons are the maxima noted in the initial 100-yr period.

Only localized concentrations of radionuclides are expected near the waste sites; once the material is brought to the surface, normal erosion processes would tend to disperse it. The effects of this scenario on populations also depend directly on the number of people assumed to be living on the site.

3.1.2 Ground-Water Migration Scenario

Leaching of radionuclides from soil or buried waste and subsequent transport by ground water is a difficult process to quantify. On the Hanford Site, the distance between the bottom of a site containing radioactive material and the top of the ground water varies from a few meters to over 70 m. Sites in the Hanford 200 Areas are typically around 60 m above the ground-water, but sites in the 100 Areas can be as close as one or two meters. Calculation of rates of dissolution of radioactive material and transport through the vadose zone is complicated by the longer distances in the 200 Areas; therefore, representative calculations in the 100 Areas are given in the following sections. This results in a degree of conservatism from neglecting the additional dispersion and holdup that could be provided by the vadose zone.

3.1.2.1 Migration Under Existing Conditions

The average annual flow rate of the unconfined aquifer into the section of the Columbia River between river miles 360 and 356, several miles south of 100-F Area, has been estimated to be about $3 \text{ ft}^3/\text{s}$ ($0.085 \text{ m}^3/\text{s}$) (Prater et al. 1983) as determined by the Variable Thickness Transient (VTT) ground-water flow model developed for the Hanford Site (Reisenauer 1979). Over one-half of this flow occurs between river miles 359 and 360. If an average value derived from this observation of about $400 \text{ m}^3/\text{yr}$ per meter of shoreline is used as a representative value for current conditions, an approximation of the flow under a representative 100-Area 1-ha site may be taken to be about

$4 \times 10^4 \text{ m}^3/\text{yr}$. Travel times of this water from sites near the river into the river depend on the specific location of the waste site and the actual direction of the average ground-water flow. It is, however, a relatively quick transit, and a representative value of about 30 days is reasonably derived from the literature.

Radionuclides in the soil will not move to the ground water in appreciable quantities without a driving force provided by percolating water. Estimates of ground-water recharge (i.e., the amount of water trickling through the upper soil to the water table) for the Pasco Basin and the Hanford Site under present conditions vary with location and soil characteristics. The recharge is small but has been measured in the Hanford 300 Area to be around 5 cm of water per year in a wet year (Kirkham and Gee 1983). This could be sufficient to leach a small amount of material to the ground water every year. A leach rate of 0.01 yr^{-1} is assumed here for most materials, and 0.001 yr^{-1} , for activated metals (Staley, Turi and Schreiber 1979). This assumed leach rate is consistent with the Availability Correction Factor discussed earlier.

For an inventory I (pCi) in a site, the annual average water concentration can be estimated from the following equation (Codell 1983).

$$\bar{W}(t) = \frac{I\lambda_L e^{-\lambda_R t}}{F(\lambda_L + 2\lambda_R)T} \quad 1 - e^{-(\lambda_L + 2\lambda_R)t} \quad (3.1)$$

where $\bar{W}(t)$ = the average water concentration, pCi/L, at time t ,

I = the initial inventory, pCi,

λ_L = the leach rate constant, yr^{-1} ,

λ_R = the radiological decay constant, yr^{-1} ,

F = the dilution flow, L/yr, and

T = an averaging time of 1 yr.

Equation 3.1 does not include any correction for retardation; that is, materials are assumed to move with water.

For an idealized waste site of area 1-ha, depth 1-m, and normalized concentration of 1 pCi/g, this equation implies a maximum ground-water

concentration of 5 pCi/L for loosely distributed material and of 0.5 pCi/L for activated metals. This concentration is assumed to result immediately downgradient of the waste site.

3.1.2.2 Migration Enhanced by Irrigation

Before the establishment of the Hanford Site, the area within about 3 km of the river from about the 100-K Area to several miles south of 100-F Area was under irrigation. It is reasonable to assume that, upon unrestricted release of the 100 Areas, a similar situation could recur. Because the supply of ground water is insufficient to support major irrigation through wells, it is assumed that the irrigation water is withdrawn from the Columbia River.

Irrigation methods have improved with time to increase the efficiency of water application so that losses to the ground water are minimized. However, some excess irrigation is required to avoid accumulation of salts in the soil. Therefore, a nominal loss of 20% of applied water is assumed. For irrigation practices in the Columbia Basin, this could be up to 30 cm/yr of recharge. This quantity of recharge throughout the 3-km band along the river would act to raise the ground water, increasing the flow to the river and decreasing the travel time. This irrigation could increase the flow beneath a hypothetical waste site to about $1 \times 10^5 \text{ m}^3/\text{yr}$.

The increased flux of water through the buried waste would increase the leaching of radionuclides to the water table. A leach rate of 0.1 yr^{-1} is assumed for most materials under these circumstances, and a leach rate of 0.01 yr^{-1} is assumed for activated components (Staley, Turi and Schreiber 1979).

Equation 3.1, applied to the hypothetical 1-ha waste site with 1 pCi/g, results in a maximum projected ground-water concentration of 20 pCi/L for most radionuclides. For activated materials, the projected ground-water concentration is 2 pCi/L. Retardation is neglected in these estimates. However, the travel time to the river is so short that retardation should be inconsequential.

For this scenario, an individual is assumed to draw drinking water from the unconfined aquifer through a well. He is assumed to drink 2 L/d from

this source. Because of the source of the irrigation water in this scenario, the enhanced migration of contaminants could impact people using the water for the Columbia River down stream (considered in Section 3.2).

3.1.2.3 Ground-Water Migration Retarded by a Barrier

Because of the potential for downward migration of buried materials to the water table, it is possible that barriers to water infiltration will be proposed and constructed. While these could conceivably eliminate recharge as a pathway, molecular diffusion would still remain as a minor source of contaminant movement to ground water.

Neglecting lateral diffusion or edge effects around the barrier (i.e., assuming one-dimensional transport downward), contaminant transport via diffusion can be described by the equation

$$J = -\theta DA \frac{\partial C}{\partial X} \quad (3.2)$$

where J = activity flux, Ci/yr,
 θ = moisture content of the soil (dimensionless),
 D = modified molecular diffusion constant, m²/yr,
 A = site area, m²,
 C = activity concentration, Ci/m³, and
 X = longitudinal coordinate, m.

This equation has boundary conditions of $C = C_0$ at the origin, assuming that the concentration at the waste/soil boundary is in equilibrium, and $C = 0$ at the water table, assuming that the flow in the ground water dilutes and removes the contamination.

One can visualize the release to the water table via the diffusion mechanism as occurring in three time phases: 1) zero release as the waste moves to the water table, 2) a period of constant release, and 3) a period of exponentially decreasing release as the source is depleted. For the maximum (second) period, the flux can be approximated as

$$J = -\theta D A C_0 / L \quad (3.3)$$

where L = a distance over which the diffusion occurs.

The distance from the buried material to the water table varies from site to site. For certain sites in the 100-F Area, the distance can be as little as 1 m (Dorian and Richards 1978). The diffusion constant in a sandy loam material, similar to Hanford soils, has been shown to be about $0.15 \text{ cm}^2/\text{day}$ ($5.4 \times 10^3 \text{ m}^2/\text{yr}$). The moisture content of Hanford soils runs about 7%. Application of these parameters to the hypothetical waste site used in the previous examples results in a nominal water concentration of 0.2 pCi/L per pCi/g.

For compatibility with the other scenarios, the impact of this concentration of each nuclide in the ground water is measured using a drinking-water scenario in which the individual is assumed to consume 2 L/d from a well immediately downgradient from the buried radioactive material.

3.2 OFFSITE SCENARIO/PATHWAYS

The release of radionuclides to the Columbia River via ground water, as described for several different circumstances in Section 3.1.2, has the potential for impacts on both individuals and population groups downstream. Doses to individuals are calculated for

- ingestion of foodstuffs contaminated by effluents deposited on the ground by irrigation with Columbia River water
- drinking sanitary water obtained from the Columbia River
- exposure to ground contaminated by irrigation with Columbia River water
- ingestion of fish taken from the Columbia River
- recreation along the Columbia - boating, swimming and shoreline activities.

While there are no regulatory limits for collective population doses, such an evaluation provides an indication of the overall impact. The dose to the population within 80-km of the site represents the summed products of average dose and number of individuals involved for all possible pathways. The units are man-rem.

In the case of releases to the Columbia River, only that portion of the full 80 km population using river water are potentially exposed. The river-related exposure pathways are drinking water, irrigated food stuff, fish consumption, and river recreation. Descriptions of river-related pathways are as follows (McCormack, Ramsdell, and Napier 1984):

- Drinking Water - The cities of Richland and Pasco obtain their municipal water from the Columbia River downstream from Hanford. The City of Kennewick began drawing a portion of its municipal water from the river in late 1980. During 1984, approximately 40% of Kennewick drinking water was drawn from the Columbia River. The total affected population was approximately 70,000.
- Irrigated Foodstuff - Columbia River water is withdrawn for irrigation of home vegetable gardens in the Riverview district of Pasco in Franklin County. Approximately 2000 people are estimated to be affected.
- River Recreation - These activities include swimming, boating, and shoreline recreation. The population residing adjacent to the river within 80 km of Hanford is assumed to be affected by these pathways and is estimated to number 125,000.
- Fish Consumption - Population doses due to consumption of fish obtained locally from the Columbia River are calculated based on an estimated total annual catch of 15,000 kg/yr without reference to a specific population group.

3.3 ADDITIONAL SCENARIO DOSE CONVERSION FACTORS

By applying the exposure conditions defined in Section 3.1 for the radiation exposure scenarios, and using the dose models discussed in Section 2, dose conversion factors for determining ARCL values are calculated. The ARCL dose conversion factors are shown in Table 3.1 for specific radiation exposure pathways for radionuclides of potential interest during decommissioning. The dose factors are in units of effective dose equivalent (EDE) rem/yr per pCi/g, except for the two offsite dose factors; individual doses

are in rem per Ci released to the river, and population doses are in man-rem per Ci released.

The new factors shown on Table 3.1 are similar in nature to the exposure pathway dose conversion factors in Table 2.1. The residential/home garden factors shown in Table 3.1 are derived directly from Table 2.1, by summing the external, inhalation, and ingestion factors.

3.4 SENSITIVITY STUDIES AND RESULTS

The dose factors in Table 3.1 and Table 2.1 provide the basic information necessary for comparison of the relative magnitude of the dose that results from the selected scenarios. Because these comparisons vary by radionuclide, and because in some instances the units of the dose factors are not directly compatible, it is easier to make the comparison using a representative 100-Area site as an example. Because the dose factors were derived for a hypothetical 1-ha site with uniform contamination, it is most appropriate to use a site with similar characteristics.

3.4.1 Example Site Description

A total of 28 solid-waste disposal sites have been identified in the 100 Areas of the Hanford Site (Harmon and King 1975). All of these sites have been retired since the early or mid-1970s, and all wastes generated since that time have been sent to the 200-Area waste sites for disposal. Solid waste burial grounds in the 100 Areas simply consist of unlined trenches that are filled with solid, radioactive wastes and covered with a meter or more of soil.

A hypothetical radionuclide inventory for the solid-waste burial grounds in the 100 Areas is given in Table 3.2. The radionuclides and relative concentrations shown in this table were normalized to 1 pCi/g.

3.4.2 Scenario Result Comparison

Results of application of the dose factors for the onsite (intruder) scenarios are presented in Table 3.3 for a hypothetical 1-ha burial ground containing 1 pCi/g of the waste spectrum of Table 3.2. Results are presented for the residential/home-garden scenario and agriculture scenario presented in Section 2. Contrasted to these scenarios are two scenarios of biotic

TABLE 3.1. Additional Scenario Dose Conversion Factors in Units of Effective Dose Equivalent

Radio-nuclide	Onsite Scenarios (rem/yr/per pCi/g)						Downriver Scenarios	
	Residential Home Garden	Biotic Transport at 1 m	Biotic Transport at 5 m	Existing Conditions	Ground Water Irrigation	Ground Water Barrier Installed	Individual (rem/Ci)	Population (man-rem/Ci)
3H	4.7E-6	---(a)	---	2.1E-7	8.8E-7	8.8E-9	8.8E-10	1.7E-5
14C	4.8E-6	---	---	7.3E-8	2.9E-6	2.9E-7	3.7E-8	2.1E-3
36Cl	2.3E-3	1.4E-3	2.4E-5	1.1E-5	4.2E-6	4.2E-7	6.4E-7	1.6E-3
57Co	2.8E-4	2.7E-22	3.1E-24	5.5E-8	2.2E-7	2.2E-8	3.2E-8	6.4E-5
60Co	6.4E-3	8.7E-7	1.1E-8	7.3E-7	2.9E-8	2.9E-7	7.3E-7	1.1E-3
56Fe	1.3E-7	9.8E-13	1.4E-14	4.3E-8	1.7E-7	1.7E-8	3.5E-8	5.9E-5
59Fe	2.9E-3	0.0E0	0.0E0	4.8E-7	1.9E-8	1.9E-7	3.8E-7	5.6E-4
59Ni	8.8E-7	4.2E-7	6.1E-9	1.4E-8	5.4E-8	5.4E-9	1.1E-8	1.7E-5
63Ni	1.5E-8	8.8E-7	1.2E-8	3.8E-8	1.5E-7	1.5E-8	2.9E-8	4.7E-5
90Sr	4.1E-3	1.1E-3	1.7E-6	8.8E-5	3.5E-4	3.5E-6	3.8E-6	9.9E-3
93Mo	2.9E-5	2.8E-5	3.0E-7	4.5E-7	1.8E-8	1.8E-8	5.1E-8	4.8E-4
94Nb	3.9E-3	1.2E-4	1.3E-6	1.8E-5	7.2E-5	7.2E-7	8.8E-5	3.2E-2
99Tc	5.1E-4	3.8E-4	5.2E-6	7.8E-8	3.1E-5	3.1E-7	2.1E-7	8.6E-4
124Sb	5.2E-3	0.0E0	0.0E0	3.0E-5	1.2E-4	1.2E-6	2.4E-7	2.4E-3
125Sb	1.2E-3	5.8E-9	8.1E-11	8.3E-8	3.3E-5	3.3E-7	1.7E-7	7.8E-4
134Cs	4.4E-3	1.8E-10	1.8E-12	2.2E-4	8.8E-4	8.8E-6	5.4E-5	4.8E-2
135Cs	2.1E-6	1.5E-6	2.2E-8	2.1E-5	8.4E-5	8.4E-7	5.1E-6	3.8E-3
137Cs	1.5E-3	2.8E-5	2.3E-7	1.5E-4	6.8E-4	6.8E-6	3.7E-5	2.7E-2
144Ce	1.4E-4	1.1E-21	1.8E-23	1.5E-4	6.1E-5	6.1E-7	2.4E-7	1.4E-3
152Eu	3.8E-3	9.9E-6	1.2E-7	4.5E-6	1.8E-5	1.8E-7	3.5E-7	6.4E-4
154Eu	3.1E-3	3.3E-6	3.8E-8	6.8E-6	2.7E-5	2.7E-7	4.2E-7	8.8E-4
226Ra	6.9E-2	4.8E-2	6.7E-4	2.4E-3	9.5E-3	9.5E-5	3.3E-5	2.1E-1
232Th	9.8E-3	1.8E-3	2.2E-5	7.8E-3	2.8E-2	2.8E-4	8.8E-5	6.8E-1
235U	2.7E-4	1.3E-5	1.7E-7	8.5E-5	2.8E-4	2.8E-6	2.8E-7	5.7E-3
238U	1.8E-4	1.5E-5	2.1E-7	6.8E-5	2.4E-4	2.4E-6	5.7E-7	5.3E-3
237Np	3.8E-3	1.5E-3	2.2E-5	1.3E-2	5.2E-2	5.2E-4	1.1E-4	1.1E-8
238Pu	1.3E-4	9.4E-6	1.3E-7	8.5E-4	3.4E-3	3.4E-5	6.4E-6	7.2E-2
239Pu	1.5E-4	1.4E-5	1.9E-7	9.3E-4	3.7E-3	3.7E-5	7.8E-6	7.9E-2
241Am	3.5E-4	1.4E-5	1.5E-6	9.8E-3	3.8E-2	3.8E-4	9.7E-5	7.8E-1

transport and three scenarios of ground-water transport. The biotic transport scenarios correspond to the standard intrusion scenarios of minimal overburden (<1 m) and greater than 5 m of overburden. For unconfined soil, the biotic transport model gives results identical to those for the agricultural scenario with resuspension and external, so they are not repeated. Barriers are assumed to be able to preclude biotic transport. The depth of

TABLE 3.2. Representative Radionuclide Inventory for the Example Site

Radionuclide	Relative Activity at T = 0 yr (pCi/g)	Relative Activity Decayed to T = 100 yr (pCi/g)	Relative Activity Decayed to T = 300 yr (pCi/g)
^{60}Co	9.1E-1 ^(a)	1.8E-6	6.7E-18
^{63}Ni	8.7E-3	4.4E-3	1.0E-3
$^{90}\text{Sr}+\text{D}$ ^(b)	1.6E-2	1.3E-3	8.3E-6
$^{137}\text{Cs}+\text{D}$	1.3E-2	1.3E-3	1.3E-5
^{152}Eu	9.2E-2	5.5E-4	6.7E-9
^{154}Eu	4.2E-2	1.4E-5	1.4E-12
$^{238}\text{U}+\text{D}$	1.8E-5	1.8E-5	1.8E-5
^{239}Pu	1.1E-4	1.1E-4	1.1E-4
TOTALS	1.0	0.0077	0.0012

(a) Where $9.1\text{E}-1 = 9.1 \times 10^{-1}$.

(b) +D means plus short-lived daughter products.

TABLE 3.3. Relative Dose Rates for the Various Onsite Scenarios for the Example Site Inventory, Effective Dose Equivalent, at T = 0 yr

Radio-nuclide	rem/yr per pCi/g						
	Residential/ Home Garden	Agriculture	Biotic Transport		Ground-Water Migration		
			1-4 m Soil	>5 m Soil	Existing	Irrigation	Barrier
^{60}Co	5.8E-3	1.4E-5	7.9E-7	1.0E-8	8.8E-7	2.6E-8	2.6E-7
^{63}Ni	1.3E-8	1.3E-8	7.5E-9	1.0E-10	3.3E-10	1.3E-9	1.3E-10
^{90}Sr	6.0E-5	6.0E-5	1.8E-5	2.7E-8	1.4E-8	5.6E-8	5.6E-8
^{137}Cs	2.0E-5	2.0E-7	2.6E-7	3.0E-9	2.0E-8	7.8E-8	7.8E-8
^{152}Eu	2.6E-4	2.4E-7	9.1E-7	1.1E-8	4.1E-7	1.7E-8	1.7E-8
^{154}Eu	1.3E-4	1.6E-7	1.4E-7	1.6E-9	2.9E-7	1.1E-8	1.1E-8
^{238}U	1.8E-9	3.2E-12	2.7E-10	3.8E-12	1.1E-9	4.3E-9	4.3E-10
^{239}Pu	1.7E-8	1.8E-9	1.5E-9	2.1E-11	1.0E-7	4.1E-7	4.1E-9
TOTAL	6.2E-3	8.0E-5	2.0E-5	5.3E-8	4.9E-8	1.9E-5	

cover is of less significance for the ground-water transport scenarios, but the nature of the cover and the activities on the cover are of concern. Thus, the calculations are provided for wastes under presently existing conditions, under assumed irrigation, and under a barrier designed to prevent water infiltration.

Table 3.4 presents results of similar calculations of potential doses to individuals and populations downstream along the Columbia River resulting from the 100% release of the hypothetical waste site inventory to the river via ground-water transport. Doses to the individual are presented in the same units Table 3.3, those to the population are in terms of total man-rem to total body.

Inspection of Tables 3.3 and 3.4 reveals that the standard scenarios employed in Kennedy and Napier (1983b), for intruders into the wastes, yield the highest projected individual annual doses. The doses to individuals offsite, shown in Table 3.4, are lower than from any of the onsite doses. Of the onsite doses, only the scenarios of biotic transport with minimal cover (very close to the agricultural scenario) and of heavy irrigation with attendant nuclide migration into drinking water approach the standard scenarios in magnitude. Thus, it is apparent that the initial selection of human-intrusion-based scenarios for ARCL determination is appropriately conservative. It also indicates that compounding of scenarios, e.g., adding together irrigation with the intruder, would result in negligible dose increases over the intruder alone.

3.4.3 Discussion of Results

The scenarios initially chosen for use in the ARCL calculations are shown to be reasonable and conservative. Offsite doses are very low in comparison to potential onsite exposures and can be considered to be negligible. (The offsite population dose in total man-rem is less than the potential dose to a single onsite individual, in rem.) Of the onsite scenarios, only biotic transport with minimal cover and the drinking-water scenario driven by high irrigation recharge approach the magnitude of the intrusion-based scenarios. Biotic transport does not appear to be a significant pathway for migration beyond that component already assumed in the agricultural scenarios.

TABLE 3.4. Relative Effective Dose Equivalent Rates for the Various Offsite Scenarios for the Example Site Inventory at T = 0 yr

<u>Radionuclide</u>	<u>Maximum Downriver Individual (rem per pCi/g)</u>	<u>Total Downriver Population (man-rem per pCi/g)</u>
⁶⁰ Co	1.0E-8	1.5E-5
⁶³ Ni	3.8E-12	6.1E-9
⁹⁰ Sr	9.1E-10	2.4E-6
¹³⁷ Cs	7.2E-9	5.3E-6
¹⁵² Eu	4.8E-10	8.8E-7
¹⁵⁴ Eu	2.6E-10	5.4E-7
²³⁸ U	1.5E-13	1.4E-9
²³⁹ Pu	1.2E-11	1.3E-7
TOTALS	1.9E-8	2.4E-5

The biotic transport and ground-water migration scenarios are, to a large degree, dependent more on the total inventory of radionuclides buried than on the concentration of these materials in the soils. This tends to mask the potential impacts of nonuniformity in the disposed material. The intrusion scenarios, on the other hand, reflect the impact of such nonuniformities, because the intruders are commonly assumed to interact with the areas of higher concentration. This conservatism helps assure that the potential for doses to future individuals is not minimized.

4.0 A STATISTICAL APPROACH FOR MAKING REMEDIAL ACTION DECISIONS BASED ON THE ARCL METHODOLOGY

This section discusses the statistical application of the ARCL method for deciding whether or not a contaminated nuclear facility or site requires remedial action before it can be released for unrestricted public use. The formulas and discussion are meant to provide an overview of the statistical remedial action decision-making approach using ARCL methodology. Step-by-step descriptions and worksheets are provided in Section 5.0.

Remedial action (RA) decisions are made by comparing radionuclide measurements (or functions thereof) to regulatory or other guidelines. Usually it is not possible to characterize 100% of a site or facility, so samples are taken and radionuclide measurements are made on the samples. The RA decision is then based on this sample information. Unfortunately, a nonstatistical decision rule (one that ignores the variability among sample radionuclide measurements) is often used. If the variability among sample measurements is large, the result can be high probabilities of making wrong RA decisions (i.e., deciding RA is required when it is not, or deciding RA is not required when it is).

When a statistical approach to making RA decisions is used, the variability among measurements is taken into account, and the probabilities of making incorrect decisions can be assessed. Further, these probabilities can be controlled at user-specified values by taking a sufficient number of samples.

4.1 STATISTICAL REMEDIAL ACTION DECISIONS

The remedial action decision-making process for decommissioning operations at Hanford is based on the requirement that the annual radiation dose to a maximally exposed individual (subsequently denoted by D) must be below 25 mrem/yr. The ARCL methodology provides the means for estimating D , given information about radionuclide concentrations and inventories. Specifically, radionuclide inventories or mean concentrations are multiplied by radionuclide dose factors to yield radionuclide doses, which are then summed over radionuclides, strata, and scenario/exposure-pathways to yield an estimate $\hat{D}$ of D .

In practice, radionuclide concentration (or inventory) information is obtained by collecting physical samples from contaminated facilities and making radionuclide measurements on the samples. Because levels of contamination in the samples are often quite variable, the estimate of D can be quite variable.^(a) This variability must be taken into account in specifying the RA decision rule.

The statistical approach to making an ARCL remedial action decision uses a hypothesis-testing framework and involves computing a one-sided $100(1-\alpha)\%$ confidence limit on D . If the confidence limit exceeds the dose limit (25 mrem/yr), then RA is required. It is possible to use either an upper or lower confidence limit in this procedure. The choice between the lower and upper confidence limits depends on which of the two potential decision errors is more important to control. If the consequences of deciding "RA is not required when it is actually needed" are more severe than the consequences of deciding "RA is required when it is not actually needed," then the upper-confidence-limit decision rule should be used. If the opposite is the case, the lower confidence limit should be used. Through discussions with decommissioning personnel, it was concluded that the more serious decision error is deciding that "RA is not required when it is actually needed," which leads to using the upper confidence limit.

The following sections describe how to compute the upper confidence limit on D .

4.2 THE 100(1- α)% UPPER CONFIDENCE LIMIT DECISION RULE

This section shows how to compute an approximate $100(1-\alpha)\%$ upper confidence limit (UCL) on the true unknown annual dose D to a maximally exposed individual where the desired confidence, $1-\alpha$, is specified by the user. This computed UCL is compared to the 25 mrem/yr limit to decide whether RA is needed.^(a) Note that only the sampling variability in estimating inventories is factored into the UCL. The dose factors are treated as if they were known

(a) It is possible to reduce this variability through the use of stratification in sampling contaminated facilities. Stratification is discussed in more detail in Sections 4.2, 4.3, and 5.1.

without error (this is obviously not the case, but it is beyond the scope of work to consider uncertainties in radionuclide dose factors).

4.2.1 Main Formulas for UCL

The method used to estimate D (discussed in Sections 4.3 and 5) suggests that the estimate $\hat{D}$ ought to be approximately normally distributed. Under this assumption, an approximate $100(1-\alpha)\%$ upper confidence limit on D is given by

$$UCL = \hat{D} + z_{1-\alpha} SD(\hat{D}), \quad (4.1)$$

where $\hat{D}$ = the estimated annual dose to a maximally exposed individual for the facility or group of facilities being studied,

$SD(\hat{D})$ = the computed standard deviation of $\hat{D}$, (a measure of the uncertainty in $\hat{D}$), and

$z_{1-\alpha}$ = the value from the standard normal distribution^(b) such that $(1-\alpha)100\%$ of the distribution lies to the left of the value.

The estimate $\hat{D}$ of D is obtained in stages. The ARCL methodology allows a maximally exposed individual to receive dose contributions from more than one scenario. We denote by $\hat{D}_s$ the estimated dose from scenario s, and by $SD(\hat{D}_s)$ the computed standard deviation of $\hat{D}_s$. We assume that the $\hat{D}_s$ variables (and levels of contamination within the scenarios) are statistically independent.^(a)

Then

$$\hat{D} = \sum_{s=1}^S \hat{D}_s \quad (4.2)$$

and

-
- (a) The UCL may be computed for the current value of dose, for the 100-yr decayed dose, etc. Also, the UCL formulas are not dependent on the 25 mrem/yr limit. Any limit may be used for comparison.
- (b) The standard normal distribution has mean zero and variance 1.

$$SD(\hat{D}) = \left\{ \sum_{s=1}^S [SD(\hat{D}_s)]^2 \right\}^{\frac{1}{2}} \quad (4.3)$$

Methods for computing $\hat{D}_s$ and $SD(\hat{D}_s)$ for each scenario are given in Section 4.3. Once $\hat{D}_s$ and $SD(\hat{D}_s)$ values are computed for a given scenario, a $100(1-\alpha)\%$ UCL may be computed for each D_s using a formula analogous to Equation (4.1). This will indicate whether the dose for a single scenario is sufficient to decide that RA is required.

4.2.2 Discussion

The uncertainty in $\hat{D}$ [measured by $SD(\hat{D})$] is assumed to be a result of sampling and measurement variability. The dose factors used in computing $\hat{D}$ are assumed to be known without error. This is certainly not the case, but it is beyond the scope of this report to consider the possible bias and uncertainty in the dose factor estimates.

The $100(1-\alpha)\%$ UCL in Equation (4.1) is computed using sample measurement data. Its interpretation is as follows: if the entire sampling, measurement, and UCL computation procedure were repeated many times on the same facility or site, the true value of D would be less than the computed UCL approximately $100(1-\alpha)\%$ of the time. Of course, we will only sample and compute the UCL once for a given situation, with a $100\alpha\%$ chance of the true value of D being larger than the calculated UCL.

It is desirable for $\hat{D}$ to be an accurate estimate of D , and for $SD(\hat{D})$ to be small. Because the dose factors are assumed to be known without error, the number of physical samples and how they are selected determines $\hat{D}$ and $SD(\hat{D})$. For example, it is quite natural to take more samples in more highly contaminated areas than in less-contaminated areas. However, unless done in the correct manner, $\hat{D}$ can be biased high and $SD(\hat{D})$ can be much too large (which could falsely indicate the need for RA).

-
- (a) Although this is not an entirely valid assumption for some situations, it should not significantly affect the results.

The solutions to these potential problems involve the use of stratified random sampling and the collection of a sufficient number of physical samples from each stratum. Stratified random sampling consists of dividing a contaminated area into subareas called strata, and performing simple random sampling within each stratum. Strata are defined so that mean concentration levels between the strata are quite different, and concentration variability within each stratum is as small as possible. Stratification for ARCL remedial action decision-making is discussed in Section 5.1. (The choice of size is discussed in Section 5.1).

4.3 COMPUTATION OF SCENARIO DOSE AND STANDARD DEVIATION ESTIMATES FOR STRATIFIED RANDOM SAMPLING

In this section formulas are given for computing $\hat{D}_s$ and $SD(\hat{D}_s)$ using radionuclide concentration data obtained from stratified random sampling at a facility. These formulas are valid for a given scenario. If the annual dose to a maximally exposed individual results from contributions from more than one scenario, the $\hat{D}_s$ and $SD(\hat{D}_s)$ values for the contributing scenarios must be combined, as discussed in Section 4.2.

4.3.1 Formulas and Notation

The notation used in this section is defined below. The estimated contribution of scenario "s" to the annual dose for a maximally exposed individual is given by

$$\hat{D}_s = \sum_{h=1}^H \sum_{i=1}^L d_{hi} y_{hi} \quad (4.4)$$

where

$$y_{hi} = m_h \bar{x}_{hi}, \quad (4.5)$$

$$\text{and } \bar{x}_{hi} = \frac{1}{n_{hi}} \sum_{\ell=1}^{n_{hi}} x_{hi\ell} \quad (4.6)$$

The standard deviation of $\hat{D}_s$ is given by

$$SD(\hat{D}_s) = \sum_{h=1}^H \left\{ \left[\sum_{i=1}^L d_{hi}^2 m_h^2 \frac{\hat{\sigma}_{hi}^2}{n_{hi}} + 2m_h^2 \left(\sum_{j < k}^L d_{hj} d_{hk} \frac{\hat{\sigma}_{hjk}}{n_{hjk}} \right) \right] \right\}^{\frac{1}{2}} \quad (4.7)$$

The notation used in the above equations is defined below

L = the total number of radionuclides present over all strata,

H = the number of strata for the scenario being considered,

d_{hi} = the dose factor (rem/yr per pCi) corresponding to radionuclide i in stratum h for the scenario under consideration,

y_{hi} = estimated inventory (pCi) of radionuclide i in the h th stratum for the scenario under consideration,

$\bar{x}_{hi}$ = estimated mean concentration (pCi/g or pCi/m²) of radionuclide i in the h th stratum,

x_{hil} = measured concentration value (pCi/g or pCi/m²) of radionuclide i in the l th sample taken from stratum h ,

m_h = the mass (or area, if concentrations on an area basis) of contaminated material (grams) in stratum h ,

n_{hi} = the number of samples from stratum h that were analyzed for radionuclide i ,

$\hat{\sigma}_{hi}^2$ = the estimated variance of concentration values of radionuclide i in stratum h . This is based on the n_{hi} concentration values x_{hik} of radionuclide i available from stratum h , and is calculated via the formula

$$\hat{\sigma}_{hi}^2 = \frac{1}{n_{hi}-1} \sum_{l=1}^{n_{hi}} (x_{hil} - \bar{x}_{hi})^2, \quad (4.8)$$

n_{hjk} = the number of samples from stratum h for which concentration values of both radionuclides j and k are available, and

$\hat{\sigma}_{hjk}$ = the estimated covariance of concentration values of radionuclides j and k in stratum h . This is based on the n_{hjk} concentration-value pairs $(x_{hj\ell}, x_{hk\ell})$ available from stratum h , and is calculated via the formula

$$\hat{\sigma}_{hjk} = \frac{1}{n_{hjk}-1} \sum_{\ell=1}^{n_{hjk}} (x_{hj\ell} - \bar{x}_{hj})(x_{hk\ell} - \bar{x}_{hk}). \quad (4.9)$$

The formula (4.7) for $SD(\hat{D}_S)$ is based on the assumption that the concentration level of a radionuclide in one stratum is independent of the levels of the radionuclide or other radionuclides in other strata. This assumption simplifies what would otherwise be a much more complicated formula that would require sophisticated techniques to estimate spatial correlations. It is believed that the between-strata correlations would not contribute significantly to the overall value of $SD(\hat{D}_S)$.

4.3.2 Decay Considerations

Equations (4.4) - (4.9) may be used to compute $\hat{D}_S$ and $SD(\hat{D}_S)$ for current contamination levels or for future decayed levels of contamination. For Hanford, a decay period (institutional control period) of t years is of interest. Formulas (4.4) - (4.9) can be used directly for decay situations if the decay factors $e^{-\lambda_i t}$ are applied to the raw radionuclide concentration values $x_{hi\ell}$ before performing any other calculations. However, it is also possible to compute $\hat{D}_S$ and $SD(\hat{D}_S)$ for decay, given the current-value summary statistics y_{hi} , $\hat{\sigma}_{hi}$, $\hat{\sigma}_{hij}$. The modified versions of Equations (4.4) and (4.7) are

$$\hat{D}_S(t) = \sum_{h=1}^H \sum_{i=1}^L d_{hi} (e^{-\lambda_i t}) y_{hi} \quad (4.10)$$

and

$$SD [\hat{D}_s(t)] = \left\{ \sum_{h=1}^H \left[\sum_{i=1}^L (e^{-\lambda_i t})^2 d_{hi}^2 m_h^2 \frac{\hat{\sigma}_{hi}^2}{n_{hi}} + 2 m_n^2 \sum_{j < k}^L d_{hj} d_{hk} e^{-(\lambda_j + \lambda_k) t} \frac{\hat{\sigma}_{hjk}}{n_{hjk}} \right] \right\}^{\frac{1}{2}} \quad (4.11)$$

where $D_s(t)$ is the annual dose to a maximally exposed individual involved in the scenario considered t years from now. The decay constants λ_i are given for various radionuclides in Table 5.3.

Once $\hat{D}_s(t)$ and $SD[\hat{D}_s(t)]$ are computed, they are then substituted into Equation (4.1) to yield the $100(1-\alpha)\%$ UCL for $D_s(t)$. Similarly, once values of $\hat{D}_s(t)$ and $SD[\hat{D}_s(t)]$ are computed for all scenarios contributing to the dose for a maximally exposed individual, they may be substituted into Equations (4.2) and (4.3). The results of Equations (4.2) and (4.3) are then substituted into (4.1) to yield the $100(1-\alpha)\%$ UCL on $D(t)$.

5.0 PROCEDURES FOR APPLYING ARCL TO A CONTAMINATED FACILITY

Application of the general method outlined in Section 2 to an actual facility can be a complicated undertaking. A multitude of interdependent calculations may be required for sites with multiple contaminated zones, various types of waste materials, and potential interactions with adjacent facilities. To permit uniform application of the concepts and scenarios presented in Section 2, a complete stepwise process is described in this section. The seven general activities required for this process are illustrated in Figure 5.1. Each of these activities is further described in Sections 5.1 through 5.7 of this section. Each of the seven basic activities is comprised of from one to five tasks. Each task, in turn, is composed of from one to eight procedures. A total of 25 separate procedures are detailed in a step-by-step manner. To aid in understanding the procedures, example worksheets of the ARCL evaluation of a Hanford facility are included in Appendix A. Because reference is made often to the nine worksheets in the text, Table 5.1 lists them and provides the figure numbers of the examples in this chapter. Copies of the worksheets are also provided in Appendix B.

Preparation of a comprehensive site radionuclide inventory is the necessary first activity. This activity includes sampling and monitoring procedures that are outside the scope of this manual, but mention is made of where these tasks fit into the entire process.

The radionuclide inventory is adjusted in the second activity for application to each of the three controlling exposure scenarios for each stratum of contaminated material. This adjustment can include corrections for radioactive decay if the proposed action is not immediate decontamination and release of the facility.

The calculation of individual radiation doses for each scenario and stratum proceeds concurrently for the three major exposure scenarios, as illustrated in Figure 5.1. The dose estimates from the residential/home-garden and agricultural scenarios are summed, and the sum compared to the resource recycle result. Then, for the more restrictive dose, the $100(1-\alpha)\%$ upper confidence limit is compared to the mandated dose limit to determine if the proposed decommissioning alternative is acceptable.

FIGURE 5.1. Required Activities During an ARCL Calculation

TABLE 5.1. Worksheets Used for ARCL Analysis of Contaminated Facilities

<u>Form</u>	<u>Worksheet Title</u>	<u>Figure Number in Text</u>
A	Summary Statistics Table--Estimated Means Standard Deviations, and Covariances per Stratum	Figure 5.3
B	Source-Term Worksheet	Figure 5.5
C	Availability Correction Worksheet	Figure 5.7
D	Stratum Dose Calculation Worksheet	Figure 5.8
E	Dose Variance Calculation Worksheet	Figure 5.9
F	Area Correction Worksheet	Figure 5.11
G	Depth Correction Worksheet	Figure 5.12
H	Mass Correction Worksheet	Figure 5.14
I	Final ARCL Worksheet	Figure 5.16

Careful completion of each of the seven activities will provide a completely documented selection of the final site disposal alternative.

5.1 ACTIVITY 1 - SITE RADIONUCLIDE INVENTORY CALCULATION

The five basic tasks required to fully define the radionuclide inventory of a particular site or facility are illustrated in Figure 5.2. Procedures for implementing these tasks are listed in Table 5.2. It should be noted that details of Tasks 1.2 through 1.4 are outside the scope of this manual. Although portions of Task 1.2 are outlined here, details for Task 1.3 are to be provided in a separate report being funded by this project, and procedures for Task 1.4 are assumed to be available to decommissioning personnel.

5.1.1 Task 1.1 - Identify Primary and All Secondary Facilities Within the Site

Generally, the primary facility will be the facility under consideration for decommissioning and for which the ARCL test is being calculated. For any particular primary facility, the most highly contaminated 1-ha area

FIGURE 5.2. Tasks Required to Perform Activity 1 - Site Radionuclide Inventory Calculation. (Asterisks (*) indicate that detailed discussion the task is outside the scope of this manual)

TABLE 5.2. Procedures Required for Activity 1 - Tasks 1.1 - 1.5

Task 1.1 Identify Primary and All Secondary Facilities Within Site

1.1.1 Selection of 1-ha Site

1.1.2 Consideration of Specific Types of Secondary Facility

Task 1.2 Locate Contamination Within Each Facility and Identify Strata

1.2.1 Generic - Locating Contamination

1.2.2 Generic - Identification of Strata

Task 1.3 Determine Required Number of Samples

1.3.1 Generic - Use of ARCL Sample Tables

Task 1.4 Collect and Analyze Samples

1.4.1 Generic - Sample Collection

1.4.2 Generic - Sample Analysis

Task 1.5 Compute Mean Concentrations and Associated Statistics

1.5.1 Preparation of Summary Statistics Tables

containing that facility will be identified. Other contaminated facilities within that 1-ha area will be referred to as secondary facilities.

Procedure 1.1.1 Selection of 1-Ha Site

The site to be considered in the ARCL calculation should contain the entire primary facility. If the primary facility is too large to be fully contained within 1 ha (10,000 m²), then the primary facility should be subdivided into 1-ha parcels. As a rule, the 1-ha area should be a regular polygon, no more than about three times as long as it is wide. Gerrymandering to include or exclude particular areas is discouraged. The 1-ha site may overlap that of another facility if several contaminated facilities are clustered in one area.

Procedure 1.1.2 Consideration of Specific Types of Secondary Facility

The purpose of ARCL determinations is to permit eventual unrestricted release of the land. If the primary facility facilities that are not intended to be released (such as reactor blocks or

permanent waste burial grounds), then these "unreleasable facilities" should be specifically excluded from the determination of the 1-ha site.

5.1.2 Task 1.2 - Locate Contamination Within Each Facility and Identify Strata

For the primary and the secondary facilities in the 1-ha region, the extent (i.e., location, surface area, depth, etc.) of the contamination should be characterized. Portions of facilities classified as not contaminated should be such that, were physical samples to be collected and analyzed for specific radionuclides, "not-detectable" results would be obtained for all radionuclides.

The purpose of this step is to determine where the contamination is and where it is not. Noncontaminated areas of facilities may not have physical samples collected from them,^(a) so it is important that only noncontaminated areas be so identified.

Actual procedures for surveying facilities are beyond the scope of this manual. Only a generic procedure is described.

Procedure 1.2.1 GENERIC - Locating Contamination

Facilities should be completely surveyed with field instrumentation. Areas with known histories of contamination, areas with measurable contamination, and representative noncontaminated areas should be identified.

Procedure 1.2.2 GENERIC - Identification of Strata

For the primary and each secondary facility, identify the rooms, areas, or portions of the facility having similar concentration levels. To do so, use knowledge of the operating history of the facility, survey instrument readings, prior or preliminary sample analysis results, etc. Group the areas having similar concentration levels into strata.

For ARCL testing purposes, it is most important that the information used to stratify each facility correlate well with the

(a) It is wise to collect and analyze a few samples from such areas to verify noncontamination.

actual concentration levels of the dominating radionuclide(s). In practice, stratification may depend on physical or natural boundaries (e.g., rooms in a building which had the same function, or sludge, concrete, fill, etc., in a retention basin), as well as on concentration levels. Stratification provides the most benefits when the strata mean concentrations (of the dominant radionuclides) are quite different, and when the standard deviations of the concentration distributions (of the dominant radionuclides) within each stratum are small.

The purpose of this step is to provide for better characterization of the contamination, with the sampling and analysis effort expended where it will do the most good. The use of stratified random sampling will allow the ARCL dose to be estimated more accurately and with better precision than with ordinary simple random sampling.

5.1.3 Task 1.3 - Determine Required Number of Samples

Proper application of the ARCL method requires that an adequate number of physical samples be taken and radionuclide concentration determinations be performed. Step-by-step procedures for determining the number of analyses required for good statistical calculations are being prepared and will be presented in a separate report. A generic procedure is listed here to highlight the importance of this task.

Procedure 1.3.1 GENERIC - Use of ARCL Sample Size Tables

Use ARCL sample size tables or formulas described in Piepel (1986) to select the number of physical samples to be collected from each facility. Use sample size allocation formulas to determine what fraction of the total sample size should be taken from each stratum.

The data needs for estimating the radionuclide inventories of each facility should also be considered at this time. The ARCL sample size indicated for a given facility may be quite small (e.g., 2 or 3 samples) if it has little contamination in the dominant radionuclide. However, the need to estimate all inventories within a certain percentage of the true value may indicate that a larger sample size (than that required for ARCL testing) is needed.

In most cases, taking both ARCL and inventory estimation needs into account, the minimum number of samples per stratum should be at least 5. Otherwise, a final decision should consider the sample size numbers indicated by both ARCL and inventory estimation sample-size tables/formulas.

5.1.4 Task 1.4 - Collect and Analyze Samples

Sample collection and analysis procedures are assumed to be available to decommissioning personnel. Only generic procedure titles are included here to illustrate the importance of this step in ARCL determinations.

Procedure 1.4.1 GENERIC - Sample Collection

Collect the number of physical samples from each stratum as determined in Procedure 1.3.1.

Procedure 1.4.2 GENERIC - Sample Analysis

Analyze each sample for every radionuclide present in the facility, if possible. It is extremely important that direct or indirect (e.g., via double sampling) measurements be made on each sample for the dominant radionuclides.

5.1.5 Task 1.5 - Compute Mean Concentrations and Associated Statistics

Once sufficient samples have been obtained, the basic input values to the ARCL calculation of radionuclide inventory can be calculated. This must be done for each identified stratum within each facility in the 1-ha site.

Procedure 1.5.1 Preparation of Summary Statistics Table

Compute radionuclide concentration means, standard deviations, and covariances, and then fill out a Summary Statistics Table (see Figure 5.3) for each stratum.

Note that the n_{hi} and the n_{hij} values for a given stratum will all be equal if each sample is analyzed for every radionuclide as suggested in Procedure 1.4.2. The ARCL test procedure can be carried out as long as n_{hi} , $n_{hij} \geq 3$, but the results may be highly questionable, especially if only a small number of samples were analyzed for the dominant radionuclide(s).

FIGURE 5.3. Form A - Summay Statistics Table

The calculation of the means and associated statistics should preferably be done automatically with a standard computer software package, with the results retained for the ARCL test process formulae. For completeness, the basic equations are repeated here from Section 4.3.

The mean concentration of radionuclide i in stratum h , $\bar{x}_{hi}$, is given by

$$\bar{x}_{hi} = \frac{1}{n_{hi}} \sum_{k=1}^{n_{hi}} x_{hik} \quad (5.1)$$

where x_{hik} = the concentration value of radionuclide i from the k th sample in stratum h for which the concentration of radionuclide i was determined (measured in pCi/g), and

n_{hi} = the number of samples in stratum h for which the concentration of radionuclide i was determined.

The estimated standard deviation ($\hat{\sigma}_{hi}$) of the concentration values x_{hik} for radionuclide i in stratum h , is given by

$$\hat{\sigma}_{hi} = \left[\frac{1}{n_{hi} - 1} \sum_{k=1}^{n_{hi}} (x_{hik} - \bar{x}_{hi})^2 \right]^{1/2} \quad (5.2)$$

The estimated covariances ($\hat{\sigma}_{hij}$) of concentration values of radionuclides i and j in stratum h are estimated as

$$\hat{\sigma}_{hij} = \frac{1}{n_{hij} - 1} \sum_{k=1}^{n_{hij}} (x_{hik} - \bar{x}_{hi})(x_{hjk} - \bar{x}_{hj}) \quad (5.3)$$

where n_{hij} is the number of samples from stratum h for which concentration values of both radionuclides i and j are available.

5.2 ACTIVITY 2 - EXPOSURE SCENARIO SOURCE-TERM PREPARATION

The three basic tasks required to prepare the source terms for the three ARCL exposure scenarios from the site inventories by stratum are illustrated in Figure 5.4. These tasks may be performed concurrently because each task requires the same input information. The procedures for implementing these tasks are listed in Table 5.3.

5.2.1 Task 2.1 - Prepare Source Terms for Residential/Home-Garden Scenario

The residential/home-garden scenario is the most important scenario for instances of buried rubble or waste and for soil contamination. As described in Section 2, buried material is assumed to be excavated to the surface, distributed, and individuals are assumed to reside continuously in the contaminated area thus created. Materials distributed on the surface are assumed to contribute to the inventory unless they are of a form that is readily recognizable as a valuable recyclable commodity, in which case they are considered in the resource recycle scenario (Section 5.2.3). The following procedures are performed for each applicable stratum.

FIGURE 5.4. Tasks Required to Perform Activity 2
Exposure Scenario Source-Term Preparation

TABLE 5.3. Procedures Required for Activity 2, Tasks 2.1 - 2.3

- Task 2.1 Prepare Source Terms for Residential/Home-Garden Scenario
 - 2.1.1 Location of Scenario-Specified Excavation Volume
 - 2.1.2 Calculation of Residential/Home-Garden Scenario Source Terms
 - 2.1.3 Decay Adjustments to Source Term
- Task 2.2 Prepare Source Term for Agricultural Scenario
 - 2.2.1 Calculation of Agricultural Scenario Source Terms
 - (2.1.3) Decay Adjustments to Source Term
- Task 2.3 Prepare Source Term for Resource Recycle Scenario
 - 2.3.1 Calculation of Resource Recycle Source Term
 - (2.1.3) Decay Adjustments to Source Term

Procedure 2.1.1 Location of Scenario-Specified Excavation Volume

Identify the location of the 10 x 20 x 5-m basement excavation volume within the 1-ha site expected to yield the largest dose. For situations where a single radionuclide dominates the dose calculations, this is equivalent to identifying the 10 x 20 x 5-m volume containing the largest inventory of the dominant nuclide. Normally, the product of inventory and dose factor is used to determine the location of greatest dose potential. This accounts for differences in radionuclide inventory and hazard.

Procedure 2.1.2 Calculation of Residential/Home-Garden Scenario Source Terms

Determine the mass of the stratum (considering only contaminated material) that is in the selected 10 x 20 x 5-m excavation volume. This step makes use of the precise characterization information (areas, depths, concentrations, etc.) of contaminated portions of facilities performed in Activity 1. The equation for calculating the inventory, y_i , of radionuclide i by stratum is

$$y_i = m_h \bar{x}_{hi} \quad (5.4)$$

The standard deviation on the estimated inventory of radionuclide i by stratum is calculated using

$$D(y_{hi}) = \left[\frac{m_h^2 \hat{\sigma}_{hi}^2}{n_{hi}} \right]^{1/2} \quad (5.5)$$

where the mean concentration, $\bar{x}_{hi}$, the concentration standard deviation of radionuclide i in stratum h , $\hat{\sigma}_{hi}$, and the number of samples of radionuclide i in stratum h , n_{hi} , are taken from Worksheet A, the Summary Statistics Tables, (see Figure 5.3) for each stratum prepared in Procedure 1.5.1, and where m_h = the mass of stratum h .

Enter the results on the Source-Term Worksheet (Form B), shown in Figure 5.5. Separate worksheets should be used for each stratum considered. If there is insufficient space on a worksheet, simply append additional copies.

For strata including radionuclides encased in material making them unavailable (for instance, contamination inside of closed pipes), consideration in the source term must be given to the ultimate availability of the contained radionuclides. The effects of radioactive decay until the material becomes available should be considered. For selected situations, the increasing availability compounded with decay may mean that the potential dose rate could peak at a time other than that chosen for site release. Iterative examination of the dose rate may be required to assure that the most restrictive time period is used in the calculation.

Procedure 2.1.3 Decay Adjustments to Source Term

The entries in Items 10 and 11 of Form B, Figure 5.5, represent the currently measured source term. However, decommissioning planning may incorporate consideration of a period of institutional control. The radionuclide inventory would be decayed to some level lower than that existing today. The effect of radioactive decay on the source inventory for a mixture of radionuclides is next determined.

5.14

- [illegible]

(c) Calculate as Item 4 times Item 7.

$$(d) \text{ S. D. } (y_{hi}) = \left(\frac{\frac{2}{n} \frac{2}{hi}}{n} \right)^{1/2}$$

(f) Calculate as Item 11 times $e^{-\lambda_i t}$ where λ_i = decay constant for radionuclide i (from Table 5.2.3) and t = number of years of decay (Item 5).

FIGURE 5.5. Form B - Source-Term Worksheet

To calculate the decayed source term and standard deviations, simply multiply the entries in Items 10 and 11 of Form B, Figure 5.5, by the term $(e^{-\lambda_i t})$ where λ_i equals the decay constant for radionuclide i (from Table 5.4) and t equals the number of years of radioactive decay (Item 5 of Form B, Figure 5.5).

5.2.2 Task 2.2 - Prepare Source Terms for Agricultural Scenario

Application of the agricultural scenario is a procedural way of assuring that no contribution from residual contamination within the 1-ha site is omitted from consideration, since it is not directly accounted for in the residential/home-garden scenario. The same person assumed to be living in the contamination from the residential/home-garden excavation is simultaneously exposed to any other sources through the crop-uptake pathway. The source-term calculation, and decay adjustments, are parallel in structure for the agricultural scenario with those for the residential/home-garden scenario. The decay procedure is, in fact, identical and so is not repeated. For the rare case of surface contamination, source terms for external and inhalation doses can be prepared in an analogous fashion.

Procedure 2.2.1 Calculation of Agricultural Scenario Source Terms

Determine the mass of the stratum (considering only contaminated material) remaining in the 1-ha area, once the 10 x 20 x 5-m volume has been removed. For certain circumstances, it may be advantageous to define more than one stratum, as a function of depth. The equations for calculating the total source term of each radionuclide are those described in Procedure 2.1.1. The results should be entered on separate copies of Form B, the Source Term Worksheet (Figure 5.5), by stratum.

5.2.3 Task 2.3 - Prepare Source Terms for Resource Recycle Scenario

Selected waste forms, such as piping, activated metal components, and other sources of contained activity, are not amenable to the agricultural activities of the residential/home-garden or agricultural scenarios until the waste form decomposes. The wastes may be stable solids which would not be dispersed during excavation and which would not allow root uptake by plants. Materials such as this may be included in the agricultural scenario by way of

TABLE 5.4. Decay Constants for Selected Radionuclides (yr⁻¹)

<u>Nuclide</u>	<u>Constant</u>	<u>Nuclide</u>	<u>Constant</u>
³ H	5.6E-2	¹³⁴ Cs	3.4E-1
¹⁴ C	1.2E-4	¹³⁵ Cs	3.4E-7
³⁶ Cl	2.2E-6	¹³⁷ Cs+D	2.3E-2
⁵⁷ Co	9.3E-1	¹⁴⁴ Ce	8.9E-1
⁶⁰ Co	1.3E-1	¹⁵² Eu	5.2E-2
⁵⁵ Fe	2.6E-1	¹⁵⁴ Eu	8.5E-2
⁵⁹ Fe	5.6E+0	²²⁶ Ra	4.3E-4
⁵⁹ Ni	8.7E-6	²³² Th	5.0E-4
⁶³ Ni	6.9E-3	²³⁵ U+D	9.8E-10
⁹⁰ Sr+D	2.4E-2	²³⁸ U+D	1.6E-10
⁹³ Mo	2.0E-4	²³⁷ Np+D	3.2E-7
⁹⁴ Nb	3.5E-5	²³⁸ Pu	7.9E-3
⁹⁹ Tc	3.2E-6	²³⁹ Pu	2.9E-5
¹²⁴ Sb	4.2E+0	²⁴¹ Am	1.6E-3
¹²⁵ Sb+D	2.5E-1		

Form C, the Availability Correction Worksheet (Figure 5.7), or they may be included in the resource recycle scenario. This scenario considers exposures to persons other than the onsite resident, and the doses are calculated independently and compared to the overall annual dose limit without being summed with the onsite values. The mechanics of the source-term preparation, and optional decay correction, are the same as described for the other two scenarios in Sections 5.2.1 and 5.2.2. The decay procedure 2.1.3 is used if needed.

Procedure 2.3.1 Calculation of Resource Recycle Source Terms

Determine the mass of material that could potentially be recycled if discovered by the onsite resident. The general equations for converting from concentration to source term by stratum are those described in Procedure 2.1.2. The results should be entered on separate copies of Form B, the Source Term Worksheet (Figure 5.5), by stratum. This procedure is optional--if there are no waste forms within the 1-ha site which

can be reasonably considered to be recyclable (e.g., for crib sites), this scenario may be omitted.

5.3 ACTIVITY 3 - RESIDENTIAL/HOME-GARDEN SCENARIO CALCULATION

The two tasks required to perform the actual calculation of the dose estimate to the resident individual for the residential/home-garden scenario are illustrated in Figure 5.6. The procedures for implementing these tasks are listed in Table 5.5.

5.3.1 Task 3.1 - Apply Availability Correction to Radionuclide Dose Factors

Most waste forms for which an availability correction would apply should also be considered for the resource recycle scenario. Wastes such as activated metals may more properly be considered there. If they are included in the residential/home-garden scenario, the dose factors of specific radionuclides contained in the actuated metals may be reduced by a factor relating their corrosional release rate to soil.

Procedure 3.1.1 Application of Availability Correction Worksheet

Radionuclides contained in chunks of metal will not be available for resuspension or root uptake even if distributed on the surface by excavation. Metal pieces corrode slowly with time, gradually releasing the contained materials. For a particular waste strata (for instance, a pipe), determine a representative waste form thickness. This may be an average over several sizes. Calculate the reduction factor RF for radionuclides incorporated in this waste form at the time of interest, t , using the equation

$$RF = 2t \text{ (corrosion rate, cm/yr)} / \text{(thickness, cm)} \quad (5.6)$$

Note that this correction should never be greater than 1.

Enter this value only for affected radionuclides in Column 5 of Form C, the Availability Correction Worksheet (shown in Figure 5.7). All other entries in Column 5 should be 1.0. Calculate the adjusted inhalation/external and ingestion dose factors and final scenario dose

FIGURE 5.6. Tasks Required to Perform Activity 3 - Residential/ Home-Garden Scenario Calculation

TABLE 5.5. Procedures Required for Activity 3, Tasks 3.1 - 3.2

Task 3.1 Apply Availability Correction

3.1.1 Application of Availability Correction Worksheet

Task 3.2 Apply Residential/Home-Garden Dose Factors

3.2.1 Application of Stratum Dose Calculation Worksheet to Residential/Home-Garden Scenario

3.2.2 Calculation of Variance of Estimated Dose

factors. The scenario dose factors for the residential/home-garden scenario are the sum of Columns 7a and 7b.

5.3.2 Task 3.2 - Obtain Residential/Home-Garden Dose Estimates

Radiation dose estimates are calculated for each stratum.

Procedure 3.2.1 Application of Stratum Dose Calculation Worksheet to Residential/Home-Garden Scenario

The Stratum Dose Calculation Worksheet (Form D) is illustrated as Figure 5.8. Radionuclide names, source terms by stratum, and standard deviations by stratum should be entered from (Form B) the Source-Term Worksheets (Figure 5.5). The scenario dose factors for Column 7 should be entered from Form C, the Availability Correction Worksheet (Figure 5.7, Column 8). The total estimated dose can be obtained as the sum of the entries in Column 8 of Form D, the Stratum Dose Calculation Worksheet (Figure 5.8); the values in Column 10 of that worksheet are

not necessary for the remainder of the calculation, but may be calculated if information on the dominant radionuclides is desired. Values for Column 10 may be calculated as Column 8 plus $z_{1-\alpha}$ times Column 9, where $z_{1-\alpha}$ may be obtained from Table 5.6, below.

TABLE 5.6. Standard Normal Distribution Values $z_{1-\alpha}$ Such that $(1-\alpha)100\%$ of the Distribution Lies to the Left of $z_{1-\alpha}$

<u>$(1-\alpha) 100\%$</u>	<u>$z_{1-\alpha}$</u>
99	2.33
97.5	1.96
95	1.645
90	1.282

The value for Item 11 of Form D, the Stratum Dose Calculation Worksheet (Figure 5.8), is required if the upper confidence limit for a stratum, scenario, or site is to be calculated. The calculation of Item 11 is complex, and is described in Procedure 3.2.2.

Procedure 3.2.2 Calculation of Variance of Estimated Dose:

This procedure is to define the calculation required to compute Item 11 of Form D, the Stratum Dose Calculation Worksheet (see Figure 5.8). As described in Section 4, the value of the UCL of dose for stratum h is determined as

$$UCL_h = \hat{D}_h + SD(\hat{D}_h) * z_{1-\alpha} \quad (5.7)$$

where the estimated standard deviation of $\hat{D}_h$, $SD(\hat{D}_h)$ is determined from the equation

$$SD(\hat{D}_h) = [Var(\hat{D}_h)]^{1/2} \quad (5.8)$$

and

$$Var(\hat{D}_h) = \sum_{i=1}^L [SD(d_{hi}y_{hi})]^2 + 2 \sum_{j < k} \left(d_{hj}d_{hk} \frac{\hat{\sigma}_{hjk}}{n_{hjk}} e^{-[\lambda_j + \lambda_k]t} \right) \quad (5.9)$$

where $SD(d_{hi}y_{hi})$ = the standard deviation of the individual radionuclide dose estimates for stratum h, from Column 9 of Form D, the Stratum Dose Calculation Worksheet (Figure 5.3),

m_h = the mass of stratum h, Item 4 of Form B from the Scenario Source Term Worksheet (Figure 5.3),

d_{hi} = the scenario dose factor for radionuclide i, from Column 7 of Form D, the Stratum Dose Calculation Worksheet (Figure 5.8),

σ_{hjk} = the estimated covariances between concentrations of radionuclides j and k in stratum h from Column 10 of Form A, the Summary Statistics Table (Figure 5.3),

n_{hjk} = the number of samples from strata h for which both radionuclides j and k were analyzed, from Column 11 of Form A, the Summary Statistics Table (Figure 5.3),

L = the total number of radionuclides, from Form A, the Summary Statistics Table (Figure 5.3),

λ_j = the decay constant for radionuclide j or k, from Table 5.4,

t = the time after the measurements were taken, corresponding to the time for which the dose, D, was estimated, from Item 5 of Form B, the Source-Term Worksheet (Figure 5.5).

When all values are tabulated, the variance may be calculated and inserted in Item 11 of the worksheet. Form E, the Dose Variance Calculation Worksheet (Figure 5.9), can be used to solve Equation 5.9. However, the use of a computer is encouraged for this calculation.

The $100(1-\alpha)\%$ UCL on dose can be calculated using any desired confidence limit. The multiplier $z_{1-\alpha}$ for several commonly chosen confidence limits is given in Table 5.6.

Form E - Dose Variance Calculation Worksheet

1. Facility and Stratum:

2. Preparer's Name: _____ 3. Date Prepared: _____ 4. Stratum Mass (m_h): _____ g 5. Decay Time (t): _____ yr

j	6.* Radio nucli de h _j	7.* SD(d _{h_jy_{h_j})}	8.* λ _{h_j}	9.* d _{h_j} \ d _{h_k}	A _{hjk} #								10.■ d _{h_j} Σd _{h_k} A _{hjk}
					k: 2	3	4	5	6	7	8		
1					e ⁻	e ⁻	e ⁻	e ⁻	e ⁻	e ⁻	e ⁻		
2						e ⁻	e ⁻	e ⁻	e ⁻	e ⁻	e ⁻		
3							e ⁻	e ⁻	e ⁻	e ⁻	e ⁻		
4								e ⁻	e ⁻	e ⁻	e ⁻		
5									e ⁻	e ⁻	e ⁻		
6										e ⁻	e ⁻		
7											e ⁻		
8													

$\Sigma [SD(d_{h_j}y_{h_j})]^2 =$ _____
 $2m_h^2 =$ _____
 $\Sigma d_{h_j} \Sigma d_{h_k} A_{hjk} =$ _____

$$11. \text{Var}(\hat{D}_h) = \{ \Sigma [SD(d_{h_j}y_{h_j})]^2 \} + \{ 2m_h^2 \} \{ \Sigma d_{h_j} \Sigma d_{h_k} A_{hjk} \} = \boxed{} + \boxed{} \boxed{} = \boxed{} = \text{Var}(\hat{D}_h)$$

* Column 6 [Radionuclide j and k]--from Form A, Col. 4.
 Column 7 [SD(d<sub>h_jy_{h_j})]--from Form D, Col. 9.
 Column 8 [λ_{h_j} and λ_{h_k}]--from Table 5.3.
 Column 9 [d_{h_j} and d_{h_k}]--from Form D, Col. 7.</sub>

$A_{hjk} = \frac{\hat{\sigma}_{hjk}}{n_{hjk}} e^{-\theta}$, where $\hat{\sigma}_{hjk}$ is from Form B, Col. 10,
 n_{hjk} is from Form B, Col. 11, and
 $\theta = (\lambda_{h_j} + \lambda_{h_k})t$.
 ■ Column 10--calculated as $d_{h_j} \Sigma d_{h_k} A_{hjk}$.

FIGURE 5.9. Form E - Dose Variance Calculation Worksheet

5.4 ACTIVITY 4 - AGRICULTURAL SCENARIO CALCULATION

The four tasks required to perform the calculation of the dose estimate to the resident individual for the scenario of exposure to all other residual contamination on the 1-ha site, exclusive of that contained in the 10 x 20 x 5-m excavation, are illustrated in Figure 5.10. The procedures for implementing these tasks are listed in Table 5.7.

5.4.1 Task 4.1 - Apply Availability Correction to Radionuclide Dose Factors for Agricultural Scenario

The availability correction as described for the residential/home-garden scenario also applies to radionuclides contained within activated components for the agricultural scenario. The Availability Correction Worksheet (shown in Figure 5.7), with its controlling Procedure 3.1.1, is used. Because there is assumed to be no contamination of surface soils, as should usually be the case, calculation of the inhalation/external dose factors may be omitted. For the special case where surface soil contamination does exist, the inhalation/external dose factors must be included. These can be obtained from Table 2.1, with corrections for area being applicable. Column 8 of the Availability Correction Worksheet (Figure 5.7) may be omitted, except for cases of surface contamination.

5.4.2 Task 4.2 - Apply Area Correction to Radionuclide Dose Factors

The ingestion dose factors derived in Section 2 of this report are presented in terms of rem/pCi, where the activity is implicitly assumed to be distributed over an area of 1 ha. If strata for the agricultural scenario covers an area less than 1 ha, then the dose factors may be reduced using the Area Correction Worksheet, shown in Figure 5.11, for those strata.

Procedure 4.2.1 Application of Form F - Area Correction Worksheet

The derivation of the area correction factor is discussed in Section 2. Dose factors for all radionuclides are adjusted by the same area correction. Dose factors input to Form F, the Area Correction Worksheet (Figure 5.11) should come from Form C, the Availability Correction Worksheet (Figure 5.7). The equation given for Item 5 in the Area Correction Worksheet is simply the parameterization of the graph shown in Figure 2.3.

FIGURE 5.10. Tasks Required to Perform Activity 4 - Agricultural Scenario Option

TABLE 5.7. Procedures Required for Activity 4, Tasks 4.1 - 4.4

- Task 4.1 Apply Availability Correction to Radionuclide Dose Factors
 - (3.1.1) Application of Availability Correction Worksheet
- Task 4.2 Apply Area Correction to Radionuclide Dose Factors
 - 4.2.1 Application of Area Correction Worksheet
- Task 4.3 Apply Depth Correction to Radionuclide Dose Factors
 - 4.3.1 Application of Depth Correction Worksheet
- Task 4.4 Apply Agricultural Scenario Dose Factors
 - 4.4.1 Application of Stratum Dose Calculation Worksheet to Agricultural Scenario
 - (3.2.2) Calculation of Variance of Estimated Dose

5.4.3 Task 4.3 - Apply Depth Correction to Radionuclide Dose Factors

The ingestion dose factors derived in Section 2 of this report are normalized to 100% root uptake, which implies that the contaminated soil in which the plants are growing extends completely to the surface. For sites with a clean overburden layer, an adjustment may be made to the ingestion portion of the dose via the dose factors, using Form G, the Depth Correction Worksheet (Figure 5.12). This correction must be made for each stratum. Note that "overburden" includes overlying strata, if considered.

Procedure 4.3.1 Application of Depth Correction Worksheet

The derivation of the depth correction factor is discussed in Section 2. Ingestion dose factors for all radionuclides for one stratum are adjusted by the same depth correction. Dose factors input to Form G, the Depth Correction Worksheet (Figure 5.12), should come from Form F, the Area Correction Worksheet (Figure 5.11), if used, otherwise from Form C, the Availability Correction Worksheet (Figure 5.7). The equation presented in Item 5 on Form G, the Depth Correction Worksheet, is simply a parameterization of the graph presented in Figure 2.2.

Form F - Area Correction Worksheet

1. Facility:
2. Preparer's Name:
3. Date Prepared:
4. Area of Contamination Within 1-ha Site _____ m² = _____ ha
5. Correction Factor = $0.1 + 9$ (Item 4) = _____.

6. <u>Radionuclide</u>	7. <u>Ingestion</u> <u>DF(rem/yr/pCi) (a)</u>	8. <u>Corrected Ingestion</u> <u>DF(rem/yr/pCi) (b)</u>
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____

- (a) From Availability Correction Worksheet Form C.
(b) Calculated as Item 5 times Column 7.

FIGURE 5.11. Form F - Area Correction Worksheet

Form G - Depth Correction Worksheet

1. Facility:
2. Preparer's Name:
3. Date Prepared:
4. Depth of Clean overburden _____ m.
5. Calculate depth correction factor (DCF) as

$$DCF = e^{-0.92(\text{Depth})} = \underline{\hspace{2cm}}.$$

If Depth > 10 meters DCF = 0.0.

6. <u>Radionuclide</u>	7. <u>Ingestion</u> <u>DF(rem/yr/pCi)(a)</u>	8. <u>Corrected Ingestion</u> <u>DF(rem/yr/pCi)(b)</u>
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
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_____	_____	_____
_____	_____	_____
_____	_____	_____

- (a) From Form F - Area Correction or Form C - Availability Worksheet.
(b) Calculated as Item 5 times Column 7.

5.4.4 Task 4.4 - Obtain Agricultural Scenario Dose Estimates

Apply the Dose Calculation Worksheet to obtain the dose estimates.

Procedure 4.4.1 Application of Stratum Dose Calculation Worksheet to Agricultural Scenario

The radionuclide names, source terms, and standard derivations should be entered on Form D, the Stratum Dose Calculation Worksheet (Figure 5.8), from Form B, the Source Term Worksheet (Figure 5.5). The scenario dose factors should be entered from Form G, the Depth Correction Worksheet (Figure 5.11). For the special case of contaminated surface soil, the dose factors may be obtained directly from Form G, the Availability Correction Worksheet (Figure 5.7). The total estimated dose can be obtained on Form D, the Stratum Dose Calculation Worksheet (Figure 5.8), as the sum of the entries in Column 8; Item 11 on that worksheet must be calculated using Procedure 3.2.2.

5.5 ACTIVITY 5 - RESOURCE RECYCLE CALCULATION

The two tasks required to perform the calculation of the dose estimates to the individual recycling contaminated metals taken from the site are illustrated in Figure 5.13. The procedures for implementing these tasks are listed in Table 5.8.

5.5.1 Task 5.1 - Apply Mass Correction to Radionuclide Dose Factors

The dose factors derived in Section 2 of this report are presented in terms of rem/pCi, where the activity is implicitly assumed to be distributed throughout 200 Mg (200 metric tons) of contaminated metal. If the quantity of material estimated to be recoverable is other than 200 Mg, a correction

FIGURE 5.13. Tasks Required to Perform Activity 5 - Resource Recycle Scenario Calculation

TABLE 5.8. Procedures Required for Activity 5, Tasks 5.1 - 5.2

Task 5.1 Apply Mass Correction to Radionuclide Dose Factors

5.1.1 Application of Mass Correction Worksheet

Task 5.2 Apply Resource Recycle Dose Factors

5.2.1 Application of Stratum Dose Calculation Worksheet
to Resource Recycle Scenario

(3.2.2) Calculation of Variance of Estimated Dose

must be made to the dose factors, before the doses can be calculated, using Form H, the Mass Correction Worksheet (Figure 5.14). No upper bound is currently postulated on the recycle mass.

Procedure 5.1.1 Application of Mass Correction Worksheet

The mass correction factor is the ratio of the mass of recyclable metal to 200 Mg. Dose Factors input to the worksheet (Figure 5.13) come from Table 2.2.

5.5.2 Task 5.2 - Obtain Resource Recycle Dose Estimate

Apply the Stratum Dose Calculation Worksheet (Figure 5.8) to obtain the dose estimates.

Procedure 5.2.1 Application of Stratum Dose Calculation Worksheet to
Resource Recycle Scenario

The radionuclide names, source terms, and standard deviations should be entered on Form D, the Stratum Dose Calculation Worksheet (Figure 5.8), from Form B, the Source Term Worksheet (Figure 5.5). The scenario dose factors should be entered from Form H, the Mass Correction Worksheet (Figure 5.14). In the Stratum Dose Calculation Worksheet, Item 11 must be calculated using Procedure 3.2.2.

5.6 ACTIVITY 6 - SELECTION OF CONTROLLING SCENARIO

The three tasks necessary to select the controlling scenario between the onsite individual and the offsite recycling worker are shown in Figure 5.15. The procedures used to make the selection are listed in Table 5.9.

1. Facility:
2. Preparer's Name:
3. Date Prepared:
4. Mass of contaminated metal, _____ Mg.
5. Calculate mass correction factor (MCF) as

[illegible]

(b) Calculated as Item 5 times Column 7.

5.31

FIGURE 5.15. Tasks Required to Perform Activity 6 - Selection of Controlling Scenario

TABLE 5.9. Procedures Required for Activity 6, Tasks 6.1 - 6.3

Task 6.1 Summation of Strata for Each Scenario

6.1.1 Summation of Strata

Task 6.2 Summation of Residential/Home Garden and Agricultural Scenario Doses

6.2.1 Summation of Multiple Scenarios

Task 6.3 Selection of the Most Restrictive Exposure Scenario

6.3.1 Restrictive Scenario Selection

5.6.1 Task 6.1 - Summation of Strata for Each Scenario

The total estimated dose, and its variance, for each scenario are computed using Procedure 6.1.1. This procedure shall be used separately for each of the three scenarios being used.

Procedure 6.1.1 Summation of Strata

Apply Form I, the Final ARCL Worksheet (Figure 5.16), for each of the scenarios employed. The total dose per stratum and the variance of the total dose per stratum came from Items 8a and 11 of Form D, the Stratum Dose Calculation Worksheet (Figure 5.8), for each stratum.

Items 7 and 8 of the Final ARCL Worksheet need not be filled out at this time unless the UCL on the dose is desired for each individual scenario.

Form I - Final ARCL Worksheet

1. Facility/Scenario: _____
2. Preparer's Name: _____
3. Date Prepared: _____

[illegible]

7. $(6a)^{1/2} = SD [\hat{\theta}_S(t)] =$ _____
8. $5a + z^{(d)}_{1-\alpha} * 7 =$ _____ (_____ % UCL)

- (a) Note: Forms A, B, C, and D are completed for each Stratum considered.
(b) From Item 8b in Form D, for each Stratum considered.
(c) From Item 11 in Form D, for each Stratum considered.
(d) Obtained from Table 5.5.

FIGURE 5.16. Form I - Final ARCL Worksheet - Summed Over All Strata

FIGURE 5.17. Task Required to Perform Activity 7 - Comparison of Calculated Doses to Standard

limit, decommissioning activities may begin. If the UCL dose is greater than the limit, remedial actions before decommissioning are indicated. Document the result.

6.0 REFERENCES

- Adams, J. A., P. W. Griffin, M. C. Hughes and D. K. Tyler. 1984. Hanford 100 Area Long-Range Decommissioning Plan. UNI-2533, UNC Nuclear Industries, Inc., Richland, Washington.
- Codell, R. B. 1983. "Transport of Radionuclides in Ground Water." In Radiological Assessment, J. E. Till and H. R. Meyer, eds, NUREG/CR-3332, U.S. Nuclear Regulatory Commission, Washington, D.C.
- Dorian, J. J., and V. R. Richards. 1978. Radiological Characterization of the Retired 100 Areas. UNI-946, United Nuclear Industries, Richland, Washington.
- Engel, R. L., J. Greenborg and M. M. Hendrickson. 1966. ISOSHL D - A Computer Code for General Purpose Isotope Shielding Analysis. BNWL-236, Pacific Northwest Laboratory, Richland, Washington.
- Federal Register, Vol. 46, No. 27. 11666-11668 (February 10, 1981). Proposed Rules: Nuclear Regulatory Commission, 10 CFR Parts 30, 40, 50, 70 and 72, "Decommissioning Criteria for Nuclear Facilities; Notice of Availability of Draft Environmental Impact Statement."
- Harmon, K. M., and J. C. King. 1975. Resource Book - Decommissioning of Contaminated Facilities at Hanford. PNL-MA-588, Pacific Northwest Laboratory, Richland, Washington.
- International Commission on Radiological Protection (ICRP). 1977. "ICRP Publication 26, Recommendations of the International Commission on Radiological Protection." Annals of the ICRP, Vol. 1, No.3. Pergamon Press, Oxford.
- International Commission on Radiological Protection (ICRP). 1979a. "ICRP Publication 30, Part 1, Limits for Intakes of Radionuclides by Workers." Annals of the ICRP, Vol. 2, No.3/4. Pergamon Press, Oxford.
- International Commission on Radiological Protection (ICRP). 1979b. "ICRP Publication 30, Supplement to Part 1, Limits for Intakes of Radionuclides by Workers." Annals of the ICRP, Vol. 3, No.1-4. Pergamon Press, Oxford.
- International Commission on Radiological Protection (ICRP). 1980. "ICRP Publication 30, Part 2, Limits for Intakes of Radionuclides by Workers." Annals of the ICRP, Vol. 4, No.3/4. Pergamon Press, Oxford.
- International Commission on Radiological Protection (ICRP). 1981. "ICRP Publication 30, Part 3 Including Addendum to Parts 1 and 2, Limits for Intakes of Radionuclides by Workers." Annals of the ICRP, Vol. 6, No.2/3. Pergamon Press, Oxford.

- International Commission on Radiological Protection (ICRP). 1982a. "ICRP Publication 30, Supplement A to Part 3, Limits for Intakes of Radionuclides by Workers." Annals of the ICRP, Vol. 6, No.2/3. Pergamon Press, Oxford.
- International Commission on Radiological Protection (ICRP). 1982b. "ICRP Publication 30, Supplement B to Part 3 Including Addendum to the Supplements of Parts 1 and 2, Limits for Intakes of Radionuclides by Workers." Annals of the ICRP, Vol. 6, No.2/3. Pergamon Press, Oxford.
- International Commission on Radiological Protection (ICRP). 1986. "ICRP Publication 48, The Metabolism of Plutonium and Related Elements." Annals of the ICRP, Vol. 16, No.2/3. Pergamon Press, Oxford.
- Kennedy, W. E., Jr., and B. A. Napier. 1983a. Allowable Residual Contamination Levels for Decommissioning the 115-F and 117-F Facilities at the Hanford Site. PNL-4704, UNI-2499, Pacific Northwest Laboratory, Richland, Washington.
- Kennedy, W. E., Jr., and B. A. Napier. 1983b. Allowable Residual Contamination Levels for Decommissioning Facilities in the 100 Areas of the Hanford Site. PNL-4722, UNI-2522, Pacific Northwest Laboratory, Richland, Washington.
- Kennedy, W. E., Jr., L. L. Cadwell and D. H. McKenzie. 1985. "Biotic Transport of Radionuclides from a Low-Level Radioactive Waste Site." Health Physics, Vol. 49(1):11-24.
- Kennedy, W. E., Jr., R. A. Peloquin, B. A. Napier and S. M. Neuder. 1986. Intruder Dose Pathway Analysis for the Onsite Disposal of Radioactive Wastes: Supplements to the ONSITE/MAXI1 Computer Program, NUREG/CR-3620, Pacific Northwest Laboratory, Richland, Washington.
- Kirkham, R. R., and G. W. Gee. 1984. "Measurements of Unsaturated Flow Below the Root Zone at an Arid Site." In Proceedings of the National Water Well Association Conference on Characterization and Monitoring of the Vadose Zone. National Water Well Association, Las Vegas, Nevada. (PNL-SA-11629)
- Konczek, G. J., J. D. Ludwick, W. E. Kennedy, Jr., and R. I. Smith. 1981. Technology, Safety and Costs of Decommissioning Reference Nuclear Research and Test Reactors. NUREG/CR-1756, U.S. Nuclear Regulatory Commission, Washington, D.C.
- MacKenzie, D. R., J. F. Smalley, C. R. Kempf, and R. E. Barletta. 1985. Evaluation of the Radioactive Inventory In, and Estimation of Isotopic Release From, the Waste in Eight Trenches at the Sheffield Low-Level Waste Burial Site, NUREG/CR-3865, Brookhaven National Laboratory, Upton, New York.

- McCormack, W. D., J. V. Ramsdell and B. A. Napier. 1984. Hanford Dose Overview Program: Standardized Methods and Data for Hanford Environmental Dose Calculations. PNL-3777, Rev. 1, Pacific Northwest Laboratory, Richland, Washington.
- McKenzie, D. H., L. L. Cadwell, L. E. Eberhardt, W. E. Kennedy, Jr., R. A. Peloquin and M. A. Simmons. 1982a. Relevance of Biotic Pathways to the Long-Term Regulation of Nuclear Waste Disposal: Topical Report on Reference Western Arid Low-Level Sites. Vol. 2, NUREG/CR-2675, U.S. Nuclear Regulatory Commission, Washington, D.C.
- McKenzie, D. H., L. L. Cadwell, L. E. Eberhardt, W. E. Kennedy, Jr., R. A. Peloquin and M. A. Simmons. 1982b. Relevance of Biotic Pathways to the Long-Term Regulation of Nuclear Waste Disposal: Topical Report on Reference Western Arid Low-Level Sites. Vol. 3, NUREG/CR-2675, U.S. Nuclear Regulatory Commission, Washington, D.C.
- Mueller, M. A., W. E. Kennedy, Jr., and J. K. Soldat. 1981. Review of Soil Contamination Guidance. PNL-3866, Pacific Northwest Laboratory, Richland, Washington.
- Napier, B. A., R. L. Roswell, W. E. Kennedy, Jr., and D. L. Streng. 1980. ARRRG and FOOD - Computer Programs for Calculating Radiation Dose to Man from Radionuclides in the Environment. PNL-3180, Pacific Northwest Laboratory, Richland, Washington.
- Napier, B. A. 1982. A Method for Determining "Allowable Residual Contamination Levels" of Radionuclide Mixtures in Soil. PNL-3852, Pacific Northwest Laboratory, Richland, Washington.
- Napier, B. A., R. A. Peloquin, W. E. Kennedy, Jr., and S. M. Neuder. 1984. Intruder Dose Pathway Analysis for the Onsite Disposal of Radioactive Wastes: The ONSITE/MAXI Computer Program. NUREG/CR-3620 (PNL-4054), Pacific Northwest Laboratory, Richland, Washington.
- Napier, B. A., R. A. Peloquin, D. L. Streng and J. V. Ramsdell. 1988. GENII - The Hanford Environmental Radiation Dosimetry Software System. Vols. 1-3, PNL-6584, Pacific Northwest Laboratory, Richland, Washington. In press.
- Oak, H. D., G. M. Holter, W. E. Kennedy, Jr., and G. J. Konzek. 1980. Technology, Safety and Costs of Decommissioning a Reference Boiling-Water Reactor Power Station. Vol. 1 and 2, NUREG/CR-0672, U.S. Nuclear Regulatory Commission, Washington, D.C.
- O'Donnell, F. R., S. J. Cotter, D. C. Kocher, E. L. Etnier and A. P. Watson. 1978. Potential Radiation Dose to Man from Recycle of Metals Reclaimed from a Decommissioned Nuclear Power Plant. NUREG/CR-0134, U.S. Nuclear Regulatory Commission, Washington, D.C.

- Oztunali, O. I., and G. W. Roles. 1986. Update of Part 61 - Impacts Analysis Methodology, NUREG/CR-4370, Envirosphere Company, New York, New York.
- Piepel, G. F. 1986. Sampling and Sample Size Considerations for Inventory Estimation and ARCL Remedial Action Testing, Pacific Northwest Laboratory, Richland, Washington. Letter report from R. G. Schreckhise (PNL) to J. A. Adams (UNC Nuclear Industries), dated October 15, 1986.
- Prater, L. S., J. T. Rieger, C. S. Cline, E. J. Jensen, T. L. Liikala, K. R. Oster and P. A. Eddy. 1984. Ground-Water Surveillance at the Hanford Site for CY 1983. PNL-5041, Pacific Northwest Laboratory, Richland, Washington.
- Reisenauer, A. E. 1979. Variable Thickness Transient Groundwater Flow Model. PNL-3160-1,2,3, Pacific Northwest Laboratory, Richland, Washington.
- Schneider, J. K., and C. E. Jenkins. 1977. Technology, Safety, and Costs of Decommissioning a Reference Nuclear Fuel Reprocessing Plant. NUREG-0278, U.S. Nuclear Regulatory Commission, Washington, D.C.
- Simmons, G. L. 1967. ISOSHLI II Code Revision to Include Calculation of Dose Rates from Shielded Bremsstrahlung Sources. BNWL-236, Supplement 1, Pacific Northwest Laboratory, Richland, Washington.
- Smith, R. I., G. J. Konzek, and W. E. Kennedy, Jr. 1978. Technology, Safety, and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station. NUREG/CR-0130, U.S. Nuclear Regulatory Commission, Washington, D.C.
- Staley, G. B., G. P. Turi and D. L. Schreiber. 1979. "Radionuclide Migration from Low-Level Waste: A Generic Overview." In Management of Low-Level Radioactive Waste, Vol. 2, eds. M. W. Carter, A. A. Moghissi and B. Kotn.
- U.S. Atomic Energy Commission (AEC). 1974. Termination of Operating Licenses for Nuclear Reactors. Regulatory Guide 1.86, Directorate of Regulatory Standards, Washington, D.C.
- U.S. Code of Federal Regulations (CFR). Title 40, Part 190, "Environmental Protection Standards for Nuclear Power Operations."
- U.S. Code of Federal Regulations (CFR). Title 40, 192, "Environmental Protection Standards for Uranium Mill Tailings."
- U.S. Nuclear Regulatory Commission (NRC). 1976. Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use of Termination of Licenses for By-Product, Source, or Special Nuclear Material. RS2-1131, U.S. Nuclear Regulatory Commission, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1980. Draft Environmental Impact Statement Concerning Proposed Rulemaking Exemption from Licensing Requirements for Smelted Alloys Containing Residual Technetium-99 and Low-Enriched Uranium. NUREG-0512, U.S. Nuclear Regulatory Commission, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1981. Draft Environmental Impact Statement on 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste." Vol. 1-4, NUREG-0782, U.S. Nuclear Regulatory Commission, Washington, D.C.

Washington State University. 1970. Alfalfa Irrigation. E.M 3468, Cooperative Extension Service, Pullman, Washington.

APPENDIX A

EXAMPLE CALCULATION

APPENDIX A

EXAMPLE CALCULATION

As an example application of the Allowable Residual Contamination Level (ARCL) technique described in this report, the complete calculation of a 90th percentile upper confidence limit dose for an actual facility is presented in Table A.5 and A.6. The 115-D/DR Gas Recirculation Building, located in the 100-D Area of the Hanford Site, was used as the example facility. The 115-D/DR Building was connected to the D and DR reactors by tunnels and was adjacent to the 117-D Filter Building. The inventory of radiocontaminants present in the 117-D Building were added to the inventory values used for the 115-D/DR Building which complicated the ARCL calculations. A portion of the contamination in the 115-D/DR Building was located inside of large-diameter pipes, which necessitated the need to examine the Recycle scenario. This example calculation includes the following scenarios:

- Residential/Home Garden--for the 115-D/DR Building itself;
- Agricultural--for the radionuclides in the 115-D/DR Building not included in the Residential/Home Garden scenario above along with the radiocontaminants in the 117-D Building; and
- Resource Recycle--for the radionuclides in the pipes.

The available sampling data are summarized in Table A.1 for the 115-D/DR Building, excluding the data for the associated piping, which are presented in Tables A.2 and A.3. Table A.4 summarizes the data for the 117-D Building. Evaluation of the data suggested that the 115-D/DR Building and its surroundings should be separated into the following strata:

1. main floor and basement of the 115-D/DR Building, excluding a portion of Dry Room #1;
2. the excluded portion of Dryer Room #1 in the 115-D/DR Building;
3. 115-D Tunnel;

4. 115-DR Tunnel;
5. 117-D Building's Inlet Duct, A Filter Cell, and A Outlet Seal Pit;
6. 117-D Building's Outlet Duct, A Inlet Seal Pit, B Inlet Seal Pit, B Filter Cell, and B Outlet Seal Pit; and
7. Pipes in the 115-D/DR Building.

Summary statistics for each of these strata were prepared using the SAS(a) standard computer package and are presented in copies of Form A, the Summary Statistics Table (Figure 5.3), for each stratum. For each stratum, the measured "gross α " values were treated as if they represented ^{239}Pu .

Calculations for the Agricultural scenario were performed for each of the strata with the exception of the pipes. For the area of highest dose potential (i.e., a section of 115-D/DR Dryer Room #1 and surrounding material) the calculations for the Residential/Home Garden scenario are shown. For the pipes, the Resource Recycle scenario calculations are presented.

Results for each stratum were summed for the Agricultural scenario on Form D. The results were also summed for the two strata involved in the Residential/Home Garden scenario. These two values were then added together and compared to the Resource Recycle scenario. It was apparent that the Residential/Home Garden scenario dominated the final answer, and that it in turn was largely controlled by the small area of Dryer Room #1, which sampling indicated had a relatively high concentration of ^{90}Sr .

Some of the assumptions made and observations developed during the calculations are listed in the following:

- The strata were formed basically considering ^{90}Sr concentrations since it was the dominant radionuclide. In some cases, further stratification seemed appropriate; however, it was not implemented because the amount of data that would be assigned to the additional strata would not be sufficient.

(a) SAS Institute Incorporated, Cary, North Carolina.

- Post-stratification (i.e., stratifying the data after the samples are collected instead of defining the strata before the sampling is conducted) can be dangerous when there are insufficient data to support it. The data were post-stratified in the example since the samples were collected prior to the development of the statistical techniques outlined here. Besides, post-stratification seemed reasonable to do since there was one dominant radionuclide, and most of the projected dose came from a 37-sq-ft hot spot in Dryer Room #1 of the 115-D/DR Building. Pre-sampling efforts should be conducted to define the appropriate strata to be sampled in more detail.
- As indicated by footnotes in the worksheets, the data were often insufficient for computing many of the variances and covariances. Even where they could be computed, many times it was with fewer than five observations. The general solution is to collect additional samples and make sufficient analyses for each radionuclides so as to have at least five values per radionuclide per stratum. Beyond this, input from sample size tables and formulas should be considered.
- Care must be taken in estimating values for missing data or statistical summaries. This is especially true for situations where the resulting calculations are near the prescribed limit.

TABLES A.1 - A.7

SAMPLING DATA FOR 115 D/DR and 117-D BUILDINGS

TABLE A.1. Sampling Data for the 115-D/DR Gas Recirculation Building (pCi/g)

Observation Number	Room	Gross Beta	Gross Alpha ^(a)	⁶⁰ Co	¹³⁷ Cs	¹⁵² Eu	¹⁵⁴ Eu	⁹⁰ Sr	³ H	¹⁴ C
1	FR2	86	2.00	.	41.7	.	.	.	.	.
2	FR2	367	9.00	3.4	250.0	16.5	15.1	6.5	.	.
3	FR1	49	0.00	.	0.0	.	.	.	.	.
4	FR1	283	0.00	.	0.0	.	.	.	.	.
5	PB1	13	0.50	.	0.0	.	.	.	.	.
6	PB2	14	1.00	.	0.0	.	.	.	.	.
7	CB1	18	0.50	.	0.0	.	.	.	.	.
8	CB1	7850	6.20	1.4	272.0	6.6	5.5	44.0	.	.
9	CB2	23	0.50	.	0.0	.	.	.	.	.
10	CB2	24795	(b)	.	4.9	.	.	.	.	.
11	CB2	16792	(b)	.	5.2	.	.	.	.	.
12	CB2	106	2.00	11.5	7.2	10.7	.	.	.	.
13	CB2	217	2.00	36.6	28.2	40.6	.	.	.	.
14	DR1	179	1.00	.	0.0	14.0	16.1	4.4	5.2	12.7
15	DR1	202	4.00	4.0	8.1	.	.	46.9	.	.
16	DR1	12567	3.20	51.5	1970.0	13.6	20.9	3480.0	.	.
17	DR1	1089	14.00	.	0.0	11.5	16.7	6.3	55.5	6.2
18	DR1	143	1.00	.	16.9	9.4	3.8	2.2	.	.
19	DR2	13	0.50	.	0.0	.	.	.	.	.
20	DR2	22	0.50	.	0.0	.	.	.	.	.
21	DR2	3938	50.00	.	8.4	.	.	.	.	.
22	DR2	8598	0.21	.	11.1	27.4	16.0	13.6	.	.
23	DR2	155	5.00	.	16.2	.	.	.	.	.
24	DR2	24	0.50	.	0.0	.	.	.	.	.
25	DR3	17	0.50	.	0.0	.	.	.	.	.
26	DR3	41	0.00	.	0.0	.	.	.	.	.
27	DR3	20	0.50	.	10.2	.	.	.	.	.
28	DR3	227	4.00	.	7.1	.	.	.	.	.
29	DR3	19	0.50	.	0.0	.	.	.	.	.
30	DR4	19	1.00	.	0.0	.	.	.	.	.
31	DR4	534	7.00	.	0.0	.	.	.	.	.
32	DR4	264	2.00	.	0.0	.	.	.	.	.
33	DR4	130	0.50	.	0.0	.	.	.	.	.
34	DR4	19	0.00	.	5.0	.	.	.	.	.
35	DR4	13	0.50	.	0.0	.	.	.	.	.
36	DR5	16	0.00	.	0.0	.	.	.	.	.
37	DR5	662	8.00	.	0.0	.	.	.	.	.
38	DR5	1840	25.00	.	7.7	.	.	.	.	.

(a) The gross alpha values were treated as ²³⁹Pu values.

(b) Due to the type of method used, some gross alpha measurements were affected by the presence of beta/gamma-emitting radionuclides. Most abnormally high alpha measurements were resolved by additional measurements using another method. However, this was not done for observations 10 and 11 (which had gross alpha values of 240 and 166, respectively). Therefore, following UNC's recommendations, observations 10 and 11 were assumed to be outliers and were treated as missing observations.

TABLE A.1 (contd)

Observation Number	Room	Gross Beta	Gross Alpha ^(a)	⁶⁰ Co	¹³⁷ Cs	¹⁵² Eu	¹⁵⁴ Eu	⁹⁰ Sr	³ H	¹⁴ C
39	DR5	1740	7.00	.	12.5	.	.	5.9	.	.
40	DR5	486	7.00	.	245.0	.	.	.	.	.
41	DR5	18	0.50	.	0.0	.	.	.	.	.
42	BMT	500	7.00	.	80.6	.	.	.	.	.
43	BMT	19278	0.73	13.2	603.0	.	.	87.1	.	.
44	BMT	4865	65.00	.	28.1	.	.	.	.	.
45	BMT	129	2.00	.	144.0	.	.	.	.	.
46	BMT	77	2.00	.	36.6	.	.	.	.	.
47	BMT	1084	14.00	.	11.7	.	.	.	.	.
48	BMT	65	1.00	.	47.6	.	.	.	.	.
49	BMT	141	4.00	2.8	142.0	.	.	4.7	.	.
50	BMT	183	3.00	.	25.4	.	.	.	.	.
51	BMT	36	3.00	.	10.9	.	.	.	.	.
52	BMT	103	2.00	.	102.0	.	.	.	.	.
53	BMT	33	1.00	.	28.6	.	.	.	.	.
54	BMT	61	0.50	.	65.9	.	.	.	.	.
55	BMT	2003	21.0	.	23.7	.	.	.	.	.
56	BMT	35	1.0	.	29.8	.	.	.	.	.
57	BMT	83	0.5	.	37.2	.	.	.	.	.
58	BMT	66	1.0	.	66.0	.	.	.	.	.
59	BMT	45	1.0	.	49.8	.	.	.	.	.
60	BMT	22	0.5	.	15.4	.	.	.	.	.
61	BMT	30	0.5	.	26.1	.	.	.	.	.
62	BMT	53	1.0	.	39.6	.	.	.	.	.
63	BMT	52	1.0	.	58.5	.	.	.	.	.
64	BMT	80	2.0	.	101.0	.	.	.	.	.
65	BMT	180	3.0	.	220.0	13.4	7.8	9.5	.	.
66	BMT	90	2.0	.	139.0	.	.	.	.	.
67	BMT	62	1.0	.	88.5	.	.	.	.	.
68	BMT	168	3.0	.	167.0	.	.	.	.	.
69	TD	505	5.0	76.3	547.0	171.0	16.7	25.8	7.1	2.7
70	TD	222	4.0	.	94.4	.	.	19.8	.	.
71	TD	316	4.0	31.5	135.0	71.4	.	.	.	.
72	TD	432	4.0	15.5	73.0	38.5	.	.	.	.
73	TD	499	7.0	164.0	192.0	385.0	41.7	.	.	.
74	TD	410	7.0	119.0	190.0	256.0	31.0	.	.	.
75	TD	568	8.0	161.0	221.0	369.0	41.0	.	.	.
76	TD	879	10.0	191.0	1030.0	423.0	40.8	.	.	.
77	TD	14000	27.2	531.0	6020.0	734.0	0.0	253.0	.	.
78	TDR	261	5.0	.	448.0	12.6	7.4	6.4	2.7	25.1
79	TDR	268	6.0	.	438.0	19.1	7.8	6.5	.	.
80	TDR	194	3.0	2.7	317.0	.	.	7.1	.	.
81	TDR	197	4.0	.	211.0	45.4	24.8	.	.	.
82	TDR	171	4.0	.	208.0	64.5	.	.	.	.

(a) The gross alpha values were treated as ²³⁹Pu values.

TABLE A.2. Sampling Data for the 117-D Filter Building (pCi/g)

Observation Number	Room	²³⁹ Pu	³ H	¹⁴ C	⁶⁰ Co	⁹⁰ Sr	¹³⁷ Cs	¹⁵² Eu
1	ID	0	118	216	-	217	682	31
2	ID	9	-	-	-	-	241	-
3	ID	10	-	-	-	-	329	-
4	ID	7	-	-	-	-	214	-
5	ID	7	-	-	-	-	209	-
6	ID	10	-	-	-	-	330	-
7	ID	7	-	-	-	-	156	-
8	ID	6	-	-	-	-	39	-
9	ID	7	-	-	-	-	109	-
10	ID	5	-	-	-	-	384	-
11	OD	38	-	-	-	-	-	-
12	OD	32	-	-	-	-	-	-
13	OD	39	-	-	-	-	29	-
14	OD	36	-	-	-	-	-	-
15	OD	19	-	-	-	-	-	-
16	OD	49	-	-	-	-	34	-
17	OD	47	-	-	-	-	-	-
18	OD	15	-	-	-	-	-	-
19	OD	38	23	25	-	6	36	-
20	OD	23	-	-	-	-	34	-
21	AISP	3	-	-	-	-	101	-
22	AISP	5	17	29	17	16	174	-
23	BSIP	5	-	-	-	8	409	-
24	BSIP	8	9	24	-	2	100	-
25	AFC	12	-	-	44	396	761	-
26	AFC	18	-	-	175	739	1410	-
27	AFC	8	683	573	180	561	1070	64
28	AOSP	3	8	20	32	136	23	-
29	AOSP	4	-	-	-	-	-	-
30	BFC	10	-	-	-	-	442	-
31	BFC	12	-	-	-	-	381	-
32	BFC	15	96	132	-	3	1750	-
33	BFC	35	-	-	454	55	3230	-
34	BOSP	1	-	-	-	-	-	-
35	BOSP	3	9	25	-	10	17	-

TABLE A.3. Sampling Data for the Pipes in the 115-D/DR Building (pCi/15.5 cm²)

Observation Number	Room	Sample Number	³ H	¹⁴ C	⁹⁰ Sr	⁶⁰ Co	¹³⁷ Cs	¹⁵² Eu	¹⁵⁴ Eu	Alpha
1	D	1	.	.	390	.	51500	.	.	.
2	D	2	94.7	36.6	.	7218.0	8060	.	.	651
3	D	3	.	.	.	7846.0	19500	684	549	691
4	D	4	.	.	.	1809.0	17700	682	399	.
5	D	5	.	.	.	72.2	2210	495	201	116
6	D	6	.	.	.	71.8	33100	346	202	119
7	D	7	.	.	155.0	.	56000	498	203	.
8	D	8	.	.	54.8	.	57200	344	1287	.
9	D	9	.	.	101.0	.	69000	345	201	.
10	D	10	.	.	46.8	.	48000	1116	401	.
11	DR	1	.	.	197.0	.	35700	504	202	.
12	DR	2	115.0	4.5	.	.	82000	706	818	123
13	DR	3	.	.	170.0	72.2	89100	345	202	.
14	DR	4	300.0	6.4	167.0	.	69900	344	200	120
15	DR	5	.	.	306.0	.	96500	344	399	.
16	DR	6	.	.	286.0	72.0	60500	345	201	.
17	DR	7	.	.	.	72.1	22800	346	203	112
18	DR	8	.	.	.	71.8	22100	345	201	122
19	DR	9	13300.0	585.0	.	71.9	15700	345	201	115
20	DR	10	6515.0	315.0	.	898.0	20700	984	401	150

TABLE A.4. Sampling Data for the Pipes in the 115-D/DR Building Converted to pCi/g (values in Table A.3 divided by 155 g/cm²)

Observation Number	Room	Sample Number	³ H	¹⁴ C	⁹⁰ Sr	⁶⁰ Co	¹³⁷ Cs	¹⁵² Eu	¹⁵⁴ Eu	Alpha
1	D	1	.	.	2.51	.	332.	.	.	.
2	D	2	0.611	0.236	.	46.	52.0	.	.	4.20
3	D	3	.	.	.	50.6	126.	4.41	3.54	4.46
4	D	4	.	.	.	11.7	114.	4.40	2.57	.
5	D	5	.	.	.	0.466	14.3	3.19	1.30	0.748
6	D	6	.	.	.	0.463	214.	2.23	1.30	0.768
7	D	7	.	.	1.00	.	361.	3.21	1.31	.
8	D	8	.	.	0.354	.	369	2.22	8.30	.
9	D	9	.	.	0.652	.	445.	2.23	1.30	.
10	D	10	.	.	0.302	.	310.	7.20	2.59	.
11	DR	1	.	.	1.27	.	553.	3.25	1.30	.
12	DR	2	0.742	0.0290	.	.	529.	4.55	5.28	0.794
13	DR	3	.	.	1.10	0.466	575.	2.23	1.30	.
14	DR	4	1.94	0.0413	1.08	.	451.	2.22	1.29	0.774
15	DR	5	.	.	1.97	.	623.	2.22	2.57	.
16	DR	6	.	.	1.85	0.465	390.	2.23	1.30	.
17	DR	7	.	.	.	0.465	147.	2.23	1.31	0.723
18	DR	8	.	.	.	0.463	143.	2.23	1.30	0.787
19	DR	9	85.8	3.77	.	0.464	101.	2.23	1.30	0.742
20	DR	10	42.0	2.03	.	5.794	134.	6.35	2.59	0.968

SAMPLE ARCL WORKSHEETS FOR 115D/DR and 117-D BUILDINGS

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2. Preparer's Name: G. F. Piepel

3. Date Prepared: 8/25/88

4. Radionuclide Considered	5. Mean $\bar{x}_{hi}$ (pCi/g)	6. S. D. σ_{hi} (pCi/g)	7. n_i	8. Nuclide i(a)	9. Nuclide j(a)	10. Covariance (σ_{hij})(a)	11. Number of Samples for which both Nuclides are Analyzed (n_{hij})
Co-60	10.41	12.41	7	Co-60	Cs-137	-485.43	7
Cs-137	74.34	107.95	46	Co-60	Eu-152	230.14	4
Eu-152	18.68	18.74	9	Co-60	Eu-154	89.4	2(b)
Eu-154	11.57	5.63	7	Co-60	Sr-90	123.94	5
Sr-90	21.01	27.03	11	Co-60	H-3	3099.0	0(b)
H-3	30.35	35.57	2	Co-60	C-14	-673.7	0(b)
C-14	9.45	4.60	2	Ca-137	Eu-152	-630.49	7
α /Pu-239	4.74	10.85	65	Ca-137	Eu-154	-48.50	5
				Ca-137	Sr-90	3988.54	9
				Ca-137	H-3	19,938.9	0
				Ca-137	C-14	-3,901.1	0
				Eu-152	Eu-154	23.05	7
				Eu-152	Sr-90	-26.09	7
				Eu-152	H-3	-575.4	2(b)
				Eu-152	C-14	-471.4	2(b)
				Eu-154	Sr-90	-30.78	7
				Eu-154	H-3	48.8	2(b)
				Eu-154	C-14	-41.6	2(b)
				Sr-90	H-3	-61.0	2(b)
				Sr-90	C-14	-58.5	2(b)
				H-3	C-14	-108.7	2(b)
				α /Pu-239	Co-60	-19.76	7
				α /Pu-239	Cs-137	-174.03	43
				α /Pu-239	Eu-152	-14.78	9
				α /Pu-239	Eu-154	8.71	7
				α /Pu-239	Sr-90	28.47	11
				α /Pu-239	H-3	130.19	2(b)
				α /Pu-239	C-14	-17.79	2(b)

(a) Note: Columns 8-11 will be much longer than 4-7.

Comments: Insufficient data to estimate covariance. Values given were computed by taking into account other aspects of the data. This was done to complete this example and is not recommended as general practice. For $n_{hij} \leq 2$, use $n_{hij}=1$ in any computations requiring the values.

Form B - Source-Term Worksheet

1. Facility/Scenario and Stratum: 115-D/DR - Residential-Home Garden Scenario - Main Floor and Basement (except small portion of Dryer Room #1)
2. Preparer's Name: G. F. Piepel
3. Date Prepared: 8/25/88
4. Stratum Mass^(a) (grams): 5.25E+6 (a)
5. Number of Years of Decay: 100 (t)

8. Radionuclide Considered	7. Mean Concentration in Stratum, $\bar{x}_{hi}$ (pCi/g) (b)	8. Stratum Concentration S. D., σ_{hi} (pCi/g)	9. Number of Samples in Stratum n_{hi} (b)	10. Source Term y_{hi} (d)	11. S. D. y_{hi} (d)	12. Decayed Source Term (pCi) (e)	13. Decayed S. D. y_{hi} (f)
Co-60	10.41	12.41	7	6.47E+7	2.48E+7	1.24E+2	5.56E+1
Cs-137	74.34	107.95	45	3.90E+8	8.46E+7	3.90E+7	8.46E+6
Eu-152	16.68	10.74	9	8.78E+7	1.88E+7	0	0
Eu-154	11.67	6.63	7	8.07E+7	1.12E+7	1.24E+4	2.28E+3
Sr-90	21.01	27.03	11	1.10E+8	4.28E+7	9.98E+6	3.88E+6
H-3	30.35	35.67	2	1.59E+8	1.32E+8	5.88E+5	4.88E+5
C-14	9.45	4.80	2	4.98E+8	1.71E+7	4.90E+7	1.69E+7
α /Pu-239	4.74	10.86	85	2.49E+8	8.94E+8	2.48E+7	8.92E+8

(a) Strata are defined to consist only of contaminated material.

(b) Copy from Summary Statistics Table (Table 5.2.1).

(c) Calculate as Item 4 times Item 7.

$$(d) \text{ S. D. } (y_{hi}) = \left(\frac{n_{hi}^2 \sigma_{hi}^2}{n_{hi}} \right)^{1/2}$$

(e) Calculate as Item 10 times $e^{-\lambda_i t}$ where λ_i = decay constant for radionuclide i (from Table 5.2.3) and t = number of years of decay (Item 5).

(f) Calculate as Item 11 times $e^{-\lambda_i t}$ where λ_i = decay constant for radionuclide i (from Table 5.2.3) and t = number of years of decay (Item 5).

Form C - Availability Correction Worksheet

1. Facility/Scenario and Stratum: 115 D/DR Residential-Home Garden Scenario - Main Floor and Basement (except small portion of Dry Room #1)
2. Preparer's Name: G. F. Piepel
3. Date Prepared: 6/28/88

4. Radionuclide Considered	5. Reduction Factor(a)	6. Unmodified Dose Factors(b)		7. Corrected Dose Factors(c)		8. Residential/HomeGarden(a) Scenario Dose Factors (rem/yr per pCi)
		6a. Inhalation (rem/yr per pCi)	6b. Ingestion (rem/yr per pCi)	7a. Inhalation (rem/yr per pCi)	7b. Ingestion (rem/yr per pCi)	
Co-60	1.0	4.3E-12	1.0E-14	4.3E-12	1.0E-14	4.3E-12
Cs-137	1.0	1.0E-12	1.0E-14	1.0E-12	1.0E-14	1.0E-12
Eu-152	1.0	2.0E-12	1.7E-15	2.0E-12	1.7E-15	2.0E-12
Eu-154	1.0	2.1E-12	2.6E-15	2.1E-12	2.6E-15	2.1E-12
Sr-90	1.0	1.1E-14	2.7E-12	1.1E-14	2.7E-12	2.7E-12
H-3	1.0	2.4E-20	3.1E-14	2.4E-20	3.1E-14	3.1E-14
C-14	1.0	4.1E-18	3.1E-14	4.1E-18	3.1E-14	3.1E-14
α /Pu-239	1.0	8.7E-14	1.1E-14	8.7E-14	1.1E-14	9.8E-14

(a) RF = 1.0 for radionuclides not contained in activated metals.

RF = $2t$ (corrosion rate cm/yr)/(thickness in cm).

(b) From Table 2.2.

(c) Calculated as the product of columns 5 and 6, by radionuclide.

(d) Complete this column only for the Residential/Home Garden scenario. The scenario dose factors are calculated as Column 7a plus Column 7b.

Form E--Dose Variance [Var(D_h)] Calculation Worksheet for Stratum h

1. Facility and Stratum: 115D/DR - Main floor & basement (except small portion of Dryer Room #1)
Residential Home Garden Scenario
2. Preparer's Name: GF Piepel 3. Date Prepared: 6/25/86 4. Stratum Mass (m_h): 5.25E+6 g 5. Decay Time (t): 100 yr

					A _{hjk} #								
6.*		Radionuclide h _k :			2	3	4	5	6	7	8	10.*	
Radio	7.*	8.*	9.*	k:	137Cs	152Eu	154Eu	90Sr	3H	14C	239Pu	dh _j Edh _k A _{hjk}	
nucli	SD(dh _j y _{h_j})	λ _{h_j}	λ _{h_k}	dh _j \dh _k :	2.3E-2	5.2E-2	8.5E-2	2.4E-2	5.6E-2	1.2E-4	2.9E-5		
de h _j		λ _{h_j}	dh _j \dh _k :		1.0E-12	2.0E-12	2.1E-12	2.7E-12	3.1E-14	3.1E-14	9.8E-14		
1	60Co	2.4E-10	1.3E-1	4.3E-12	$\frac{-485.43}{7}e^{-15.3}$	$\frac{230.14}{4}e^{-18.2}$	$\frac{69.4}{2}e^{-21.5}$	$\frac{123.94}{5}e^{-15.4}$	$\frac{3099.0}{1}e^{-18.6}$	$\frac{-673.7}{1}e^{-13.0}$	$\frac{-19.75}{7}e^{-13.0}$	-2.02E-28	
2	137Cs	8.5E-6	2.3E-2	1.0E-12		$\frac{-630.49}{7}e^{-7.50}$	$\frac{-48.50}{5}e^{-10.8}$	$\frac{3988.54}{9}e^{-4.70}$	$\frac{19.939}{1}e^{-7.90}$	$\frac{-3901.1}{1}e^{-2.31}$	$\frac{-174.03}{43}e^{-2.30}$	-1.01E-24	
3	152Eu	0	5.2E-2	2.0E-12			$\frac{23.05}{7}e^{-13.7}$	$\frac{-28.09}{7}e^{-7.60}$	$\frac{-575.4}{2}e^{-10.8}$	$\frac{-471.4}{2}e^{-5.21}$	$\frac{-14.78}{9}e^{-5.20}$	-9.26E-26	
4	154Eu	4.8E-9	8.5E-2	2.1E-12				$\frac{-90.78}{7}e^{-10.9}$	$\frac{48.8}{2}e^{-14.1}$	$\frac{-41.8}{2}e^{-8.51}$	$\frac{8.71}{7}e^{-8.50}$	-6.81E-28	
5	90Sr	1.0E-5	2.4E-2	2.7E-12					$\frac{-61.0}{2}e^{-8.00}$	$\frac{-58.5}{2}e^{-2.41}$	$\frac{28.47}{11}e^{-2.40}$	-1.58E-25	
6	3H	1.5E-8	5.6E-2	3.1E-14						$\frac{-108.7}{2}e^{-5.61}$	$\frac{130.19}{2}e^{-5.60}$	5.98E-28	
7	14C	5.2E-7	1.2E-4	3.1E-14							$\frac{-17.79}{2}e^{-0.015}$	-2.66E-26	
8	239Pu	6.8E-7										-1.29E-29	

$$1.73E-10 = \Sigma[SD(dh_{jy_{h_j}})]^2$$

$$2m_h^2 = 5.51E+13$$

$$\Sigma dh_j \Sigma dh_k A_{hjk} = -1.29E-24$$

$$11. \text{Var}(D_h) = \{\Sigma[SD(dh_{jy_{h_j}})]^2\} + \{2m_h^2\}\{\Sigma dh_j \Sigma dh_k A_{hjk}\} = 1.73E-10 + 5.51E+13 \cdot -1.29E-24 = 1.02E-10 = \text{Var}(D_h)$$

- * Column 6 [Radionuclide j and k]--from Form A, Col. 4.
Column 7 [SD(dh_jy_{h_j})]--from Form D, Col. 9.
Column 8 [λ_{h_j} and λ_{h_k}]--from Table 5.3.
Column 9 [dh_j and dh_k]--from Form D, Col. 7.

- # A_{hjk} = $\frac{\hat{\sigma}_{hjk}}{n_{hjk}}e^{-\theta}$, where $\hat{\sigma}_{hjk}$ is from Form B, Col. 10,
n_{hjk} is from Form B, Col. 11, and
 $\theta = (\lambda_{h_j} + \lambda_{h_k})t$.
■ Column 10--calculated as dh_jEdh_kA_{hjk}.

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- [illegible]

11. Variance of Estimated Dose (rem/yr)^{2(f)}:

- (a) From Form B - Source Term Worksheet (column 10 or 12), for $T = \frac{100}{\text{years}}$.
- (b) From Form C - Availability, Form F - Area, Form G - Depth, or Form H - Mass Correction Worksheets, as appropriate.
- (c) Calculate as the product of Columns 5 and 7.
- (d) Calculate as the product of Columns 8 and 7.
- (e) Calculate as $(\text{Column 8}) \times z_{1-\alpha}$ (Column 9), where $z_{1-\alpha}$ is chosen to correspond with the desired confidence level from Table 5.5.
- (f) Calculate using Procedure 3.2.2, Columns 7, 9, and Columns 8, 9, 10, and 11 from Form A - Summary Statistics Worksheet on Form E (Dose Variance Calculation Worksheet).

Form B - Source-Term Worksheet

1. Facility/Scenario and Stratum: 115-D/DR - Agricultural Scenario - Main Floor and Basement (except small portion of Dryer Room #1)
2. Preparer's Name: G. F. Piepel
3. Date Prepared: 8/25/88
4. Stratum Mass^(a) (grams): 1.86E+7 (m)
5. Number of Years of Decay: 100 (t)

6. Radionuclide Considered	7. Mean Concentration in Stratum, $\bar{x}_{hi}$ (pCi/g) (b)	8. Stratum Concentration S. D., (b) σ_{hi} (pCi/g)	9. Number of Samples in Stratum n_{hi} (b)	10. Source Term y_{hi} (d)	11. S. D. y_{hi} (d)	12. Decayed Source Term (pCi) (e)	13. Decayed S. D. y_{hi} (f)
Co-60	10.41	12.41	7	1.73E+8	7.79E+7	3.91E+2	1.78E+2
Cs-137	74.34	107.95	45	1.23E+9	2.67E+8	1.23E+8	2.67E+7
Eu-152	16.88	10.74	9	2.77E+8	5.94E+7	=0	=0
Eu-154	11.57	5.83	7	1.02E+8	3.53E+7	3.90E+4	7.17E+3
Sr-90	21.01	27.03	11	3.40E+8	1.35E+8	3.17E+7	1.22E+7
H-3	30.35	35.57	2	5.09E+8	4.10E+8	1.88E+8	1.65E+8
C-14	9.45	4.80	2	1.67E+8	5.40E+7	1.55E+8	5.34E+7
α/Pu-239	4.74	10.85	65	7.87E+7	2.19E+7	7.85E+7	2.19E+7

(a) Strata are defined to consist only of contaminated material.

(b) Copy from Summary Statistics Table (Table 5.2.1).

(c) Calculate as Item 4 times Item 7.

$$(d) \text{ S. D. } (y_{hi}) = \left(\frac{\sum_{h=1}^H \sum_{i=1}^I \bar{x}_{hi}^2}{n_{hi}} \right)^{1/2}$$

(e) Calculate as Item 10 times $e^{-\lambda_i t}$ where λ_i = decay constant for radionuclide i (from Table 5.2.3) and t = number of years of decay (Item 5).

(f) Calculate as Item 11 times $e^{-\lambda_i t}$ where λ_i = decay constant for radionuclide i (from Table 5.2.3) and t = number of years of decay (Item 5).

Form C - Availability Correction Worksheet

1. Facility/Scenario and Stratum: 116 D/DR Agricultural Scenario - Main Floor and Basement (except small portion of Dry Room #1)

2. Preparer's Name: G. F. Piepel

3. Date Prepared: 8/28/88

4. Radionuclide Considered	5. Reduction Factor(a)	6. Unmodified Dose Factors(b)		7. Corrected Dose Factors(c)		8. Residential/HomeGarden(d) Scenario Dose Factors (rem/yr per pCi)
		6a. Inhalation (rem/yr per pCi)	6b. Ingestion (rem/yr per pCi)	7a. Inhalation (rem/yr per pCi)	7b. Ingestion (rem/yr per pCi)	
Co-60	1.0	Not	8.7E-15	Not	8.7E-15	Not Applicable
Cs-137	1.0	Applicable	8.7E-15	Applicable	8.7E-15	
Eu-152	1.0	for	1.2E-15	for	1.2E-15	
Eu-154	1.0	Agricultural	1.8E-15	Agricultural	1.8E-15	
Sr-90	1.0	Scenario	1.8E-12	Scenario	1.8E-12	
H-3	1.0		2.1E-14		2.1E-14	
C-14	1.0		2.1E-14		2.1E-14	
α /Pu-239	1.0		7.1E-15		7.1E-15	

(a) RF = 1.0 for radionuclides not contained in activated metals.

RF = $2t$ (corrosion rate cm/yr)/(thickness in cm).

(b) From Table 2.2.

(c) Calculated as the product of columns 5 and 6, by radionuclide.

(d) Complete this column only for the Residential/Home Garden scenario. The scenario dose factors are calculated as Column 7a plus Column 7b.

Form G. Depth Correction Worksheet

1. Facility: 115 D/DR Agricultural Scenario - Main Floor and Basement
(except for small portion of Dryer Room #1)

2. Preparer's Name: G. F. Piepel

3. Date Prepared: 6/26/86

4. Depth of Clean overburden 3.35 m.

5. Calculate depth correction factor (DCF) as

$$DCF = e^{-0.92(\text{Depth})} = \underline{0.046}.$$

If Depth > 10 meters DCF = 0.0.

6. Radionuclide	7. Ingestion DF (rem/pCi)(a)	8. Corrected Ingestion DF (rem/pCi)(b)
<u>Co-60</u>	<u>6.7E-15</u>	<u>3.1E-16</u>
<u>Cs-137</u>	<u>6.7E-15</u>	<u>3.1E-16</u>
<u>Eu-152</u>	<u>1.2E-15</u>	<u>5.5E-17</u>
<u>Eu-154</u>	<u>1.6E-15</u>	<u>7.4E-17</u>
<u>Sr-90</u>	<u>1.8E-12</u>	<u>8.3E-14</u>
<u>H-3</u>	<u>2.1E-14</u>	<u>9.7E-16</u>
<u>C-14</u>	<u>2.1E-14</u>	<u>9.7E-16</u>
<u>Pu-239</u>	<u>7.1E-15</u>	<u>3.3E-16</u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>

(a) From Area Correction or Availability Worksheet.

(b) Calculated as Item 5 times Column 7.

Form E--Dose Variance $[\text{Var}(\hat{D}_h)]$ Calculation Worksheet for Stratum h

1. Facility and Stratum: 115D/DR - Main floor & basement (except for small portion of Dryer Room #1)
Agricultural Scenario
2. Preparer's Name: GF Piepel 3. Date Prepared: 6/25/86 4. Stratum Mass (m_h): 1.66E+7 g 5. Decay Time (t): 100 yr

6.*					Ahjk#								10.*		
Radio nucli de h _j		Radionuclide h _k :			2	3	4	5	6	7	8				
7.*		8.*	9.*												
SD(dh _j y _{h_j})		λ _{h_j}	λ _{h_k}										dh _j Idh _k A _{h_jk}		
				137Cs	152Eu	154Eu	90Sr	3H	14C	239Pu					
				2.3E-2	5.2E-2	8.5E-2	2.4E-2	5.6E-2	1.2E-4	2.9E-5					
				3.1E-16	5.5E-17	7.4E-17	8.3E-14	9.7E-16	9.7E-16	3.3E-16					
1	60Co	5.5E-14	1.3E-1	3.1E-16	$\frac{-485.43}{7}e^{-15.3}$	$\frac{230.14}{4}e^{-18.2}$	$\frac{-69.4}{2}e^{-21.5}$	$\frac{123.92}{5}e^{-15.4}$	$\frac{3099.0}{1}e^{-18.6}$	$\frac{-673.7}{1}e^{-13.0}$	$\frac{-19.76}{7}e^{-13.0}$	-3.16E-34			
2	137Cs	8.3E-9	2.3E-2	3.1E-16		$\frac{-630.49}{7}e^{-7.50}$	$\frac{-48.50}{5}e^{-10.8}$	$\frac{3988.54}{9}e^{-4.70}$	$\frac{19.938.9}{1}e^{-7.90}$	$\frac{-3901.1}{1}e^{-2.31}$	$\frac{-174.03}{43}e^{-2.30}$	-1.03E-29			
3	152Eu	0	5.2E-2	5.5E-17			$\frac{23.05}{7}e^{-13.7}$	$\frac{-28.09}{7}e^{-7.60}$	$\frac{-575.4}{2}e^{-10.8}$	$\frac{-471.4}{2}e^{-5.21}$	$\frac{-14.78}{9}e^{-5.20}$	-7.82E-32			
4	154Eu	5.3E-13	8.5E-2	7.4E-17				$\frac{-30.78}{7}e^{-10.9}$	$\frac{48.8}{2}e^{-14.1}$	$\frac{-41.8}{2}e^{-8.51}$	$\frac{8.71}{7}e^{-8.50}$	-7.93E-34			
5	90Sr	1.0E-6	2.4E-2	8.3E-14					$\frac{-61.0}{2}e^{-8.00}$	$\frac{-58.5}{2}e^{-2.41}$	$\frac{28.47}{11}e^{-2.40}$	-2.05E-28			
6	3H	1.5E-9	5.6E-2	9.7E-16						$\frac{-108.7}{2}e^{-5.61}$	$\frac{130.19}{2}e^{-5.60}$	-1.10E-31			
7	14C	5.2E-8	1.2E-4	9.7E-16							$\frac{-17.78}{2}e^{-0.015}$	-2.81E-30			
8	239Pu	7.2E-9													
					2m _h ² = 5.51E+14								Idh _j Idh _k A _{h_jk} = -2.19E-28		
1.01E-12 = Σ[SD(dh _j y _{h_j})] ²															
11. Var(Ō _h) = {Σ[SD(dh _j y _{h_j})] ² } + {2m _h ² }{Idh _j Idh _k A _{h_jk} } =					1.01E-12		+ 5.51E+14		-2.19E-28		= 8.85E-13 = Var(Ō _h)				

- * Column 6 [Radionuclide j and k]--from Form A, Col. 4.
Column 7 [SD($d_{hj}y_{hj}$)]--from Form D, Col. 9.
Column 8 [λ_{hj} and λ_{hk}]--from Table 5.3.
Column 9 [d_{hj} and d_{hk}]--from Form D, Col. 7.

- # $A_{hjk} = \frac{\hat{\sigma}_{hjk}}{n_{hjk}} e^{-\theta}$, where $\hat{\sigma}_{hjk}$ is from Form B, Col. 10,
 n_{hjk} is from Form B, Col. 11, and
 $\theta = (\lambda_{hj} + \lambda_{hk})t$.
■ Column 10--calculated as $d_{hj} I d_{hk} A_{hjk}$.

A.20

- [illegible]

8a. TOTAL = 2.8E-8

- (a) From Form B - Source Term Worksheet (column 10 or 12), for $T = \frac{100}{\text{years}}$.
- (b) From Form C - Availability, Form F - Area, Form G - Depth, or Form H - Mass Correction Worksheets, as appropriate.
- (c) Calculate as the product of Columns 5 and 7.
- (d) Calculate as the product of Columns 8 and 7.
- (e) Calculate as $(\text{Column 8}) \times z_{1-\alpha}$ (Column 9), where $z_{1-\alpha}$ is chosen to correspond with the desired confidence level from Table 5.5.
- (f) Calculate using Procedure 3.2.2, Columns 7, 9, and Columns 8, 9, 10, and 11 from Form A - Summary Statistics Worksheet on Form E (Dose Variable Calculation Worksheet).

Form A - Summary Statistics Table - Estimated Means, Standard Deviations, and Covariances per Stratum

1. Facility and Stratum: 115 D/DR-D Tunnel (Labeled as TD in Table A.1)

2. Preparer's Name: G. F. Piepel

3. Date Prepared: 6/25/86

4. Radionuclide Considered	5. Mean $\bar{x}_{hi}$ (pCi/g)	6. S. D. σ_{hi} (pCi/g)	7. n_i	8. Nuclide i(a)	9. Nuclide j(a)	10. Covariance (σ_{hij}) (a)	11. Number of Samples for which both Nuclides are Analyzed (n_{hij})
Co-60	181.18	162.43	8	Co-60	Cs-137	310728.8	8
Cs-137	944.71	1927.51	8	Co-60	Eu-152	34991.9	8
Eu-152	985.99	225.81	8	Co-60	Eu-154	-1858.2	8
Eu-154	28.53	18.97	8	Co-60	Sr-90	21583.8	2(b)
Sr-90	99.53	132.94	3	Co-60	H-3	3899.8	1(b)
H-3	7.1	14.2	1(c)	Co-60	C-14	-837.7	1(b)
C-14	2.7	5.4	1(c)	Cs-137	Eu-152	370248.8	8
α /Pu-239	8.47	7.33	8	Cs-137	Eu-154	-31971.8	8
				Cs-137	Sr-90	438885.8	3
				Cs-137	H-3	19938.9	1(b)
				Cs-137	C-14	-3981.1	1(b)
				Eu-152	Eu-154	-1495.5	8
				Eu-152	Sr-90	23922.8	2(b)
				Eu-152	H-3	-574.4	1(b)
				Eu-152	C-14	-471.4	1(b)
				Eu-154	Sr-90	-1887.4	2(b)
				Eu-154	H-3	48.8	1(b)
				Eu-154	C-14	-41.8	1(b)
				Sr-90	H-3	-61.8	1(b)
				Sr-90	C-14	-58.5	1(b)
				H-3	C-14	-188.7	1(b)
				α /Pu-239	Co-60	1222.2	8
				α /Pu-239	Cs-137	13885.5	8
				α /Pu-239	Eu-152	1548.8	8
				α /Pu-239	Eu-154	-188.7	8
				α /Pu-239	Sr-90	1743.4	3
				α /Pu-239	H-3	138.19	1(b)
				α /Pu-239	C-14	-17.79	1(b)

(a) Note: Columns 8-11 will be much longer than 4-7.

Comments:

- (b) Insufficient data to estimate covariance. Values given were computed by taking into account other aspects of the data. This was done to complete this example and is not recommended as general practice. For $n_{hij} \leq 2$, use $n_{hij}=1$ in any computations requiring the values.
- (c) Insufficient to compute S.D. Values given computed assuming $\sum x^2 = 2.8$, which was done to complete this example.

Form B - Source-Term Worksheet

1. Facility/Scenario and Stratum: 116 D/DR - D Tunnel, Agricultural Scenario
2. Preparer's Name: G. F. Piepel
3. Date Prepared: 6/26/86
4. Stratum Mass^(a) (grams): 4.01E+6 (a)
5. Number of Years of Decay: 100 (b)

6. Radionuclide Considered	7. Mean Concentration in Stratum, $\bar{x}_{hi}$ (pCi/g) (b)	8. Stratum Concentration S. D., (b) σ_{hi} (pCi/g)	9. Number of Samples in Stratum n_{hi} (b)	10. Source Term y_{hi} (d)	11. S. D. y_{hi} (d)	12. Decayed Source Term (pCi) (e)	13. Decayed S. D. y_{hi} (f)
Co-60	161.18	162.43	8	6.46E+8	2.30E+8	1.46E+3	5.20E+2
Cs-137	944.71	1927.51	8	3.79E+9	2.58E+8	3.79E+8	2.58E+8
Eu-152	306.99	225.01	8	1.23E+9	3.19E+8	-0	-0
Eu-154	28.53	16.97	8	1.14E+9	2.78E+7	2.31E+4	5.84E+3
Sr-90	99.63	132.94	3	3.99E+8	3.08E+8	3.62E+7	2.79E+7
H-3	7.1	14.2	1	2.85E+7	5.69E+7	1.05E+5	2.11E+5
C-14	2.7	5.4	1	1.08E+7	2.17E+7	1.07E+7	2.14E+7
α /Pu-239	8.47	7.33	9	3.47E+7	9.80E+6	3.39E+7	9.77E+6

(a) Strata are defined to consist only of contaminated material.

(b) Copy from Form A - Summary Statistics Table (Table 5.2.1).

(c) Calculate as Item 4 times Item 7.

$$(d) \text{ S. D. } (y_{hi}) = \left(\frac{\sum_{hi}^2}{n_{hi}} \right)^{1/2}$$

(e) Calculate as Item 10 times $e^{-\lambda_i t}$ where λ_i = decay constant for radionuclide i (from Table 5.2.3) and t = number of years of decay (Item 5).

(f) Calculate as Item 11 times $e^{-\lambda_i t}$ where λ_i = decay constant for radionuclide i (from Table 5.2.3) and t = number of years of decay (Item 5).

Form C - Availability Correction Worksheet

1. Facility/Scenario and Stratum: 115 D/DR-D Tunnel, Agricultural Scenario

2. Preparer's Name: Q. F. Piepel

3. Date Prepared: 8/26/86

4. Radionuclide Considered	5. Reduction Factor(a)	6. Unmodified Dose Factors(b)		7. Corrected Dose Factors(c)		8. Residential/HomeGarden(a) Scenario Dose Factors (rem/yr per pCi)
		6a. Inhalation (rem/yr per pCi)	6b. Ingestion (rem/yr per pCi)	7a. Inhalation (rem/yr per pCi)	7b. Ingestion (rem/yr per pCi)	
Co-60	1.0	Not	8.7E-15	Not	8.7E-15	Not Applicable
Cs-137	1.0	Applicable	8.7E-15	Applicable	8.7E-15	
Eu-152	1.0		1.2E-15	for	1.2E-15	
Eu-154	1.0		1.6E-15	Agricultural	1.6E-15	
Sr-90	1.0		1.8E-12	Scenario	1.8E-12	
H-3	1.0		2.1E-14		2.1E-14	
C-14	1.0		2.1E-14		2.1E-14	
α /Pu-239	1.0		7.1E-15		7.1E-15	

(a) RF = 1.0 for radionuclides not contained in activated metals.

RF = $2t$ (corrosion rate cm/yr)/(thickness in cm).

(b) From Table 2.2.

(c) Calculated as the product of columns 5 and 6, by radionuclide.

(d) Complete this column only for the Residential/Home Garden scenario. The scenario dose factors are calculated as Column 7a plus Column 7b.

Form G - Correction Worksheet

1. Facility: 115 D/DR - D Tunnel, Agricultural Scenario

2. Preparer's Name: G. F. Piepel

3. Date Prepared: 6/26/86

4. Depth of Clean overburden 3.35 m.

5. Calculate depth correction factor (DCF) as

$$DCF = e^{-0.92(\text{Depth})} = \underline{0.046}.$$

If Depth > 10 meters DCF = 0.0.

6. Radionuclide	7. Ingestion DF (rem/pCi) (a)	8. Corrected Ingestion DF (rem/pCi) (b)
<u>Co-60</u>	<u>6.7E-15</u>	<u>3.1E-16</u>
<u>Cs-137</u>	<u>6.7E-15</u>	<u>3.1E-16</u>
<u>Eu-152</u>	<u>1.2E-15</u>	<u>5.5E-17</u>
<u>Eu-154</u>	<u>1.6E-15</u>	<u>7.4E-17</u>
<u>Sr-90</u>	<u>1.8E-12</u>	<u>8.3E-14</u>
<u>H-3</u>	<u>2.1E-14</u>	<u>9.7E-16</u>
<u>C-14</u>	<u>2.1E-14</u>	<u>9.7E-16</u>
<u>α/Pu-239</u>	<u>7.1E-15</u>	<u>3.3E-16</u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>

(a) From Area Correction or Availability Worksheet.

(b) Calculated as Item 5 times Column 7.

Form E--Dose Variance [Var($\hat{D}_h$)] Calculation Worksheet for Stratum h

1. Facility and Stratum: 115D/DR - D Tunnel - Agricultural Scenario

2. Preparer's Name: GF Piepel 3. Date Prepared: 6/25/86 4. Stratum Mass (m_h): 4.01E+6 g 5. Decay Time (t): 100 yr

6.*		Radio nuclide de hj				Radio nuclide de hj						10.*									
7.*		8.*		9.*		2		3		4		5		6		7		8		10.*	
SD(dh _{jy} h _j)		λ _{hj}		λ _{hk}		137Cs		152Eu		154Eu		90Sr		3H		14C		239Pu		dhjIdhkJAhjk	
SD(dh _{jy} h _j)		λ _{hj}		λ _{hk}		2.3E-2		5.2E-2		8.5E-2		2.4E-2		5.6E-2		1.2E-4		2.9E-5		dhjIdhkJAhjk	
SD(dh _{jy} h _j)		λ _{hj}		λ _{hk}		3.1E-16		5.5E-17		7.4E-17		8.3E-14		9.7E-16		9.7E-14		3.3E-16		dhjIdhkJAhjk	
1	60Co	1.6E-13	1.3E-1	3.1E-16	3.11E+5 8	15.3	3.50E+4 8	18.2	-1.85E+3 6	21.5	2.16E+4 2	15.4	3.10E+3 1	18.6	-637.7 1	13.0	3.70E+5 8	13.0	2.57E-32		
2	137Cs	1.2E-7	2.3E-2	3.1E-16	-31971.0 6	7.50	4.38E+5 3	10.8	1.99E+4 1	4.70	-3901 1	7.90	-1495.5 6	2.31	2.39E+4 2	2.30	4.04E-27				
3	152Eu	0	5.2E-2	5.5E-17	-574.4 1	13.7	-471.4 1	7.60	-1807.4 2	10.8	48.8 1	5.21	-41.8 1	5.20	3.37E-31						
4	154Eu	1.7E-12	8.5E-2	7.4E-17	-61.0 1	10.9	-58.5 1	14.1	-108.7 1	8.51	1222.2 8	8.50	-1.63E-31								
5	90Sr	2.32E-6	2.4E-2	8.3E-14	1.38E+4 9		1548. 8	-100.7 6	1.40E-25												
6	3H	1.0E-10	5.6E-2	9.7E-16	1743.4 3		130.19 1	-17.79 1	2.00E-28												
7	14C	1.0E-8	1.2E-4	9.7E-16					5.61E-30												
8	239Pu	1.1E-8																			

2.2 - 1.61E+13

IdhjIdhkJAhjk = 1.44E-24

$$11. \text{Var}(\hat{D}_h) = \{ \Sigma [\text{SD}(\text{dh}_{jy}h_j)]^2 \} + \{ 2m_h^2 \} \{ \text{Idh}_{jy}h_j \} = 5.40E-12 + 1.61E+13 + 1.44E-25 = 1.00E-11 = \text{Var}(\hat{D}_h)$$

* Column 6 [Radionuclide j and k]--from Form A, Col. 4.
Column 7 [SD(dh_{jy}h_j)]--from Form D, Col. 9.
Column 8 [λ_{hj} and λ_{hk}]--from Table 5.3.
Column 9 [dh_j and dh_k]--from Form D, Col. 7.

Ahjk = $\frac{\hat{\sigma}_{hjk}}{n_{hjk}} e^{-\theta}$, where $\hat{\sigma}_{hjk}$ is from Form B, Col. 10,
n_{hjk} is from Form B, Col. 11, and
 $\theta = (\lambda_{hj} + \lambda_{hk})t$.
■ Column 10--calculated as dh_jIdh_kAhjk.

A.26

- [illegible]

1.0E-11

- (a) From Form B - Source Term Worksheet (column 10 or 12), for $T = 100$ years.
- (b) From Form C - Availability, Form F - Area, Form G - Depth, or Form H - Mass Correction Worksheets, as appropriate.
- (c) Calculate as the product of Columns 5 and 7.
- (d) Calculate as the product of Columns 8 and 7.
- (e) Calculate as (Column 8) $\times z_{1-\alpha}$ (Column 9), where $z_{1-\alpha}$ is chosen to correspond with the desired confidence level from Table 5.3.4.
- (f) Calculate using Procedure 3.2.2, Columns 7, 9, and Columns 8, 9, 10, and 11 from Form A - Summary Statistics Worksheet on Form E (Dose Variance Calculation Worksheet).

Form A - Summary Statistics Table - Estimated Means, Standard Deviations, and Covariances per Stratum

1. Facility and Stratum: 115 D/DR - DR Tunnel, Agricultural Scenario (Labeled as Room TDR in Table A.1)

2. Preparer's Name: G. F. Piepel

3. Date Prepared: 6/25/86

4. Radionuclide Considered	5. Mean $\bar{x}_{hi}$ (pCi/g)	6. S. D. σ_{hi} (pCi/g)	7. n_i	8. Nuclide i(a)	9. Nuclide j(a)	10. Covariance (σ_{hij}) (a)	11. Number of Samples for which both Nuclides are Analyzed (n_{hij})
Co-60	2.7	12.4	1(c)	Co-60	Cs-137	-287.7	1(b)
Cs-137	324.40	116.88	5	Co-60	Eu-152	276.8	0(b)
Eu-152	36.40	24.03	4	Co-60	Eu-154	122.8	0(b)
Eu-154	13.33	9.93	3	Co-60	Sr-90	58.0	1(b)
Sr-90	8.87	8(d)	3	Co-60	H-3	3099.0	0(b)
H-3	2.7	5.4	1(c)	Co-60	C-14	-673.7	0(b)
C-14	25.1	50.2	1(c)	Cs-137	Eu-152	-3063.7	4
α /Pu-239	4.40	1.14	6	Cs-137	Eu-154	-1331.1	3
				Cs-137	Sr-90	-27.55	3
				Cs-137	H-3	19938.9	1(b)
				Cs-137	C-14	-3901.1	1(b)
				Eu-152	Eu-154	170.07	3
				Eu-152	Sr-90	-43.2	2(b)
				Eu-152	H-3	-574.4	1(b)
				Eu-152	C-14	-471.4	1(b)
				Eu-154	Sr-90	-22.8	2(b)
				Eu-154	H-3	48.8	1(b)
				Eu-154	C-14	-41.8	1(b)
				Sr-90	H-3	-61.0	1(b)
				Sr-90	C-14	-58.5	1(b)
				H-3	C-14	-108.7	1(b)
				α /Pu-239	Co-60	14.2	1(b)
				α /Pu-239	Cs-137	89.6	5
				α /Pu-239	Eu-152	-10.5	4
				α /Pu-239	Eu-154	-8.5	3
				α /Pu-239	Sr-90	-0.5	3
				α /Pu-239	H-3	130.19	1(b)
				α /Pu-239	C-14	-17.79	1(b)

(a) Note: Columns 8-11 will be much longer than 4-7.

Comments:

- (b) Insufficient data to estimate covariance. Values given were computed by taking into account other aspects of the data. This was done to complete this example and is not recommended as general practice. For $n_{hij} \leq 2$, use $n_{hij}=1$ in any computations requiring the values.
- (c) Insufficient to compute S.D. Values given computed assuming $\hat{\sigma}^2/x-2.0$, which was done to complete this example.
- (d) Value computed from 3 data values was artificially small. Value of 6 chosen.

Form B - Source-Term Worksheet

1. Facility/Scenario and Stratum: 115 D/DR - D Tunnel, Agricultural Scenario
2. Preparer's Name: Q. F. Piepel
3. Date Prepared: 8/28/88
4. Stratum Mass^(a) (grams): 4.01E+8 (a)
5. Number of Years of Decay: 100 (t)

8. Radionuclide Considered	7. Mean Concentration in Stratum, $\bar{x}_{hi}$ (pCi/g) (b)	8. Stratum Concentration S. D., (b) σ_{hi} (pCi/g)	9. Number of Samples in Stratum n_{hi} (b)	10. Source Term y_{hi} (d)	11. S. D. y_{hi} (d)	12. Decayed Source Term (pCi) (e)	13. Decayed S. D. y_{hi} (f)
Co-60	161.18	162.43	8	6.48E+8	2.30E+8	1.48E+3	5.20E+2
Cs-137	944.71	1927.51	9	3.79E+9	2.68E+9	3.79E+8	2.68E+8
Eu-152	305.99	225.01	8	1.23E+9	3.19E+8	0	0
Eu-154	28.53	16.97	9	1.14E+9	2.78E+7	2.31E+4	5.64E+3
Sr-90	99.53	132.94	3	3.99E+8	3.08E+8	3.62E+7	2.79E+7
H-3	7.1	14.2	1	2.86E+7	5.69E+7	1.05E+6	2.11E+6
C-14	2.7	5.4	1	1.08E+7	2.17E+7	1.07E+7	2.14E+7
α/Pu-239	8.47	7.33	9	3.47E+7	9.80E+6	3.39E+7	9.77E+6

(a) Strata are defined to consist only of contaminated material.

(b) Copy from Form A - Summary Statistics Table (Table 5.2.1).

(c) Calculate as Item 4 times Item 7.

$$(d) \text{ S. D. } (y_{hi}) = \left(\frac{\sum_{hi}^2}{n_{hi}} \right)^{1/2}$$

(e) Calculate as Item 10 times $e^{-\lambda_i t}$ where λ_i = decay constant for radionuclide i (from Table 5.2.3) and t = number of years of decay (Item 5).

(f) Calculate as Item 11 times $e^{-\lambda_i t}$ where λ_i = decay constant for radionuclide i (from Table 5.2.3) and t = number of years of decay (Item 5).

Form C - Availability Correction Worksheet

1. Facility/Scenario and Stratum: 115 D/DR - D Tunnel, Agricultural Scenario

2. Preparer's Name: G. F. Piepel

3. Date Prepared: 6/28/86

4. Radionuclide Considered	5. Reduction Factor(a)	6. Unmodified Dose Factors(b)		7. Corrected Dose Factors(c)		8. Residential/HomeGarden(a) Scenario Dose Factors (rem/yr per pCi)
		6a. Inhalation (rem/yr per pCi)	6b. Ingestion (rem/yr per pCi)	7a. Inhalation (rem/yr per pCi)	7b. Ingestion (rem/yr per pCi)	
Co-60	1.0	Not	8.7E-15	Not	8.7E-15	Not Applicable
Cs-137	1.0	Applicable	8.7E-15	Applicable	8.7E-15	
Eu-152	1.0		1.2E-15		1.2E-15	
Eu-154	1.0		1.8E-15		1.8E-15	
Sr-90	1.0		1.8E-12		1.8E-12	
H-3	1.0		2.1E-14		2.1E-14	
C-14	1.0		2.1E-14		2.1E-14	
α/Pu-239	1.0		7.1E-15		7.1E-15	

(a) RF = 1.0 for radionuclides not contained in activated metals.

RF = $2t$ (corrosion rate cm/yr)/(thickness in cm).

(b) From Table 2.2.

(c) Calculated as the product of columns 5 and 6, by radionuclide.

(d) Complete this column only for the Residential/Home Garden scenario. The scenario dose factors are calculated as Column 7a plus Column 7b.

Form G - Depth Correction Worksheet

1. Facility: 115 D/DR Tunnel, Agricultural Scenario

2. Preparer's Name: G. F. Piepel

3. Date Prepared: 6/26/86

4. Depth of Clean overburden 3.35 m.

5. Calculate depth correction factor (DCF) as

$$DCF = e^{-0.92(\text{Depth})} = \underline{0.046}.$$

If Depth > 10 meters DCF = 0.0.

6. Radionuclide

7. Ingestion DF (rem/pCi) (a)

8. Corrected Ingestion
DF (rem/pCi) (b)

<u>Co-60</u>	<u>6.7E-15</u>	<u>3.1E-16</u>
<u>Cs-137</u>	<u>6.7E-15</u>	<u>3.1E-16</u>
<u>Eu-152</u>	<u>1.2E-15</u>	<u>5.5E-17</u>
<u>Eu-154</u>	<u>1.6E-15</u>	<u>7.4E-17</u>
<u>Sr-90</u>	<u>1.8E-12</u>	<u>8.3E-14</u>
<u>H-3</u>	<u>2.1E-14</u>	<u>9.7E-16</u>
<u>C-14</u>	<u>2.1E-14</u>	<u>9.7E-16</u>
<u>αPu-239</u>	<u>7.1E-15</u>	<u>3.3E-16</u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>

(a) From Area Correction or Availability Worksheet.

(b) Calculated as Item 5 times Column 7.

Form E--Dose Variance [Var(D_h)] Calculation Worksheet for Stratum h

1. Facility and Stratum: 115D/DR - DR Tunnel - Agricultural Scenario

2. Preparer's Name: GF Piepel 3. Date Prepared: 6/25/86 4. Stratum Mass (m_h): 4.46E+6g 5. Decay Time (t): 100 yr

6.*		7.*		8.*		9.*		10.*	
j	Radio nuclide	SD(d _h jy _h j)	λ _h j	d _h j	λ _h k	d _h k	A _h jk	d _h jZd _h kA _h jk	10.*
1	⁶⁰ Co	3.9E-14	1.3E-1	3.1E-16	2.3E-2	8.1E-16	137Cs	-287.7 1	-3.21E-34
2	¹³⁷ Cs	7.2E-9	2.3E-2	3.1E-16	5.2E-2	5.5E-17	152Eu	276.8 1	4.67E-27
3	¹⁵² Eu	0	5.2E-2	5.5E-17	8.5E-2	7.4E-17	154Eu	122.8 1	-1.07E-30
4	¹⁵⁴ Eu	3.8E-13	8.5E-2	7.4E-17	2.4E-2	8.3E-14	90Sr	58.0 1	-8.42E-33
5	⁹⁰ Sr	1.2E-7	2.4E-2	8.3E-14	5.6E-2	9.7E-16	3H	3099 1	-3.99E-29
6	³ H	8.6E-11	5.6E-2	9.7E-16	1.2E-4	9.7E-14	14C	-0.5 3	1.53E-31
7	¹⁴ C	2.2E-7	1.2E-4	9.7E-14	2.9E-5	3.3E-16	239Pu	-17.79 1	-5.61E-28
8	²³⁹ Pu	7.5E-10							

11. Var(D_h) = {Σ[SD(d_hjy_hj)]²} + {2m_h²} {Σd_hjZd_hkA_hjk} = 6.29E-14 + 3.98E+13 + 4.07E-27 = 1.44E-13 = Var(D_h)

* Column 6 [Radionuclide j and k]--from Form A, Col. 4.
Column 7 [SD(d_hjy_hj)]--from Form D, Col. 9.
Column 8 [λ_hj and λ_hk]--from Table 5.3.
Column 9 [d_hj and d_hk]--from Form D, Col. 7.

A_hjk = $\frac{\hat{\sigma}_{hjk}}{n_{hjk}} e^{-B}$, where $\hat{\sigma}_{hjk}$ is from Form B, Col. 10,
n_hjk is from Form B, Col. 11, and
B = (λ_hj + λ_hk)t.
Column 10--calculated as d_hjZd_hkA_hjk.

A. 32

- [illegible]

1.4E-13

- From Form B - Source Term Worksheet (column 10 or 12), for $T =$ _____ years.
- From Form C - Availability, Form F - Area, Form G - Depth, or Form H - Mass Correction Worksheets, as appropriate.
- Calculate as the product of Columns 5 and 7.
- Calculate as the product of Columns 6 and 7.
- Calculate as $(\text{Column 8}) \times z_{1-\alpha}$ (Column 9), where $z_{1-\alpha}$ is chosen to correspond with the desired confidence level from Table 5.5.
- Calculate using Procedure 3.2.2, Columns 7, 9, and Columns 8, 9, 10, and 11 from Form A - Summary Statistics Worksheet on Form E (Dose Variance Calculation Worksheet).

Form A - Summary Statistics Table - Estimated Means, Standard Deviations, and Covariances per Stratum

1. Facility and Stratum: 115 D/DR - Small portion of Dryer Room #1 (Labeled as observation Number 16, Room DR1 in Table A.1)

2. Preparer's Name: G. F. Piepel

3. Date Prepared: 8/25/88

4. Radionuclide Considered	5. Mean $\bar{x}_{hi}$ (pCi/g)	6. S. D. σ_{hi} (pCi/g)	7. n_i	8. Nuclide i(a)	9. Nuclide j(a)	10. Covariance $(\sigma_{hij})(a)$	11. Number of Samples for which both Nuclides are Analyzed (n_{hij})
Co-60	51.5	25.8	1(c)	Co-60	Cs-137	-4574.3	1(b)
Cs-137	1978.8	985.8	1(c)	Co-60	Eu-152	184.9	1(b)
Eu-152	13.8	6.8	1(c)	Co-60	Eu-154	268.3	1(b)
Eu-154	28.9	18.4	1(c)	Co-60	Sr-90	34568.8	1(b)
Sr-90	3488.88	1748.8	1(c)	Co-60	H-3	587.7	1(b)
H-3	17.82	25.31	8(d)	Co-60	C-14	-127.2	1(b)
C-14	11.68	9.88	8(d)	Cs-137	Eu-152	-2813.2	1(b)
α/Pu-239	9.28	1.88	1(c)	Cs-137	Eu-154	-717.1	1(b)
				Cs-137	Sr-90	1199738.8	1(b)
				Cs-137	H-3	24938.4	1(b)
				Cs-137	C-14	-4858.1	1(b)
				Eu-152	Eu-154	43.8	1(b)
				Eu-152	Sr-90	-3431.3	1(b)
				Eu-152	H-3	-49.9	1(b)
				Eu-152	C-14	-48.2	1(b)
				Eu-154	Sr-90	-6876.5	1(b)
				Eu-154	H-3	118.8	1(b)
				Eu-154	C-14	-95.4	1(b)
				Sr-90	H-3	-10569.5	1(b)
				Sr-90	C-14	-18122.3	1(b)
				H-3	C-14	-187.3	1(b)
				α/Pu-239	Co-60	23.1	1(b)
				α/Pu-239	Cs-137	-189.1	1(b)
				α/Pu-239	Eu-152	-3.3	1(b)
				α/Pu-239	Eu-154	5.8	1(b)
				α/Pu-239	Sr-90	-868.2	1(b)
				α/Pu-239	H-3	-9.7	1(b)
				α/Pu-239	C-14	-5.2	1(b)

(a) Note: Columns 8-11 will be much longer than 4-7.

Comments:

(b) Insufficient data to estimate covariance. Values given were computed by taking into account other aspects of the data. This was done to complete this example and is not recommended as general practice. For $n_{hij} \leq 2$, use $n_{hij}=1$ in any computations requiring the values.

(c) Values from the one sample are entered as means, and assumes $\hat{\sigma}/\bar{x} = 8.5$; for example only.

(d) Means and S.D. based on four available data points in the whole building.

Form B - Source-Term Worksheet

1. Facility/Scenario and Stratum: 115 D/DR-DR - Residential Home Garden Scenario - Small Portion of Dry Room #1
2. Preparer's Name: G. F. Piepel
3. Date Prepared: 8/28/88
4. Stratum Mass^(a) (grams): 8.3E+4 (a)
5. Number of Years of Decay: 100 (t)

6. Radionuclide Considered	7. Mean Concentration in Stratum, $\bar{x}_{hi}$ (pCi/g) (b)	8. Stratum Concentration S. D., (b) σ_{hi} (pCi/g)	9. Number of Samples in Stratum n_{hi} (b)	10. Source Term y_{hi} (d)	11. S. D. y_{hi} (d)	12. Decayed Source Term (pCi) (e)	13. Decayed S. D. y_{hi} (f)
Co-60	51.5	25.8	1	4.29E+6	2.15E+6	9.71E+0	4.88E+0
Cs-137	1970.0	985.0	1	1.84E+8	8.20E+7	1.84E+7	8.20E+6
Eu-152	13.8	8.8	1	1.13E+6	5.88E+5	-0	-0
Eu-154	20.9	10.4	1	1.74E+6	8.88E+5	3.53E+2	1.78E+2
Sr-90	3480.0	1740.0	1	2.90E+8	1.45E+8	2.83E+7	1.32E+7
H-3	1782.0	25.31	0 (use 1)	1.47E+6	2.11E+6	5.44E+3	7.81E+3
C-14	11.88	9.88	0 (use 1)	9.73E+5	8.21E+5	9.81E+5	8.11E+5
α /Pu-239	3.20	1.80	1	2.66E+5	1.33E+5	2.85E+5	1.33E+5

(a) Strata are defined to consist only of contaminated material.

(b) Copy from Form A - Summary Statistics Table (Table 5.2.1).

(c) Calculate as Item 4 times Item 7.

$$(d) \text{ S. D. } (y_{hi}) = \left(\frac{\sum_{hi}^2}{n_{hi}} \right)^{1/2}$$

(e) Calculate as Item 10 times $e^{-\lambda_i t}$ where λ_i = decay constant for radionuclide i (from Table 5.2.3) and t = number of years of decay (Item 5).

(f) Calculate as Item 11 times $e^{-\lambda_i t}$ where λ_i = decay constant for radionuclide i (from Table 5.2.3) and t = number of years of decay (Item 5).

Form C - Availability Correction Worksheet

1. Facility/Scenario and Stratum: 115 D/DR - Residential-Home Garden Scenario, Small Portion of Dryer Room #1

2. Preparer's Name: G. F. Piepel

3. Date Prepared: 8/28/86

4. Radionuclide Considered	5. Reduction Factor(a)	6. Unmodified Dose Factors(b)		7. Corrected Dose Factors(c)		8. Residential/HomeGarden(d) Scenario Dose Factors (rem/yr per pCi)
		6a. Inhalation (rem/yr per pCi)	6b. Ingestion (rem/yr per pCi)	7a. Inhalation (rem/yr per pCi)	7b. Ingestion (rem/yr per pCi)	
Co-60	1.0	4.3E-12	1.0E-14	4.3E-12	1.0E-14	4.3E-12
Cs-137	1.0	1.0E-12	1.0E-14	1.0E-12	1.0E-14	1.0E-12
Eu-152	1.0	2.0E-12	1.7E-15	2.0E-12	1.7E-15	2.0E-12
Eu-154	1.0	2.1E-12	2.6E-15	2.1E-12	2.6E-15	2.1E-12
Sr-90	1.0	1.1E-14	2.7E-12	1.1E-14	2.7E-12	2.7E-12
H-3	1.0	2.4E-20	3.1E-14	2.4E-20	3.1E-14	3.1E-14
C-14	1.0	4.1E-18	3.1E-14	4.1E-18	3.1E-14	3.1E-14
α/Pu-239	1.0	8.7E-14	1.1E-14	8.7E-14	1.1E-14	9.8E-14

(a) RF = 1.0 for radionuclides not contained in activated metals.
RF = 2t (corrosion rate cm/yr)/(thickness in cm).

(b) From Table 2.2.

(c) Calculated as the product of columns 5 and 6, by radionuclide.

(d) Complete this column only for the Residential/Home Garden scenario. The scenario dose factors are calculated as Column 7a plus Column 7b.

Form E--Dose Variance [Var($\hat{D}_h$)] Calculation Worksheet for Stratum h

1. Facility and Stratum: 1150/DR - Small portion of Dryer Room #1 - Residential Home

Garden Scenario

2. Preparer's Name: GF Piepel 3. Date Prepared: 6/26/86 4. Stratum Mass (m_h): 8.3E+4 g 5. Decay Time (t): 100 yr

				A _{hjk} #								
6.*	k:			2	3	4	5	6	7	8		
Radio nucli de hj	Radionuclide hk:			137Cs	152Eu	154Eu	90Sr	3H	14C	239Pu		
7.*	8.*	9.*	λ _h k:	2.3E-2	5.2E-2	8.5E-2	2.4E-2	5.6E-2	1.2E-4	2.9E-5	10.■	
SD(d _h jy _h j)	λ _h j	d _h j\dhk:	1.0E-12	2.0E-12	2.1E-12	2.7E-12	3.1E-14	3.1E-14	9.8E-14	d _h jΣd _h kA _{hjk}		
1	60Co	2.11E-11	1.3E-1	4.3E-12	-457.8 1	164.8 1	268.2 1	34567 1	587.7 1	-127.2 1	-2813 1	7.51E-26
2	137Cs	8.2E-6	2.3E-2	1.0E-12	-717.1 1	1.2E+6 1	24930 1	-4856 1	43.8 1	-3436 1		6.29E-22
3	152Eu	0	5.2E-2	2.0E-12		-49.8 1	-40.2 1	-6872 1	110.6 1	-95.4 1		-1.83E-25
4	154Eu	3.7E-10	8.5E-2	2.1E-12			-10569 1	-10122 1	-107.3 1	-23.1 1		-1.11E-24
5	90Sr	3.6E-5	2.4E-2	2.7E-12				-189.1 1	-3.3 1	5.0 1		8.65E-26
6	3H	2.4E-10	5.6E-2	3.1E-14					-668.2 1	-9.7 1		-2.45E-27
7	14C	2.5E-8	1.2E-4	3.1E-14						-5.2 1		-1.56E-26
8	239Pu	1.3E-8										

2m,2 = 1.38E+10

Σd_hjΣd_hkA_{hjk} = 6.28E-22

$$1.43E-9 = \Sigma [SD(d_{hj}y_{hj})]^2$$

$$2m_h^2 = 1.38E+10$$

$$\Sigma d_{hj} \Sigma d_{hk} A_{hjk} = 6.28E-22$$

$$11. \text{Var}(\hat{D}_h) = \{\Sigma [SD(d_{hj}y_{hj})]^2\} + \{2m_h^2\} \{\Sigma d_{hj} \Sigma d_{hk} A_{hjk}\} = 1.43E-9 + 1.38E+10 \cdot 6.28E-22 = 1.44E-9 = \text{Var}(\hat{D}_h)$$

* Column 6 [Radionuclide j and k]--from Form A, Col. 4.

Column 7 [SD(d_hjy_hj)]--from Form D, Col. 9.

Column 8 [λ_hj and λ_hk]--from Table 5.3.

Column 9 [d_hj and d_hk]--from Form D, Col. 7.

■ A_{hjk} = $\frac{\hat{\sigma}_{hjk}}{n_{hjk}} e^{-\theta}$, where $\hat{\sigma}_{hjk}$ is from Form B, Col. 10,
n_{hjk} is from Form B, Col. 11, and
 $\theta = (\lambda_{hj} + \lambda_{hk})t$.

■ Column 10--calculated as d_{hj}Σd_{hk}A_{hjk}.

Form D - Stratum Dose Calculation Worksheet

1. Facility/Scenario and Stratum: 115 D/DR - Residential-Home Garden Scenario - Small Portion of Dryer Room #1
2. Preparer's Name: Q. F. Piepel
3. Date Prepared: 8/26/86

4. Radionuclide Considered	5. Source Term (pCi) (a)	6. Source Term S. D. (pCi) (a)	7. Scenario Dose Factor(b) (rem/yr per pCi)	8. Estimated Dose(c) (rem/yr)	9. S. D. of Dose(d) (rem/yr)	10. Estimated 90 % UCL Dose(e) (rem/yr)
Co-60	9.70E+0	4.00E+0	4.3E-12	4.2E-11	2.11E-11	8.9E-11
Cs-137	1.84E+7	8.23E+6	1.0E-11	1.8E-5	8.2E-6	2.7E-5
Eu-152	-0	-0	2.0E-12	-0	-0	-0
Eu-154	3.53E+2	1.78E+2	2.1E-12	7.4E-10	3.7E-10	1.2E-9
Sr-90	2.83E+7	1.32E+7	2.7E-12	7.1E-5	3.6E-5	1.2E-4
H-3	64.4E+3	7.81E+3	3.1E-14	1.7E-10	2.4E-10	4.8E-10
C-14	9.81E+5	8.11E+5	3.1E-14	3.0E-8	2.5E-8	8.2E-8
α/Pu-239	2.85E+5	1.33E+5	9.8E-14	2.8E-8	1.3E-8	4.3E-8
				8.7E-5		

8a. TOTAL =

11. Variance of Estimated Dose (rem/yr)^{2(f)}:

1.40E-9

- (a) From Form B - Source Term Worksheet (column 10 or 12), for T = 100 years.
- (b) From Form C - Availability, Form F - Area, Form G - Depth, or Form H - Mass Correction Worksheets, as appropriate.
- (c) Calculate as the product of Columns 5 and 7.
- (d) Calculate as the product of Columns 8 and 7.
- (e) Calculate as (Column 8) + z_{1-α} (Column 9), where z_{1-α} is chosen to correspond with the desired confidence level from Table 5.5.
- (f) Calculate using Procedure 3.2.2, Columns 7, 9, and Columns 8, 9, 10, and 11 from Form A - Summary Statistics Worksheet on Form E (Dose Variance Calculation Worksheet).

Form A - Summary Statistics Table - Estimated Means, Standard Deviations, and Covariances per Stratum

1. Facility and Stratum: 117D Stratum A (Inlet Duct, A Filter Cell, A Outlet Seal Pit) - Labeled Room ID, AFC, AOSP in Table A.4)

2. Preparer's Name: G. F. Piepel 3. Date Prepared: 6/25/88

4. Radionuclide Considered	5. Mean $\bar{x}_{hi}$ (pCi/g)	6. S. D. σ_{hi} (pCi/g)	7. n_i	8. Nuclide i(a)	9. Nuclide j(a)	10. Covariance $(\sigma_{hij})(a)$	11. Number of Samples for which both Nuclides are Analyzed (n_{hij})
Pu-239	7.63	4.17	15	Pu-239	H-3	1271.67	3
H-3	271.33	365.01	3	Pu-239	C-14	838.83	3
C-14	289.87	288.38	3	Pu-239	Co-60	265.42	4
Co-60	107.75	88.72	4	Pu-239	Sr-90	1637.86	5
Sr-90	489.80	248.78	6	Pu-239	Cs-137	968.81	14
Cs-137	425.50	410.72	14	Pu-239	Eu-152	97.29	2(b)
Eu-152	47.60	23.33	2	H-3	C-14	100182.8	3
				H-3	Co-60	29463.6	2(b)
				H-3	Sr-90	82331.7	3
				H-3	Cs-137	167602.8	3
				H-3	Eu-152	8515.7	2(b)
				C-14	Co-60	22632.3	2(b)
				C-14	Sr-90	82284.8	3
				C-14	Cs-137	141112.0	3
				C-14	Eu-152	6541.3	2(b)
				Co-60	Sr-90	18227.7	4
				Co-60	Cs-137	40624.7	4
				Co-60	Eu-152	1883.2	1(b)
				Sr-90	Cs-137	119417.0	5
				Sr-90	Eu-152	6768.9	2(b)
				Cs-137	Eu-152	9582.1	2(b)

(a) Note: Columns 8-11 will be much longer than 4-7.

Comments:

(b) Insufficient data to estimate covariance. Values given were computed by taking into account other aspects of the data. This was done to complete this example and is not recommended as general practice. For $n_{hij} \leq 2$, use $n_{hij}=1$ in any computations requiring the values.

Form B - Source-Term Worksheet

1. Facility/Scenario and Stratum: 117 D - Stratum B, Agricultural Scenario
2. Preparer's Name: G. F. Piepel
3. Date Prepared: 8/28/88
4. Stratum Mass^(a) (grams): 8.77E+6 (m)
6. Number of Years of Decay: 100 (t)

8. Radionuclide Considered	7. Mean Concentration in Stratum, $\bar{x}_{hi}$ (pCi/g) (b)	8. Stratum Concentration S. D., (b) σ_{hi} (pCi/g)	9. Number of Samples in Stratum n_{hi} (b)	10. Source Term y_{hi} (d)	11. S. D. y_{hi} (d)	12. Decayed Source Term (pCi) (e)	13. Decayed S. D. y_{hi} (f)
Pu-239	21.65	18.07	20	1.90E+7	3.15E+8	1.89E+7	3.14E+8
H-3	30.80	38.92	5	2.70E+7	1.45E+7	9.99E+4	5.37E+4
C-14	47.00	47.55	5	4.12E+7	1.86E+7	4.07E+7	1.84E+7
Co-60	235.50	309.00	2	2.07E+8	1.92E+8	4.88E+2	4.34E+2
Sr-90	14.28	18.55	7	1.26E+7	8.15E+6	1.13E+6	6.68E+6
Cs-137	518.23	938.40	13	4.64E+8	2.28E+8	4.64E+7	2.28E+7
Cu-152	47.50	23.33	0 (use 1)	4.17E+7	2.05E+7	=0	=0

(a) Strata are defined to consist only of contaminated material.

(b) Copy from Form A - Summary Statistics Table.

(c) Calculate as Item 4 times Item 7.

(d) $S. D. (y_{hi}) = \left(\frac{\bar{x}_{hi}^2}{n_{hi}} \right)^{1/2}$

(e) Calculate as Item 10 times $e^{-\lambda_i t}$ where λ_i = decay constant for radionuclide i (from Table 5.2.3) and t = number of years of decay (Item 5).

(f) Calculate as Item 11 times $e^{-\lambda_i t}$ where λ_i = decay constant for radionuclide i (from Table 5.2.3) and t = number of years of decay (Item 5).

TABLE 5.4.3. Depth Correction Worksheet

1. Facility: 117 D - Stratum A, Agricultural Scenario

2. Preparer's Name: G. F. Piepel

3. Date Prepared: 6/26/86

4. Depth of Clean overburden 6.7 m.

5. Calculate depth correction factor (DCF) as

$$DCF = e^{-0.92(\text{Depth})} = \underline{2.1E-3}.$$

If Depth > 10 meters DCF = 0.0.

6. Radionuclide	7. Ingestion DF (rem/pCi)(a)	8. Corrected Ingestion DF(rem/pCi)(b)
<u>Pu-239</u>	<u>1.1E-14</u>	<u>2.3E-17</u>
<u>H-3</u>	<u>3.1E-14</u>	<u>6.5E-17</u>
<u>C-14</u>	<u>3.1E-14</u>	<u>6.5E-17</u>
<u>Co-160</u>	<u>1.0E-14</u>	<u>2.1E-17</u>
<u>Sr-90</u>	<u>2.7E-12</u>	<u>5.7E-15</u>
<u>Cs-137</u>	<u>1.0E-14</u>	<u>2.1E-17</u>
<u>Eu-152</u>	<u>1.7E-15</u>	<u>3.6E-18</u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>

(a) From Area Correction or Availability Worksheet.

(b) Calculated as Item 5 times Column 7.

Form E--Dose Variance [Var($\hat{D}_h$)] Calculation Worksheet for Stratum h

1. Facility and Stratum: 117D Stratum A - Agricultural Scenario

2. Preparer's Name: GF Piepel 3. Date Prepared: 6/26/86 4. Stratum Mass (m_h): 8.19E+5 g 5. Decay Time (t): 100 yr

6.*		7.*		8.*		9.*		10.*	
Radio nuclide j	SD($d_{hj}y_{hj}$)	λ_{hj}	d_{hj}	Radionuclide hk	λ_{hk}	d_{hk}	A_{hjk}	$\hat{D}_{hjk}$	$\hat{D}_{hjk}$
1 ²³⁹ Pu	2.0E-11	2.9E-5	2.3E-17	³ H	5.6E-2	6.5E-17	127.67 3	5.60	4.06E-30
2 ³ H	4.2E-11	5.6E-2	6.5E-17	¹⁴ C	1.2E-4	6.5E-17	838.8 3	-0.015	5.34E-30
3 ¹⁴ C	8.5E-9	1.2E-4	6.5E-17	⁶⁰ Co	1.3E-1	2.1E-17	265.42 4	13.0	6.96E-28
4 ⁶⁰ Co	1.6E-15	1.3E-1	2.1E-17	⁹⁰ Sr	2.4E-2	5.7E-15	1537.05 5	2.40	1.13E-34
5 ⁹⁰ Sr	4.7E-8	2.4E-2	5.7E-15	¹³⁷ Cs	2.3E-2	2.1E-17	966.81 14	2.30	2.60E-29
6 ¹³⁷ Cs	1.9E-10	2.3E-2	2.1E-17	¹⁵² Eu	5.2E-2	3.6E-18	97.29 2	5.20	2.00E-34
7 ¹⁵² Eu	0	5.2E-2	3.6E-18						
8									

$2.28E-15 = \Sigma [SD(d_{hj}y_{hj})]^2$

$2m_h^2 = 1.34E+12$

$\Sigma d_{hj} \Sigma d_{hk} A_{hjk} = 7.31E-28$

11. $Var(\hat{D}_h) = \{ \Sigma [SD(d_{hj}y_{hj})]^2 \} + \{ 2m_h^2 \} \{ \Sigma d_{hj} \Sigma d_{hk} A_{hjk} \} = 2.28E-15 + 1.34E+12 \cdot 7.31E-28 = 3.3E-15 = Var(\hat{D}_h)$

* Column 6 [Radionuclide j and k]--from Form A, Col. 4.
 Column 7 [SD($d_{hj}y_{hj}$)]--from Form D, Col. 9.
 Column 8 [λ_{hj} and λ_{hk}]--from Table 5.3.
 Column 9 [d_{hj} and d_{hk}]--from Form D, Col. 7.

$A_{hjk} = \frac{\hat{\sigma}_{hjk}}{n_{hjk}} e^{-\theta}$, where $\hat{\sigma}_{hjk}$ is from Form B, Col. 10,
 n_{hjk} is from Form B, Col. 11, and
 $\theta = (\lambda_{hj} + \lambda_{hk})t$.
 ■ Column 10--calculated as $d_{hj} \Sigma d_{hk} A_{hjk}$.

A.42

Form D - Stratum Dose Calculation Worksheet

1. Facility/Scenario and Stratum: 117 Stratum A, Agricultural Scenario
2. Preparer's Name: G. F. Piepel
3. Date Prepared: 8/28/88

4. Radionuclide Considered	5. Source Term (pCi) (a)	6. Source Term S. D. (pCi) (a)	7. Scenario Dose Factor (b) (rem/yr per pCi)	8. Estimated Dose (c) (rem/yr)	9. S. D. of Dose (d) (rem/yr)	10. Estimated 90 % UCL Dose (e) (rem/yr)
Pu-239	6.15E+6	8.79E+6	2.9E-17	1.4E-10	2.0E-11	1.7E-10
H-3	8.21E+6	8.40E+6	6.5E-17	5.3E-11	4.2E-11	1.1E-10
C-14	2.18E+8	1.31E+8	6.6E-17	1.4E-8	8.5E-9	2.5E-8
Co-60	1.99E+2	7.47E+1	2.1E-17	4.2E-15	1.8E-15	8.2E-15
Sr-90	3.05E+7	8.20E+6	5.7E-16	1.7E-7	4.7E-8	2.3E-7
Cs-137	3.48E+7	8.99E+6	2.1E-17	7.3E-10	1.9E-10	9.7E-10
Eu-152	~8	~0	3.8E-18	~0	~0	~0
			8a. TOTAL =	1.9E-7		

11. Variance of Estimated Dose (rem/yr)^{2(f)}:

3.3E-15

- From Form B - Source Term Worksheet (column 10 or 12), for $T = 100$ years.
- From Form C - Availability, Form F - Area, Form G - Depth, or Form H - Mass Correction Worksheets, as appropriate.
- Calculate as the product of Columns 5 and 7.
- Calculate as the product of Columns 8 and 7.
- Calculate as $(\text{Column 8}) \div z_{1-\alpha}$ (Column 9), where $z_{1-\alpha}$ is chosen to correspond with the desired confidence level from Table 5.5.
- Calculate using Procedure 3.2.2, Columns 7, 9, and Columns 8, 9, 10, and 11 from Form A - Summary Statistics Worksheet on Form E (Dose Variance Calculation Worksheet).

Form A - Summary Statistics Table - Estimated Means, Standard Deviations, and Covariances per Stratum

1. Facility and Stratum: 117D - Stratum B (Outlet Duct, A Inlet Sand Pit, B Filter Cell, B Outlet Seal Pit) - Labeled as Room OD, AISP, BISP, BFC, and BOSP in Table A.4)

2. Preparer's Name: Q. F. Piepel 3. Date Prepared: 8/26/86

4. Radionuclide Considered	5. Mean $\bar{x}_{hi}$ (pCi/g)	6. S. D. σ_{hi} (pCi/g)	7. n_i	8. Nuclide i(a)	9. Nuclide j(a)	10. Covariance (σ_{hij}) (a)	11. Number of Samples for which both Nuclides are Analyzed (n_{hij})
Pu-239	21.65	16.07	20	Pu-239	H-3	93.2	5
H-3	38.68	36.92	5	Pu-239	C-14	24.75	5
C-14	47.00	47.55	5	Pu-239	Co-60	4985.6	2(b)
Co-60	235.50	309.00	2	Pu-239	Sr-90	134.48	7
Sr-90	14.28	18.55	7	Pu-239	Cs-137	2528.46	13
Cs-137	518.23	938.40	13	Pu-239	Eu-152	374.9	0(b)
Eu-152	47.50	23.33	0(c)	H-3	C-14	1735.75	5
				H-3	Co-60	11488.3	1(b)
				H-3	Sr-90	-83.4	5
				H-3	Cs-137	27216.9	5
				H-3	Eu-152	861.34	0(b)
				C-14	Co-60	4693.0	1(b)
				C-14	Sr-90	-107.75	5
				C-14	Cs-137	35538.0	5
				C-14	Eu-152	1109.3	0(b)
				Co-60	Sr-90	5732.0	2(b)
				Co-60	Cs-137	209985.6	2(b)
				Co-60	Eu-152	7209.0	0(b)
				Sr-90	Cs-137	17980.8	7
				Sr-90	Eu-152	432.8	0(b)
				Cs-137	Eu-152	21892.9	0(b)

(a) Note: Columns 8-11 will be much longer than 4-7.

Comments:

(b) Insufficient data to estimate covariance. Values given were computed by taking into account other aspects of the data. This was done to complete this example and is not recommended as general practice. For $n_{hij} \leq 2$, use $n_{hij}=1$ in any computations requiring the values.

(c) In the absence of any data, the same value for Stratum A are used.

Form B - Source-Term Worksheet

1. Facility/Scenario and Stratum: 117 D - Stratum B, Agricultural Scenario

2. Preparer's Name: G. F. Piepel

3. Date Prepared: 6/28/88

4. Stratum Mass^(a) (grams): 8.77E+5 (a)

5. Number of Years of Decay: 100 (t)

6. Radionuclide Considered	7. Mean Concentration in Stratum, $\bar{x}_{hi}$ (pCi/g) (b)	8. Stratum Concentration S. D., s_{hi} (pCi/g) (b)	9. Number of Samples in Stratum n_{hi} (b)	10. Source Term y_{hi} (d)	11. S. D. y_{hi} (d)	12. Decayed Source Term (pCi) (e)	13. Decayed S. D. y_{hi} (f)
Pu-239	21.65	18.07	20	1.90E+7	3.15E+6	1.89E+7	3.14E+6
H-3	30.80	38.92	6	2.70E+7	1.45E+7	9.99E+4	5.37E+4
C-14	47.00	47.55	6	4.12E+7	1.68E+7	4.07E+7	1.84E+7
Co-60	235.50	309.00	2	2.07E+8	1.92E+8	4.68E+2	4.34E+2
Sr-90	14.28	18.65	7	1.26E+7	8.15E+6	1.13E+8	5.58E+5
Cs-137	518.23	938.40	13	4.54E+8	2.28E+8	4.54E+7	2.28E+7
Cu-152	47.50	23.83	0 (use 1)	4.17E+7	2.05E+7	=0	=0

(a) Strata are defined to consist only of contaminated material.

(b) Copy from Form A - Summary Statistics Table.

(c) Calculate as Item 4 times Item 7.

(d) S. D. (y_{hi}) = $\left(\frac{s_{hi}^2}{n_{hi}} \right)^{1/2}$

(e) Calculate as Item 10 times $e^{-\lambda_i t}$ where λ_i = decay constant for radionuclide i (from Table 5.2.3) and t = number of years of decay (Item 5).

(f) Calculate as Item 11 times $e^{-\lambda_i t}$ where λ_i = decay constant for radionuclide i (from Table 5.2.3) and t = number of years of decay (Item 5).

A.45

- [illegible]

- (a) $RF = 1.0$ for radionuclides not contained in activated metals.
 $RF = 2t$ (corrosion rate cm/yr)/(thickness in cm).
 (b) From Table 2.2.
 (c) Calculated as the product of columns 5 and 6, by radionuclide.
 (d) Complete this column only for the Residential/Home Garden scenario. The scenario dose factors are calculated as Column 7a plus Column 7b.

Form G - Depth Correction Worksheet

1. Facility: 117 D - Stratum B, Agricultural Scenario
2. Preparer's Name: G. F. Piepel
3. Date Prepared: 6/26/86
4. Depth of Clean overburden 6.7 m.
5. Calculate depth correction factor (DCF) as
 $DCF = e^{-0.92(\text{Depth})} = \underline{2.1E-3}$.

If Depth > 10 meters DCF = 0.0.

6. Radionuclide	7. Ingestion DF (rem/pCi)(a)	8. Corrected Ingestion DF(rem/pCi)(b)
<u>Pu-239</u>	<u>1.1E-14</u>	<u>2.3E-17</u>
<u>H-3</u>	<u>3.1E-14</u>	<u>6.5E-17</u>
<u>C-14</u>	<u>3.1E-14</u>	<u>6.5E-17</u>
<u>Co-160</u>	<u>1.0E-14</u>	<u>2.1E-17</u>
<u>Sr-90</u>	<u>2.7E-12</u>	<u>5.7E-15</u>
<u>Cs-137</u>	<u>1.0E-14</u>	<u>2.1E-17</u>
<u>Eu-152</u>	<u>1.7E-14</u>	<u>3.6E-17</u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>

- (a) From Area Correction or Availability Worksheet.
 (b) Calculated as Item 5 times Column 7.

Form E--Dose Variance [Var($\hat{D}_h$)] Calculation Worksheet for Stratum h

1. Facility and Stratum: 117D - Stratum B - Agricultural Scenario

2. Preparer's Name: GF Piepel 3. Date Prepared: 6/27/86 4. Stratum Mass (m_h): 8.77E+5 g 5. Decay Time (t): 100 yr

6.*		k:				Ahjk#						10.■
Radio nucli de hj	7.* SD(dh _{hj} y _{hj})	8.* λ _{hj}	9.* λ _{hk}	hk:	2	3	4	5	6	7	8	d _{hj} Σd _{hk} A _{hjk}
					3H	14C	60Co	90Sr	137Cs	152Eu		
					5.6E-2	1.2E-4	1.3E-1	2.4E-2	2.3E-2	5.2E-2		
					6.5E-17	6.5E-17	2.1E-17	5.7E-15	2.1E-17	3.6E-17		
1	239Pu	9.2E-11	2.9E-5	2.3E-17	93.2 5	24.75 5	4966 2	134.48 13	2528.46 13	374.9 1	e-	2.46E-31
2	3H	3.5E-12	5.6E-2	6.5E-17		1735.75 5	11408.3 1	83.4 5	27217 5	861.34 1	e-	1.02E-32
3	14C	1.2E-9	1.2E-4	6.5E-17			4693 1	107.75 5	35538 5	1109.3 1	e-	2.60E-31
4	60Co	9.1E-15	1.3E-1	2.1E-17				5732 2	289966 2	7209 1	e-	8.49E-35
5	60Sr	3.2E-9	2.4E-2	5.7E-15					17981 7	432.8 1	e-	2.84E-30
6	137Cs	4.7E-10	2.3E-2	2.1E-17						21893 1	e-	9.15E-33
7	152Eu	0	5.2E-2	3.6E-17							e-	
8												

2-2-1.54E+12

Σd_{hj}Σd_{hk}A_{hjk} = 3.37E-30

$$1.19E-17 = \Sigma [SD(d_{hj}y_{hj})]^2$$

$$2m_h^2 = 1.54E+12$$

$$\Sigma d_{hj} \Sigma d_{hk} A_{hjk} = 3.37E-30$$

$$11. \text{Var}(\hat{D}_h) = \{ \Sigma [SD(d_{hj}y_{hj})]^2 \} + \{ 2m_h^2 \} \{ \Sigma d_{hj} \Sigma d_{hk} A_{hjk} \} = 1.19E-17 + 1.54E+12 \cdot 3.37E-30 = 1.7E-17 = \text{Var}(\hat{D}_h)$$

* Column 6 [Radionuclide j and k]--from Form A, Col. 4.
 Column 7 [SD($d_{hj}y_{hj}$)]--from Form D, Col. 9.
 Column 8 [λ_{hj} and λ_{hk}]--from Table 5.3.
 Column 9 [d_{hj} and d_{hk}]--from Form D, Col. 7.

$A_{hjk} = \frac{\hat{\sigma}_{hjk}}{n_{hjk}} e^{-\theta}$, where $\hat{\sigma}_{hjk}$ is from Form B, Col. 10,
 n_{hjk} is from Form B, Col. 11, and
 $\theta = (\lambda_{hj} + \lambda_{hk})t$.
 ■ Column 10--calculated as $d_{hj} \Sigma d_{hk} A_{hjk}$.

A.48

- [illegible]

8a. TOTAL =

- (a) From Form B - Source Term Worksheet (column 10 or 12), for $T = 100$ years.
- (b) From Form C - Form C - Availability, Form F - Area, Form G - Depth, or Form H - Mass Correction Worksheets, as appropriate.
- (c) Calculate as the product of Columns 5 and 7.
- (d) Calculate as the product of Columns 8 and 7.
- (e) Calculate as $(\text{Column } 8) \times z_{1-\alpha}$ (Column 9), where $z_{1-\alpha}$ is chosen to correspond with the desired confidence level from Table 5.5.
- (f) Calculate using Procedure 3.2.2, Columns 7, 9, and Columns 8, 9, 10, and 11 from Form A - Summary Statistics Worksheet on Form E (Dose Variance Calculation Worksheet).

Form I - Final ARCL Worksheet - Summed Over All Strata

1. Facility/Scenario: 115 D/DR and 117D - Residential-Home Garden Scenario

2. Preparer's Name: G. F. Piepel

3. Date Prepared: 6/27/86

4. Stratum or Scenario Considered(a)	5. Total Dose per Stratum or Scenario (rem/yr)(b)	6. Variance of the Dose(c) (rem/yr) ²
<u>115D/DR, Main</u>	<u>7.0E-5</u>	<u>1.0E-10</u>
<u>Floor and Basement</u>	<u> </u>	<u> </u>
<u>in 20 x 10 x 5 m</u>	<u> </u>	<u> </u>
<u>Excavation</u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>
<u>115D/DR, Small</u>	<u>8.7E-5</u>	<u>1.4E-9</u>
<u>Portion of Dryer</u>	<u> </u>	<u> </u>
<u>Room #1</u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>
(TOTALS)	5a = <u>1.6E-4</u>	6a = <u>1.5E-9</u>

7. $(6a)^{1/2} = SO [\hat{D}_S(t)] =$ 3.9E-5

8. $5a + z^{(d)}_{1-\alpha} * 7 =$ 2.1E-4 (90 % UCL)

- (a) Note: Forms A, B, C, and D are completed for each Stratum considered.
 (b) From Item 8b in Form D for each Stratum considered.
 (c) From Item 11 in Form D for each Stratum considered.
 (d) Obtained from Table 5.5.

Form I - Final ARCL Worksheet - Summed Over All Strata

1. Facility/Scenario: 115 D/DR and 117D - Agricultural Scenario

2. Preparer's Name: G. F. Piepel

3. Date Prepared: 6/27/86

4. Stratum or Scenario Considered(a)	5. Total Dose per Stratum or Scenario (rem/yr)(b)	6. Variance of the Dose(c) (rem/yr) ²
<u>115D/DR, Balance</u>	<u>2.8E-6</u>	<u>8.9E-13</u>
<u>of Main Floor</u>		
<u>and Basement</u>		
<u>115D/DR-D Tunnel</u>	<u>1.5E-7</u>	<u>1.0E-11</u>
<u>115D/DR-DR Tunnel</u>	<u>1.6E-7</u>	<u>1.4E-13</u>
<u>117D-Stratum A</u>	<u>1.9E-7</u>	<u>3.3E-15</u>
<u>117D-Stratum B</u>	<u>1.0E-8</u>	<u>1.7E-17</u>
<u>Room #1</u>		
(TOTALS)	5a = <u>6.5E-6</u>	6a = <u>1.1E-11</u>
7. $(6a)^{1/2} = SD [\hat{\theta}_S(t)] =$ <u>3.3E-6</u>		
8. $5a + z_{1-\alpha}^{(d)} * 7 =$ <u>1.1E-5</u>		(<u>90</u> % UCL)

- (a) Note: Forms A, B, C, and D are completed for each Stratum considered.
 (b) From Item 8b in Form D for each Stratum considered.
 (c) From Item 11 in Form D for each Stratum considered.
 (d) Obtained from Table 5.5.

Form I - Final ARCL Worksheet - Summed Over All Strata

1. Facility/Scenario: 115 D/DR and 117D - Residential-Home Garden and Agricultural Scenario

2. Preparer's Name: G. F. Piepel

3. Date Prepared: 6/27/86

4. Stratum or Scenario Considered(a)	5. Total Dose per Stratum or Scenario (rem/yr)(b)	6. Variance of the Dose(c) (rem/yr) ²
<u>Res.-Home Garden</u>	<u>1.6E-4</u>	<u>1.5E-9</u>
<u>Agricultural</u>	<u>6.5E-6</u>	<u>1.1E-11</u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>
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<u> </u>	<u> </u>	<u> </u>
(TOTALS)	5a = <u>1.7E-4</u>	6a = <u>1.5E-9</u>

7. $(6a)^{1/2} = SD [\hat{D}_S(t)] =$ 3.9E-5

8. $5a + z^{(d)}_{1-\alpha} * 7 =$ 2.2E-4 (90 % UCL)

- (a) Note: Forms A, B, C, and D are completed for each Stratum considered.
 (b) From Item 8b in Form D for each Stratum considered.
 (c) From Item 11 in Form D for each Stratum considered.
 (d) Obtained from Table 5.5.

Form A - Summary Statistics Table - Estimated Means, Standard Deviations, and Covariances per Stratum

1. Facility and Stratum: 115 D/DR Building/Pipes

2. Preparer's Name: B. A. Napier

3. Date Prepared: 6/25/86

4. Radionuclide Considered	5. Mean $\bar{x}_{hi}$ (pCi/g)	6. S. D. σ_{hi} (pCi/g)	7. n_i	8. Nuclide i(a)	9. Nuclide j(a)	10. Covariance $(\sigma_{hij})(a)$	11. Number of Samples for which both Nuclides are Analyzed (n_{hij})
H-3	26	38.	5	C-14	Sr-90	-8.38	1
C-14	1.2	1.8	5	C-14	Co-60	-48.94	3
Sr-90	1.2	0.71	10	C-14	Cs-137	-212.	5
Co-60	11.	19.	11	C-14	Eu-152	-8.34	4
Cs-137	300.	190.	20	C-14	Eu-154	-1.67	5
Eu-152	3.3	1.6	18	C-14	Pu-239	-8.83	4
Eu-154	2.3	1.8	18	Sr-90	Co-60	-4.8 x 10 ⁻⁴	2
α /Pu-239	1.6	1.6	10	Sr-90	Cs-137	16.7	10
				Sr-90	Eu-152	-8.45	9
				Sr-90	Eu-154	-8.599	9
				Sr-90	Pu-239	-8.18	1
				Co-60	Cs-137	-990.	11
				Co-60	Eu-152	10.3	10
				Co-60	Eu-154	11.3	10
				Co-60	Pu-239	38.8	8
				Cs-137	Eu-152	-62.2	18
				Cs-137	Eu-154	45.8	18
				Cs-137	Pu-239	-79.1	10
				Eu-152	Eu-154	8.558	18
				Eu-152	Pu-239	8.883	9
				Eu-154	Pu-239	8.678	9

(a) Note: Columns 8-11 will be much longer than 4-7.

Comments:

(b) H-3 can be ignored for Resource Recycle Scenario since DF = 0.

Form B - Source-Term Worksheet

1. Facility/Scenario and Stratum: 115 D/DR Building/Pipes
2. Preparer's Name: B. A. Napier
3. Date Prepared: 8/25/88
4. Stratum Mass^(a) (grams): 8.26E+7g (a)
5. Number of Years of Decay: 100 (t)

6. Radionuclide Considered	7. Mean Concentration in Stratum, $\bar{x}_{hi}$ (pCi/g) (b)	8. Stratum Concentration S. D., b_i (pCi/g) (b)	9. Number of Samples in Stratum n_{hi} (b)	10. Source Term y_{hi} (d)	11. S. D. y_{hi} (d)	12. Decayed Source Term (pCi) (e)	13. Decayed S. D. y_{hi} (f)
C-14	1.2	1.0	5	9.9E+7	5.9E+7	9.8E+7	5.8E+7
Sr-90	1.2	0.71	10	9.9E+7	1.9E+7	9.0E+8	1.7E+8
Co-60	11.	19.	11	0.1E+8	4.7E+8	2.1E+3	1.1E+3
Cs-137	300.	190.	20	2.5E+10	3.5E+9	2.5E+9	3.5E+8
Eu-152	3.3	1.6	10	2.7E+8	2.9E+7	1.6E+8	1.0E+6
Eu-164	2.3	1.0	10	1.9E+8	3.5E+7	3.9E+4	7.10+3
Pu-239	1.5	1.5	10	1.2E+8	3.9E+7	1.2E+8	3.9E+7

(a) Strata are defined to consist only of contaminated material.

(b) Copy from Form A - Summary Statistics Table.

(c) Calculate as Item 4 times Item 7.

(d) S. D. (y_{hi}) = $\left(\frac{s_{hi}^2}{n_{hi}} \right)^{1/2}$

(e) Calculate as Item 10 times $e^{-\lambda_i t}$ where λ_i = decay constant for radionuclide i (from Table 5.2.3) and t = number of years of decay (Item 5).

(f) Calculate as Item 11 times $e^{-\lambda_i t}$ where λ_i = decay constant for radionuclide i (from Table 5.2.3) and t = number of years of decay (Item 5).

Form H - Mass Correction Worksheet

1. Facility: 115 D/DR Building, Pipes
2. Preparer's Name: B. A. Napier
3. Date Prepared: 6/25/86
4. Mass of contaminated metal, 82.6 Mg.
5. Calculate mass correction factor (MCF) as

$$\text{MCF} = \text{Item 4}/200 = \underline{0.41}$$

6. <u>Radionuclide</u>	7. <u>Recycle DF (rem/yr/pCi) (a)</u>	<u>Adjusted DF (rem/yr/pCi) (b)</u>
<u>C-14</u>	<u>1.6E-7</u>	<u>6.6E-8</u>
<u>Sr-90</u>	<u>2.0E-3</u>	<u>8.3E-4</u>
<u>Co-60</u>	<u>2.1E-1</u>	<u>8.7E-2</u>
<u>Cs-137</u>	<u>5.0E-2</u>	<u>2.1E-2</u>
<u>Eu-152</u>	<u>9.0E-2</u>	<u>3.7E-2</u>
<u>Eu-154</u>	<u>9.5E-2</u>	<u>3.9E-2</u>
<u>Pu-239</u>	<u>5.0E-1</u>	<u>2.1E-1</u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>
<u> </u>	<u> </u>	<u> </u>
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<u> </u>	<u> </u>	<u> </u>
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(a) From Table 2.2.

(b) Calculated as Item 5 times Column 7.

Form E--Dose Variance [Var($\hat{D}_h$)] Calculation Worksheet for Stratum h

1. Facility and Stratum: 115-D/DR Paper - Recycle Scenario

2. Preparer's Name: BA Napier 3. Date Prepared: 6/25/86 4. Stratum Mass (m_h): 8.26E+7g 5. Decay Time (t): 100 yr

6.*		7.*		8.*		9.*		10.*	
Radio nuclide j	SD($d_{hjy}h_j$)	λ_{hj}	d_{hj}	Radionuclide k	λ_{hk}	d_{hk}	A_{hjk}	$d_{hj}I_{dhk}A_{hjk}$	
1 ^{14}C	3.8E-12	1.2E-4	6.6E-20	^{90}Sr	2.4E-2	8.3E-16	$1.36E-16$	-8.66E-33	
2 ^{90}Sr	1.4E-9	2.4E-2	8.3E-16	^{60}Co	1.3E-1	8.7E-14	$1.36E-16$	-2.26E-30	
3 ^{60}Co	9.6E-11	1.3E-1	8.7E-14	^{137}Cs	2.3E-2	2.1E-14	$1.36E-16$	1.48E-31	
4 ^{137}Cs	7.4E-6	2.3E-2	2.1E-14	^{152}Eu	5.2E-2	3.7E-14	$1.36E-16$	-3.49E-27	
5 ^{152}Eu	5.9E-9	5.2E-2	3.7E-14	^{154}Eu	8.5E-2	3.9E-14	$1.36E-16$	2.86E-30	
6 ^{154}Eu	2.8E-10	8.5E-2	3.9E-14	^{239}Pu	2.9E-5	2.1E-13	$1.36E-16$	1.25E-31	
7 ^{239}Pu	8.2E-6	2.9E-5	2.1E-13						
8									

11. $Var(\hat{D}_h) = \{ \sum [SD(d_{hjy}h_j)]^2 \} + \{ 2m_h^2 \} \{ \sum d_{hj}I_{dhk}A_{hjk} \} = 1.22E-10 + 1.36E+16 + 3.49E-27 = 7.46E-11 = Var(\hat{D}_h)$

* Column 6 [Radionuclide j and k]--from Form A, Col. 4.
 Column 7 [SD($d_{hjy}h_j$)]--from Form D, Col. 9.
 Column 8 [λ_{hj} and λ_{hk}]--from Table 5.3.
 Column 9 [d_{hj} and d_{hk}]--from Form D, Col. 7.

* $A_{hjk} = \frac{\hat{\sigma}_{hjk}}{n_{hjk}} e^{-\theta}$, where $\hat{\sigma}_{hjk}$ is from Form B, Col. 10,
 n_{hjk} is from Form B, Col. 11, and
 $\theta = (\lambda_{hj} + \lambda_{hk})t$.
 * Column 10--calculated as $d_{hj}I_{dhk}A_{hjk}$.

Form I - Final ARCL Worksheet - Summed Over All Strata

1. Facility/Scenario: 115 D/DR Building, Recycle Scenario, Pipes

2. Preparer's Name: B. A. Napier

3. Date Prepared: 6/26/86

4. Stratum or Scenario Considered(a)	5. Total Dose per Stratum or Scenario (rem/yr)(b)	6. Variance of the Dose(c) (rem/yr) ²
<u>Pipes</u>	<u>7.8E-5</u>	<u>7.5E-11</u>
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
(TOTALS)	5a = <u>7.8E-5</u>	6a = <u>7.5E-11</u>

7. $(6a)^{1/2} = SD [\hat{D}_S(t)] =$ 8.6E-6

8. $5a + z_{1-\alpha}^{(d)} * 7 =$ 8.9E-5 (90 % UCL)

- (a) Note: Forms A, B, C, and D are completed for each Stratum considered.
 (b) From Item 8b in Form D, for each Stratum considered.
 (c) From Item 11 in Form D, for each Stratum considered.
 (d) Obtained from Table 5.5.

APPENDIX B

COPIES OF THE WORKSHEETS AND TABLES USED
FOR PERFORMING ARCL CALCULATIONS

TABLE B.1. Worksheets Used for ARCL Analysis of Contaminated Facilities

<u>Form</u>	<u>Title</u>	<u>Figure Number in Text</u>
A	Summary Statistics Table--Estimated Means Standard Deviations, and Covariances per Stratum	Figure 5.3
B	Source-Term Worksheet	Figure 5.5
C	Availability Correction Worksheet	Figure 5.7
D	Stratum Dose Calculation Worksheet	Figure 5.8
E	Dose Variance Calculation Worksheet	Figure 5.9
F	Area Correction Worksheet	Figure 5.11
G	Depth Correction Worksheet	Figure 5.12
H	Mass Correction Worksheet	Figure 5.14
I	Final ARCL Worksheet	Figure 5.16

TABLE B.2. Decay Constants for Selected Radionuclides (yr⁻¹)

<u>Nuclide</u>	<u>Constant</u>	<u>Nuclide</u>	<u>Constant</u>
3H	5.6E-2	134Cs	3.4E-1
14C	1.2E-4	135Cs	3.4E-7
36Cl	2.2E-6	137Cs+D	2.3E-2
57Co	9.3E-1	144Ce	8.9E-1
60Co	1.3E-1	152Eu	5.2E-2
55Fe	2.6E-1	154Eu	8.5E-2
59Fe	5.6E+0	226Ra	4.3E-4
59Ni	8.7E-6	232Th	5.0E-4
63Ni	6.9E-3	235U+D	9.8E-10
90Sr+D	2.4E-2	238U+D	1.6E-10
93Mo	2.0E-4	237Np+D	3.2E-7
94Nb	3.5E-5	238Pu	7.9E-3
99Tc	3.2E-6	239Pu	2.9E-5
124Sb	4.2E+0	241Am	1.6E-3
125Sb+D	2.5E-1		

TABLE B.3. ARCL Scenario-Specific Dose Conversion Factors
in Effective Dose Equivalent rem/yr per pCi

	Scenario (rem/yr per pCi/Stratum)			
	Scenario 1	Scenario 2	Scenario 3	Scenario 4
³ H	--	2.4E-20	3.1E-14	2.1E-14
¹⁴ C	1.6E-19	4.1E-18	3.1E-14	2.1E-14
³⁶ Cl	1.4E-17	4.7E-16	1.5E-12	1.0E-12
⁵⁷ Co	1.4E-14	1.3E-13	7.3E-16	4.9E-16
⁶⁰ Co	2.1E-13	4.3E-12	1.0E-14	6.7E-15
⁵⁵ Fe	3.6E-18	5.9E-17	2.5E-17	1.7E-17
⁵⁹ Fe	7.5E-14	1.9E-12	2.3E-16	1.5E-16
⁵⁹ Ni	1.9E-17	7.3E-17	3.8E-16	2.5E-16
⁶³ Ni	1.9E-17	6.7E-19	1.0E-15	6.7E-16
⁹⁰ Sr+D	2.0E-15	1.1E-14	2.7E-12	1.8E-12
⁹³ Mo	1.4E-14	2.0E-16	1.9E-14	1.3E-14
⁹⁴ Nb	1.2E-13	2.6E-12	7.3E-15	4.9E-15
⁹⁹ Tc	1.7E-18	3.0E-17	3.4E-13	2.3E-13
¹²⁴ Sb	1.4E-13	3.5E-12	1.0E-14	6.7E-15
¹²⁵ Sb+D	3.4E-14	8.0E-13	3.4E-15	2.3E-15
¹³⁴ Cs	1.2E-13	2.9E-12	1.4E-14	9.3E-15
¹³⁵ Cs	6.5E-17	8.7E-18	1.4E-15	9.3E-16
¹³⁷ Cs+D	5.0E-14	1.0E-12	1.0E-14	6.7E-15
¹⁴⁴ Ce+D	4.4E-15	9.3E-14	1.1E-15	7.1E-16
¹⁵² Eu	9.0E-14	2.0E-12	1.7E-15	1.2E-15
¹⁵⁴ Eu	9.5E-14	2.1E-12	2.5E-15	1.6E-15
²²⁶ Ra	3.4E-14	3.5E-12	4.3E-11	2.8E-11
²³² Th	2.0E-14	4.9E-12	1.6E-12	1.1E-12
²³⁵ U+D	1.0E-14	1.7E-13	7.3E-15	4.9E-15
²³⁸ U+D	6.0E-15	6.7E-14	1.2E-16	8.0E-17
²³⁷ Np+D	7.5E-13	5.7E-13	1.4E-12	9.3E-13
²³⁸ Pu	4.7E-13	8.0E-14	9.3E-15	6.2E-15
²³⁹ Pu	5.0E-13	8.7E-14	1.1E-14	7.1E-15
²⁴¹ Am	4.5E-13	1.3E-13	1.0E-13	6.7E-14

TABLE B.4. Standard Normal Distribution Values $z_{1-\alpha}$
Such that $(1-\alpha)100\%$ of the Distribution
Lies to the Left of $z_{1-\alpha}$

<u>$(1-\alpha) 100\%$</u>	<u>$z_{1-\alpha}$</u>
99	2.33
97.5	1.96
95	1.645
90	1.282

Form A - Summary Statistics Table

1. Facility and Stratum:

2. Preparer's Name: _____

3. Date Prepared: _____

[illegible]

(a) Note: Columns 8-11 will be much longer than 4-7.

Comments:

Form B - Source-Term Worksheet

1. Facility/Scenario and Stratum:
2. Preparer's Name: _____
3. Date Prepared: _____
4. Stratum Mass ^(a) (grams): _____ (g)
5. Number of Years of Decay: _____ (t)

[illegible]

- (a) Strata are defined to consist only of contaminated material.
- (b) Copy from Form A - Summary Statistics Table (Table 5.2.1).
- (c) Calculate as Item 4 times Item 7.

$$(d) \text{ S. D. } (y_{hi}) = \left(\frac{\sum h^2}{n} \right)^{1/2}$$

- (e) Calculate as Item 10 times $e^{-\lambda_i t}$ where λ_i = decay constant for radionuclide i (from Table 5.2.3) and t = number of years of decay (Item 5).
- (f) Calculate as Item 11 times $e^{-\lambda_i t}$ where λ_i = decay constant for radionuclide i (from Table 5.2.3) and t = number of years of decay (Item 5).

...the fact that the *Journal of the American Medical Association* is the only journal in the world to publish a special section on the ethics of medicine.

- [illegible]

- (a) RF = 1.0 for radionuclides not contained in activated metals.
RF = 2t (corrosion rate cm/yr)/(thickness in cm).
(b) From Table 2.2.
(c) Calculated as the product of columns 5 and 6, by radionuclide.
(d) Complete this column only for the Residential/Home Garden scenario. The scenario dose factors are calculated as Column 7a plus Column 7b.

B.8

- [illegible]

11. Variance of Estimated Dose (rem/yr)^{2(f)}:

- (a) From Form B - Source Term Worksheet (column 10 or 12), for $T = \underline{\hspace{2cm}}$ years.
- (b) From Form C - Availability, Form F - Area, Form G - Depth, or Form H - Mass Correction Worksheets, as appropriate.
- (c) Calculate as the product of Columns 5 and 7.
- (d) Calculate as the product of Columns 6 and 7.
- (e) Calculate as $(\text{Column 8}) \times z_{1-\alpha}$ (Column 9), where $z_{1-\alpha}$ is chosen to correspond with the desired confidence level from Table 5.5.
- (f) Calculate using Procedure 3.2.2, Columns 7, 9, and Columns 8, 9, 10, and 11 from Form A - Summary Statistics Worksheet on Form E (Dose Variance Calculation Worksheet).

Form E--Dose Variance [Var($\hat{D}_h$)] Calculation Worksheet for Stratum h

1. Facility and Stratum: _____

2. Preparer's Name: _____ 3. Date Prepared: _____ 4. Stratum Mass (m_h): _____ g 5. Decay Time (t): _____ yr

6.* Radio nucli de h _j				A _{hjk} #								10.■ d _h Σd _h kA _{hjk}
7.* SD(d _h jy _h j)				8.* λ _h j				9.* d _h j\ d _h k				
Radionuclide h _k				k:	2	3	4	5	6	7	8	
1				—e ⁻	—e ⁻	—e ⁻	—e ⁻	—e ⁻	—e ⁻	—e ⁻	—e ⁻	
2					—e ⁻	—e ⁻	—e ⁻	—e ⁻	—e ⁻	—e ⁻	—e ⁻	
3						—e ⁻	—e ⁻	—e ⁻	—e ⁻	—e ⁻	—e ⁻	
4							—e ⁻	—e ⁻	—e ⁻	—e ⁻	—e ⁻	
5								—e ⁻	—e ⁻	—e ⁻	—e ⁻	
6									—e ⁻	—e ⁻	—e ⁻	
7										—e ⁻	—e ⁻	
8											—e ⁻	

2

Σd_hΣd_hkA_{hjk} =

$\Sigma [SD(d_{hj}y_{hj})]^2 =$ _____
 $2m_h^2 =$ _____
 $\Sigma d_{hj} \Sigma d_{hk} A_{hjk} =$ _____

11. $Var(\hat{D}_h) = \{ \Sigma [SD(d_{hj}y_{hj})]^2 \} + \{ 2m_h^2 \} \{ \Sigma d_{hj} \Sigma d_{hk} A_{hjk} \} =$ _____ $+$ _____ $=$ _____ $= Var(\hat{D}_h)$

* Column 6 [Radionuclide j and k]--from Form A, Col. 4.
 Column 7 [SD(d<sub>h_jy_{h_j})]--from Form D, Col. 9.
 Column 8 [λ_{h_j} and λ_{h_k}]--from Table 5.3.
 Column 9 [d_{h_j} and d_{h_k}]--from Form D, Col. 7.</sub>

$A_{hjk} = \frac{\hat{\sigma}_{hjk}}{n_{hjk}} e^{-\theta}$, where $\hat{\sigma}_{hjk}$ is from Form B, Col. 10,
 n_{hjk} is from Form B, Col. 11, and
 $\theta = (\lambda_{h_j} + \lambda_{h_k})t$.
 ■ Column 10--calculated as d_{hj}Σd_{hk}A_{hjk}.

B.9

Form F - Area Correction Worksheet

1. Facility:
2. Preparer's Name:
3. Date Prepared:
4. Area of Contamination Within 1-ha Site _____ m² = _____ ha
5. Correction Factor = $0.1 + 9 \text{ (Item 4)}$ = _____.

6. Radionuclide

7. Ingestion
DF (rem/yr/pCi) (a)

8. Corrected Ingestion
DF (rem/yr/pCi) (b)

This image shows a single sheet of white paper with horizontal blue or grey ruling lines. The lines are evenly spaced and run across the width of the page. There are no margins, text, or other markings on the paper.

- (a) From Availability Correction Worksheet.
(b) Calculated as Item 5 times Column 7.

Form G. Depth Correction Worksheet

1. Facility:
2. Preparer's Name:
3. Date Prepared:
4. Depth of Clean overburden _____ m.
5. Calculate depth correction factor (DCF) as

$$DCF = e^{-0.92(\text{Depth})} = \underline{\hspace{2cm}}.$$

If Depth > 10 meters DCF = 0.0.

6. <u>Radionuclide</u>	7. <u>Ingestion</u> <u>OF(rem/yr/pCi) (a)</u>	8. <u>Corrected Ingestion</u> <u>DF(rem/yr/pCi) (b)</u>
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____

- (a) From Form F - Area Correction or Form C - Availability Worksheet.
(b) Calculated as Item 5 times Column 7.

Form H - Mass Correction Worksheet

1. Facility:
2. Preparer's Name:
3. Date Prepared:
4. Mass of contaminated metal, _____ Mg.
5. Calculate mass correction factor (MCF) as

$$\text{MCF} = \text{Item 4} / 200 = \underline{\hspace{2cm}}$$

6. <u>Radionuclide</u>	7. <u>Recycle</u> <u>DF (rem/yr/pCi) (a)</u>	<u>Adjusted</u> <u>DF (rem/yr/Ci) (b)</u>
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____

(a) From Table 2.2.

(b) Calculated as Item 5 times Column 7.

Form I - Final ARCL Worksheet - Summed Over All Strata

1. Facility/Scenario: _____

2. Preparer's Name: _____

3. Date Prepared: _____

4. Stratum or Scenario Considered(a)	5. Total Dose per Stratum or Sce- nario (rem/yr)(b)	6. Variance of the Dose(c) (rem/yr) ²
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____
_____	_____	_____

(TOTALS)

5a = _____

6a = _____

7. $(6a)^{1/2} = SD [\hat{\theta}_S(t)] =$ _____

8. $5a + z^{(d)} * 7 =$ _____ (_____ % UCL)

(a) Note: Forms A, B, C, and D are completed for each Stratum considered.

(b) From Item 8b in Form D for each Stratum considered.

(c) Form D for each Stratum considered.

(d) Obtained from Table 5.5.

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