AR TARGET SHEET

The following document was too large to scan as one unit, therefore, it has been broken down into sections.

EDMC#:	0000003
SECTION:	4 OF 11

DOCUMENT #:	DOE/EIS-0113
TITLE:	Final EIS Disposal of Hanford Defense High-Level, Transuranic and Tank Wastes

APPENDIX F

METHOD FOR CALCULATING RADIATION DOSE

APPENDIX F

METHOD FOR CALCULATING RADIATION DOSE[a]

The radiological impacts associated with waste disposal operations, and the potential long-term hazards from disposed waste or continued storage, are presented in Chapter 5, Appendix H, and Appendix R in terms of calculated radiation doses to members of the general public.[b] The doses are based on the radionuclide releases summarized in Appendices B, H, and Q and on the environmental behavior of the radionuclides once they are released. This appendix contains details of the assumptions, models, parameters, and data required for calculation of environmental transport and human dose. Information is also provided in this appendix describing the computer codes used and their relationships to internationally accepted models.

F.1 INTRODUCTION

F.1.1 Doses During the Operational Period (Predisposal Activities)

Two groups of potentially exposed individuals are considered in dose calculations for the period in which the waste management operations or predisposal activities occur: 1) occupationally exposed workers and 2) offsite population. Doses to occupationally exposed workers would usually result from direct irradiation by concentrated sources, while members of the general public might be exposed to very dilute concentrations of radionuclides in the environment. Doses to the public resulting from transportation of wastes off site are discussed separately (see Appendix I). Radiation doses to miners and the public (principally from naturally occurring radon) from construction of the geologic repository are not incurred as a result of the defense waste disposal alternatives and are therefore not presented in this EIS except as part of the cumulative impacts in Section 5.1.4.

F.1.1.1 Occupational Dose

To calculate occupational doses from external exposure, it is necessary to compute an exposure rate, determine a quality factor of the radiation present, and estimate the amount of time each worker actually spends in the radiation field during various phases of the operation.

The operations and facilities at Hanford associated with the various alternatives are, in general, at a conceptual stage of development. Therefore they were not used directly to

(a) In accordance with common practice, the term "dose," when applied to individuals and populations, is used for brevity in this report instead of the more precise term "dose equivalent" as defined by the International Commission on Radiation Units and Measurements (ICRU 1980).

(b) The doses included in this EIS are based on the calculational methods of ICRP-2 rather than ICRP 26/30 because DOE Order 5480 for environmental exposure and the EPA guidance for radiation protection had not been issued by the time the Draft Environmental Impact Statement was published. The time required to change the doses would have significantly delayed the issuance of this final statement.

develop estimates of exposure times and dose rates. Instead, extrapolations of historical worker exposure data were used. Historically, operations in Hanford facilities handling radioactive materials, over a wide range of activities involving many man-years of radiation work, have resulted in an average annual external dose of about 0.5 man-rem/man-year to the exposed workers. This average value has been maintained by facility design features, operational philosophy, and administrative controls. It is expected that this average dose rate will be maintained in future disposal operations. Therefore occupational doses have been estimated using this average rate and the projected man-years needed to implement each alternative.

F.1.1.2 Public Dose

Radiation doses to the general public during the operational period are possible only if radionuclides are released and reach areas outside the Hanford Site boundaries. Radionuclide release rates have ~~been estimated for~~ each process step necessary for handling the wastes in each alternative. These atmospheric release rates are summarized in Appendix B for routine releases and in Appendix H for potential accidents. No release of radionuclides directly to surface waters is postulated for waste management operations.

There are two general types of radionuclide release to the environment: 1) controlled, low-level releases that continue for relatively long periods, such as occur during normal operation of nuclear facilities, and 2) abrupt, accidental releases. These two types combine three basic scenarios for public exposure to radionuclides:

- acute releases to the atmosphere
- chronic (routine) releases to the atmosphere
- exposure to residual contamination.

There are many possible exposure pathways for each of the three basic exposure scenarios, as illustrated in Figure F.1. For example, in an acute release to the atmosphere, a member of the public may be irradiated by passing clouds of radionuclides, he may inhale some, or some may deposit on the ground and plants around his home, resulting in a source of long-term exposure from a short-term release. For chronic releases, air submersion and inhalation are continuing pathways, and deposition on the ground and plants may accumulate to relatively high levels for very small, but continuous, releases.

F.1.2 Doses During the Postoperational Period

The doses calculated for members of the offsite public during implementation of waste disposal are based on relatively well characterized data on process emissions, population distribution, and regional crop production. However, for future dose projections, each of these factors becomes more uncertain and a slightly revised modeling method is required. Rather than concentrating only on atmospheric dispersion of released radionuclides, long-term analyses also consider possible transport of radionuclides through groundwater and via surface water, as well as possible intrusion directly into the waste by individuals.

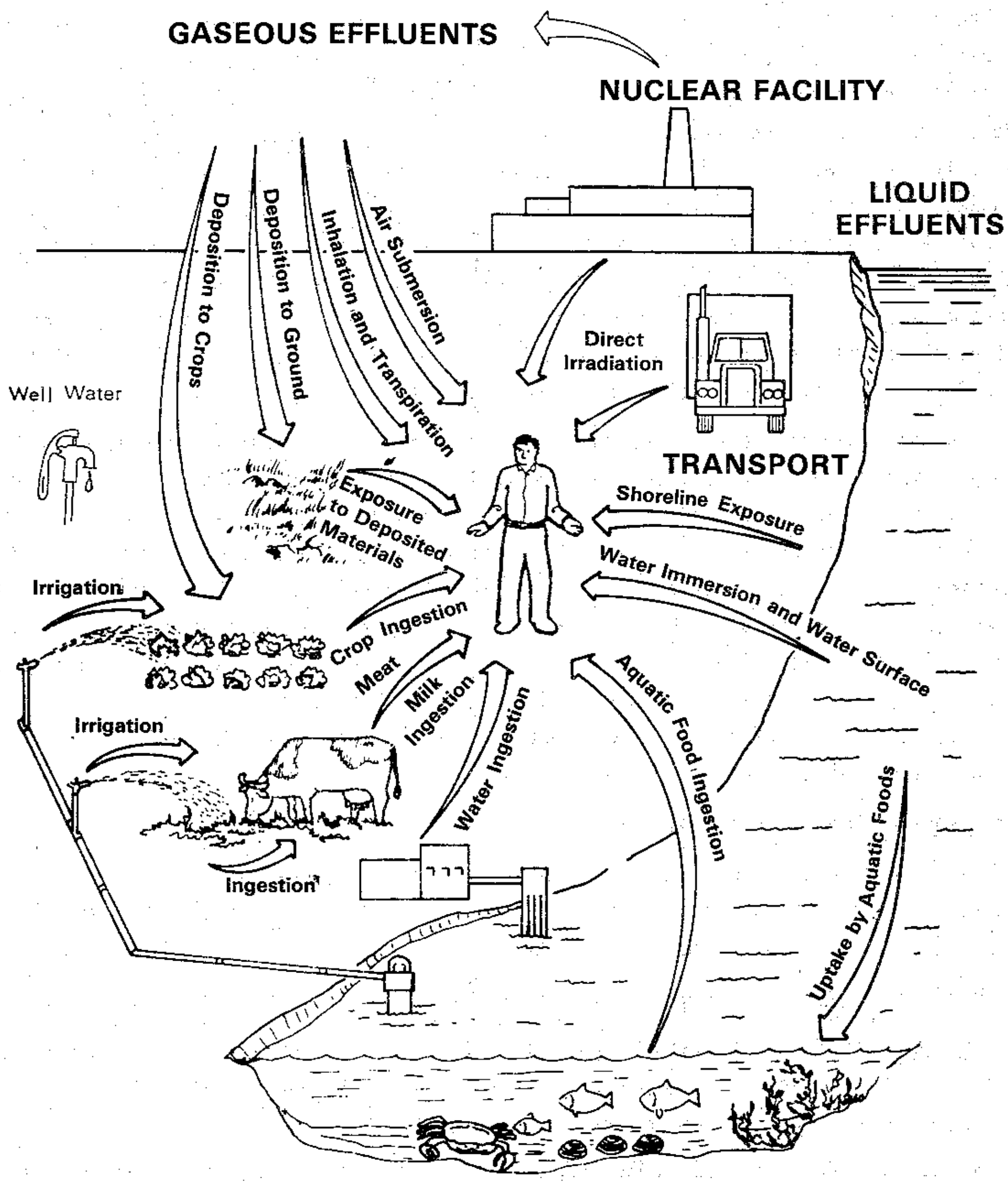


FIGURE F.1. Potential Environmental Exposure Pathways

For the scenarios that release radionuclides to the atmosphere, the pathways considered for long-term analysis are the same as those described for the short term. In addition, scenarios resulting in chronic release to surface waters are analyzed.

A different method is used for exposure scenarios for which a total population is small or cannot be determined. Many possible modes of exposure can be postulated for individuals that would result in minimal impact to the rest of society. Most of these are so-called "intruder" scenarios and are those that involve individuals intruding into a waste site. The doses calculated are maximum annual doses to an individual.

The dominant transport mechanism for radionuclides disposed of in the soil is groundwater leaching and transport. Specific radionuclides interact differently with Hanford soils. Thus site-specific modeling of the groundwater flow through the local aquifers must be done to determine times and concentrations of releases to the environment. Groundwater modeling is usually performed in two steps: 1) groundwater flow models are used to determine the groundwater potentials, flow paths, and travel times, and 2) contaminant transport models are then applied to simulate mass transport and geochemical interactions. Depending on the level of detail required, computer codes for groundwater modeling can be run for one-, two-, or three-dimensional simulations. (Appendix P discusses groundwater transport-modeling and limitations.) Groundwater models can be used to generate values for either radionuclide seepage to the Columbia River or contamination levels in well water.

A set of computer programs and standardized data have been established for use in performing radiation dose calculations for operational releases for Hanford (McCormack et al. 1984). Similar codes exist for long-term releases and for intruder scenarios. The various computer programs described below, used to assess radiation dose, use a consistent set of assumptions to calculate dose from sources both internal and external to the body. External sources include contaminated air, water and surfaces. Internal sources result from ingestion or inhalation of radionuclides. For all sources, doses may be calculated for various commitment periods. In all cases, resultant doses are presented for the adult man as defined in Publication 23 of the International Commission on Radiological Protection (ICRP 1975). As a rule, dose as a function of age is not considered.

F.1.3 Types of Dose Used in This EIS

Radiation dose is proportional to the quantity of energy deposited per unit mass of irradiated tissue. Definitions of length of time of exposure and length of time following exposure are what determine the format of the dose reported.

There are five basic categories of public radiation doses that could be calculated.

1. One-year dose from one year of exposure (external plus internal). This is the dose currently used for comparison with occupational exposure standards and the one originally used for comparison with public standards.

   The one-year dose is used at Hanford as a measure of potential short-term impact from accidental releases during waste management operations.

2. Committed dose from one-year external exposure plus extended internal dose accumulated as a result of a one-year intake (ingestion plus inhalation). Normally, a 50- or 70-year dose-commitment period is used. This dose is the one currently being used by most of those who calculate public doses, and is the one used for occupational record-keeping in NRC (1982b).

   The committed dose is used as a measure of the potential longer-term impact of accidents and routine releases.

3. Accumulated dose from a lifetime (50 or 70 years) of external exposure plus intake via ingestion and inhalation. This includes the effects of radionuclide accumulation or decay in the environment during the exposure period. This can also be construed as the lifetime committed dose from continuous exposure. This dose relates most closely to the risk of health effects from lifetime radiation exposure.

   The accumulated dose is used as a measure of the total impact of any operation which results in chronic releases over a period of several years' duration, or from long-lasting, relatively constant, groundwater contamination.

4. Maximum annual dose during a lifetime (50 to 70 years). This dose is calculated for each year of exposure, accounting for each year's external exposure plus the internal dose from nuclides taken in during the year of interest and all previous years. The maximum annual dose is identified by inspection for each organ. This type corresponds most closely to the existing guides for occupational and public exposure which contain standards for annual radiation dose.

   The maximum annual dose is calculated for scenarios of human intrusion or long-term occasional exposure to disposed wastes.

5. Integrated or cumulative dose from very long-term population exposure (up to 10,000 years). This dose is calculated as a sum of lifetime accumulated doses to populations over long periods. It gives a measure of the total impact of a very long, time-dependent release of radionuclides to the environment.

   The integrated population dose is used in conjunction with long-term groundwater and surface-water scenarios.

Each of these types of radiation dose is used in appropriate portions of Section 5, Appendix H, and Appendix R of this EIS. A simplified table describing the type of dose used in each descriptive scenario is given as Table F.1.

F.1.4 External Dosimetry

For calculating external dose factors, the penetrating power of the radiation emitted determines whether it contributes to skin dose only or to both skin and total-body doses. The beta and gamma radiation that can penetrate more than 7×10^{-3} cm of tissue is considered to contribute to skin dose; radiation which can penetrate 5 cm of tissue is considered to contribute to total-body dose (and dose to internal organs). The dose factors for most external exposures are derived assuming that the contaminated medium is infinite compared with the range of the emitted radiations. Under this assumption, the energy emitted per gram of medium equals the energy absorbed per gram. Corrections are applied for differences in energy absorption between tissue and air or water, physical geometry of the specific exposure situation, and the conversion from MeV per disintegration per gram to rem.

Concentrated sources of radiation, such as buried wastes, are modeled using the shielding code ISOSHLD (Engel, Greenborg and Hendrickson 1966). ISOSHLD is a computer code that can be used to perform gamma-ray shielding calculations for isotope sources in a wide variety

TABLE F.1. Types of Radiation Dose Used in the Various Scenarios of this EIS

Scenario (Location in EIS)	One-Year Dose	Committed Dose	Accumulated Dose	Maximum Annual Dose	Integrated Population Dose
Occupational (Chapter 5)	X				
Occupational Accidents (Chapter 5, Appendix H)	X	X			
Routine Releases (Chapter 5)		X	X		
Intrusion into Waste (Appendix R)					
- Drilling			X	X	
- Habitation			X	X	
- Excavation			X	X	
Groundwater Transport (Appendix R)					
- Drinking-Water Well			X	X	X
- Irrigation Well			X	X	X
- To River			X	X	X

of source and shield configurations. Attenuation calculations are performed by point kernel integration; for most geometries this is accomplished using Simpson's rule for numerical integration. Buildup factors are calculated by the code based on 1) the number of mean free paths of material between the source and detector points, 2) the effective atomic number of a particular shield region (the last region unless otherwise chosen), and 3) the point isotropic buildup data available as Taylor coefficients. This procedure allows calculation of geometry-specific dose factors.

F.1.5 Internal Dosimetry

The dose model used is derived from that originally given by ICRP 1959 in Publication 2 for body burden and maximum permissible concentration. Effective decay energies for radionuclides are calculated using the ICRP model. This model is based on the assumption that the entire quantity of a given radionuclide is located at the center of a spherical organ with an appropriate effective radius (Soldat 1976). Metabolic parameters for the standard man are used (ICRP 1975). Some of the parameters are updated from later ICRP publications.

Several radionuclides are handled as special cases. For the radionuclides ^{3}H and ^{14}C, the accumulated dose for the organs total-body and bone are calculated as above. Since these radionuclides distribute evenly in the rest of the body, the doses for all the other organs are set equal to that for total body. For isotopes of sodium, the doses to all organs, including bone, are set equal to the total-body dose.

The model for the gastrointestinal (GI) tract also differs. The GI tract--stomach, small intestine (SI), upper large intestine (ULI), and lower large intestine (LLI)--is modeled as a four-compartment system with a plug flow. Since there is no long-term storage or retention, the dose in any one year is equal to the dose commitment for that year. The portions of the GI tract are assumed to be irradiated from radionuclides uniformly distributed in the material passing through each compartment.

The internal distribution of radionuclides following inhalation adds a degree of complexity due to differing retention in the lungs. The model of the respiratory tract adopted by the Task Group on Lung Dynamics (ICRP 1966) forms the general basis for the mathematical models developed to calculate the dose from the inhalation of radionuclides.

F.1.5.1 Critical Groups

The doses calculated for this EIS are based on the metabolism of the "standard man." This mathematical representation of an average male worker obviously does not fit every individual in the general public. Actual doses depend partly on age- and sex-specific relationships between annual intakes and dose (e.g., body size) and partly on age-specific factors influencing annual intake (e.g., milk consumption in children). Further complications arise from general lifestyle considerations. The resulting variations are too numerous to attempt to calculate anything but average values, and to qualify the resulting conclusions. The "standard man" (adult male worker) parameters are the usual representation for these purposes.

Even if there were no differences with age in the uptake and retention of a radionuclide, the dose in a particular tissue per unit intake of the radionuclide would be greater in children than in adults because of the smaller masses of their organs and tissues. For the extreme case of a child in the first year of life, whose body mass at age 6 mo is about 7 kg (ICRP 1975), the committed dose equivalent in an organ or tissue per unit intake of a short-lived radionuclide emitting poorly penetrating radiations would be about ten times greater than for a 70-kg adult. As described by Adams (1981), this factor would be about two for intakes of long-lived radionuclides that are retained long in body tissues (e.g., ^{239}Pu) because the child grows during the prolonged irradiation. For radionuclides emitting penetrating photons, the modifying factors for body size are smaller, the committed dose per unit intake of a radionuclide being approximately inversely proportional to body mass$^{2/3}$ rather than body mass (Adams 1981). Although organ mass is not a constant proportion of body mass, and the shapes and relative positions of organs change with age, these differences will usually have only a small effect on the factors discussed above. Therefore, to allow for body size alone, committed dose equivalents per unit intake for young members of the public will be greater than those for workers by factors ranging from less than 2 up to 10, the actual value for any age depending not only on the mass of the individual but also on the types of radiation emitted by the radionuclide and its retention in body tissues.

Children can also have a very different metabolism from that of adults, taking up different fractions (often more) of a chemical substance from the blood into their organs and tissues and eliminating it at different rates (often more rapidly). For a radioisotope of a chemical element in the substance, uptake and retention by the organs and tissues of the body will additionally depend on its radioactive half-life. Relevant data are scarce but the following examples will serve to illustrate the nature of the problem.

From considerations of water balance, the mean life of water in the body is about 14 days for adults and 6 days for infants aged 6 mo (ICRP 1975), and that of the long-lived radionuclide tritium in the form of tritiated water will have similar values. Consequently, the committed dose equivalent to body tissues from unit intake of tritium as tritiated water will be only about four times greater for such infants than for adults, rather than the ten times greater factor derived above that would be expected on the basis of their differences in mass alone. Similarly, because of the more rapid turnover of the long-lived ^{137}Cs in people of smaller mass (Cryer and Baverstock 1972), the committed dose equivalent in body tissues from unit intake of the radionuclide is actually less for the 6-mo infant than for adults (Hoenes and Soldat 1977).

The mean life of iodine in the thyroid also increases with age, but this may be accompanied by a small decrease in the uptake into the gland from the blood (Medical Research Council 1975; UNSCEAR, 1977; Dunning and Schwarz 1981; Stather, Greenhalgh and Adams 1983). For the relatively short-lived radionuclide ^{131}I, differences in biological turnover are of little consequence because its rate of loss from the thyroid is dominated by radioactive decay and its mean life in that organ is therefore about the same at all ages. Thus the committed dose equivalent to the thyroid per unit intake of ^{131}I is about seven times greater for the infant aged 6 mo than for adults (Medical Research Council 1975), reflecting their approximately tenfold difference in thyroid mass. However, for the very long-lived ^{129}I, the more rapid biological turnover in young people tends to offset their smaller mass, and the committed dose equivalent to the thyroid per unit intake of ^{129}I for the 6-mo child is only about twice that for adults (UNSCEAR 1977).

Papworth and Vennart (1973) and Leggett, Eckerman and Williams (1982) have described how the uptake of strontium into bone and its retention there vary with age. Papworth and Vennart have given values for the committed dose equivalent in red bone marrow and on bone surfaces from unit intake of dietary ^{90}Sr and ^{89}Sr. For the longer-lived ^{90}Sr, the value for a 6-mo infant is about five times the adult value, but for the much shorter-lived ^{89}Sr this ratio lies between 20 and 40, the actual value depending on the model used for the dosimetry of the radionuclide in bone. There may be additional contributions to the committed dose equivalent from other organs and tissues for which the factors might be different.

The chemical form of the radionuclide can play a role in variation of dose. Compounds of the same radionuclide found in the environment or in food may be metabolized differently. The resultant changes in dose values must be considered very carefully. For example, increased absorption of a radionuclide from the gastrointestinal tract into the blood will decrease the committed dose equivalent to the lower part of the tract but will increase the doses in other tissues of the body. Other factors, such as particle size of airborne radionuclides, can also affect the value of dose calculated.

F.1.5.2 Other Dosimetry Systems

The dosimetry model recommended in ICRP-26 (ICRP 1977) and applied in the ICRP-30 (ICRP 1979) is based on more recent human metabolic parameters. The models for uptake and retention of radionuclides in body organs are more complex. The contribution to organ dose resulting from decay of radionuclides in other organs (crossfire) is also accounted for. Rather than report the individual organ doses, the concept of an "effective whole-body dose" (the sum of the product of each organ dose times its appropriate weighting factor) is used. The effective whole-body dose is then used for comparison to a stochastic dose limit. The stochastic effective dose equivalent limit recommended for an individual in the general public, according to ICRP-26, is 500 mrem/yr. In addition, ICRP-26 states that when prolonged exposures are expected, the effective dose equivalent should be limited to 100 mrem/yr. The weighting factors recommended by the ICRP are:

Gonads	0.25
Breast	0.15
Red bone marrow	0.12
Lung	0.12
Thyroid	0.03
Bone	0.03
Remainder	0.30

The dose results calculated for the residential/home garden scenario of Appendix R have been compared using ICRP-2 and ICRP-30 methods. While individual radionuclide results for organs may vary up or down by factors as great as 5 to 10, it can be shown that, for representative radionuclide mixtures, the calculated dose varies minimally. Because the dose is a function of ingestion, inhalation, and external exposure pathways, generalizations on total dose should not be made.

The "dose factors," rems per microcurie (μCi) ingested or inhaled, for nuclides of specific interest to this EIS are listed in Table F.2 for both the modified ICRP-2 (with the model of the Task Group on Lung Dynamics) and ICRP-30 dosimetry models. For simplicity, these are compared for an adult on the basis of a single intake followed by a 50-year commitment period. Lung translocation class assumed is also given (soluble, insoluble). For the nuclides of importance to the groundwater pathway (^{14}C, ^{99}Tc, ^{129}I), the old and new models produce critical-organ dose estimates within about 30% of each other. The older ICRP-2 "total body" differs from the weighted-organ ICRP-30 "whole body" in definition, so it is not surprising that these differ somewhat more (8 times less to 33 times more). In no case do the calculated differences affect the relative impacts of the alternatives considered. There is also a difference in the definition of bone, from the whole skeletal "bone" to the "bone surface." The ICRP-30 factors also use a quality factor for alpha particles twice that of ICRP-2.

TABLE F.2. Comparison of Radiation Dose Commitment Factors Calculated Using Modified ICRP-2 and ICRP-30 Methods, rem/μCi

	Ingestion			
	ICRP-2		ICRP-30	
Radionuclide[a]	Total Body	Organ	Whole Body	Organ
^{14}C	5.7×10^{-4}	Bone 2.8×10^{-3}	2.1×10^{-3}	All 2.1×10^{-3}
^{90}Sr	1.0	Bone 3.9	1.3×10^{-1}	Bone Surface 1.6
^{99}Tc	5.0×10^{-5}	LLI 6.1×10^{-3}	1.3×10^{-3}	LLI 4.1×10^{-3}
^{129}I	9.2×10^{-3}	Thyroid 7.2	2.8×10^{-1}	Thyroid 9.3
^{137}Cs	7.1×10^{-2}	Liver 1.1×10^{-1}	5.0×10^{-2}	Other 5.5×10^{-2}
^{239}Pu	2.6×10^{-2}	Bone 5.4×10^{-1}	4.3×10^{-1}	Bone Surface 7.8
^{241}Am	6.7×10^{-2}	Bone 1.8	2.2	Bone Surface 4.1×10^{1}

	Inhalation			
	ICRP-2[b]		ICRP-30	
Radionuclide	Total Body	Organ	Whole Body	Organ
^{14}C (as CO_2)	3.3×10^{-4}	Bone 1.7×10^{-3}	2.1×10^{-3}	All 2.1×10^{-3}
^{90}Sr (soluble)	1.7	Bone 2.6×10^{1}	2.3×10^{-1}	Bone Surface 2.7
^{99}Tc (soluble)	5.4×10^{-5}	LLI 9.9×10^{-4}	8.4×10^{-4}	Stomach 9.3×10^{-3}
^{129}I (soluble)	5.4×10^{-3}	Thyroid 4.2	1.8×10^{-1}	Thyroid 5.9
^{137}Cs (soluble)	4.2×10^{-2}	Liver 6.3×10^{-2}	3.2×10^{-2}	Other 3.5×10^{-2}
^{239}Pu (soluble)	7.5×10^{1}	Bone 1.6×10^{3}	5.1×10^{2}	Bone Surface 9.3×10^{3}
^{241}Am (soluble)	6.3×10^{1}	Bone 1.6×10^{3}	5.2×10^{2}	Bone Surface 9.3×10^{3}

(a) All ICRP-2 values assume a soluble form of the radionuclide. For reported ICRP-30 values, the largest reported GI-tract absorption is used.
(b) Incorporates model of the Task Group on Lung Dynamics for lung doses, with ICRP-2 organ doses following translocation to the bloodstream.

F.2 ENVIRONMENTAL PATHWAY AND DOSIMETRY MODELS

The doses caused by chronic and accidental releases of gaseous and liquid effluents from the facilities and processes investigated in this study were estimated using several calculational models. The models used are of the Concentration Factor type described in ICRP Publication 29 (1978). The models and parameters used were selected to give a realistic but conservative appraisal. Each model is generic and is put to specific application in the various specific computer codes. Site-specific parameters are used wherever possible.

The fundamental relationship for calculating radiation doses to people from any radionuclide exposure pathway is given in Equation (F.1) (Soldat, Robinson and Baker 1974):

$$R_{ipr} = C_{ip} U_p D_{ipr} \tag{F.1}$$

where R_{ipr} = the radiation dose equivalent or committed radiation dose equivalent from radionuclide i via exposure pathway p to organ r, rem

C_{ip} = concentration of radionuclide i in the media of exposure pathway p; for calculations involving airborne radionuclides, C_{ip} is replaced with the term χ_i, which represents the average airborne concentration of radionuclide i, pCi/m^3, pCi/L, or pCi/kg

U_p = usage parameter (exposure rate or intake rate) associated with exposure pathway p, hr/yr, L/yr, or kg/yr

D_{irp} = radiation dose equivalent factor or the committed dose equivalent factor for radionuclide i exposure pathway p and organ r to convert the concentration and usage parameters to the radiation dose equivalent or to the committed radiation dose equivalent, mrem/pCi.

An analysis of radiation doses from separate exposure pathways requires a determination of the radionuclide concentrations and exposure rate or intake rate associated with each exposure pathway. For external exposure, the concentration of radionuclides and the duration of exposure must be quantified. For ingestion of farm products grown on a contaminated site, the radionuclide concentration in separate food products must be determined by accounting for root transfer from soil, dry deposition from air onto surfaces of vegetation, or animal consumption of contaminated forage or feed. The annual diet for the exposed individual, the length of the growing season, and the holdup time between harvest and consumption must also be determined.

F.2.1 Air Submersion

Both photons and beta particles can contribute significantly to the external dose to skin. The beta dose contribution is calculated using a semi-infinite cloud model. This model can be used because the range of beta particles in air is short compared to the dimensions of plumes considered. The gamma dose calculation is more complicated because of the relatively low attenuation of photons in air. To properly determine the gamma contribution, it is necessary to perform a space integration over the plume volume.

The contribution of gamma radiation to total-body dose was estimated by calculating the tissue dose at 5-cm depth. An occupancy factor may be used to account for the fraction of the year a person is exposed to the cloud. Also a shielding factor may be employed to correct for any shielding by buildings or structures between the recipient and the cloud.

F.2.2 Inhalation Dose

The air concentrations were used along with the ventilation rate and dose factors to estimate the dose through the inhalation of radionuclides dispersed in the air.

The ventilation rate is the volume of air taken in by an individual per unit time. A value of 0.23 L/sec was used in this study (ICRP 1959).

The inhalation dose factor, given in units of rem/yr per Ci/yr intake, is dependent on the complex transport, retention, and elimination of radionuclides through the respiratory and gastrointestinal tracts. The model of the respiratory tract adopted by the task group on lung dynamics forms the general basis for the calculation of this dose factor (ICRP 1966).

F.2.3 Ground Contamination Dose

Radionuclides from the air may settle on the ground, where they can accumulate during the time of the release. These can be a source of radiation for an individual or population groups.

This dose is determined using the 1) air concentration, 2) deposition "velocity" of the radionuclides traveling to the surface from the air, 3) an exponential expression which accounts for the accumulation and radioactive decay of the radionuclide on the ground over a certain time period, 4) a dose factor, 5) an occupancy factor, and 6) an assumed geometry.

The deposition "velocity," given in terms of m/sec, is highly dependent on surface roughness, wind speed, and particle size. Based on many experimental studies, values of 0.001 m/sec for particles and 0.01 m/sec for iodine gas were selected for use in this report (AEC 1968).

The dose factor for the dose from ground irradiation is calculated by assuming that a receptor is 1 m above a large, nearly uniform, thin sheet of contamination (Soldat 1971; Fletcher and Dotson 1971). A factor of 0.5 to account for dose reduction due to ground surface roughness is also included in dose factors. These dose factors have units of rem/hr per pCi/m^2 of surface.

F.2.4 Ingestion of Food Crops

Food crops may become contaminated by deposition of radionuclides directly from the air, or from irrigation water upon the plant surfaces, or by radionuclides taken up from soil previously contaminated via air or water. Many factors must be considered when calculating doses via ingestion of these foods. These factors account for the movement of radionuclides from release to the receptor and form a complex sequence (Soldat 1971; Baker, Hoenes and Soldat 1976).

Equations used to calculate such doses are given in two parts: the first one accounts for direct deposition onto leaves and translocation to the edible parts of the plant, while the second accounts for long-term accumulation in the soil and root uptake.

The concentration of radioactive material in vegetation resulting from direct deposition onto plant foliage and uptake of radionuclides previously deposited in the soil is determined by Equation (F.2).

$$C_{iv} = \left[\frac{(d_i^a + d_i^I)r\ T_v\ (1 - \exp\ [^{\lambda}Ei\ ^te])}{y_v\ \lambda_{Ei}} + \frac{(d_i^a + d_i^I)\ f_t B_{vi}(1 - \exp\ [-\lambda_i t_b])}{p\ \lambda_i} + \frac{0.15 f_t C_{si} B_{vi}}{p} + \frac{f_w C_{ti} B_{vi}}{\rho}\right] \exp\ (-\lambda_i\ t_h) \qquad \text{(F.2)}$$

where C_{iv} = concentration of radionuclide i in the edible portion of the vegetation, pCi/kg

d_i^a = deposition rate or flux of radionuclide i, pCi/m^2-day

d_i^a = 86,400 χ_i V_{di}

86,400 = dimensional conversion factor, sec/day

χ_i = average air concentration of radionuclide i, pCi/m^3

V_{di} = deposition velocity of radionuclide i, m/sec

r = fraction of initially deposited material retained on the vegetation (dimensionless), taken to be 0.25

d_i^I = deposition rate or flux of radionuclides applied with irrigation water, pCi/m^2-day

d_i^I = C_{iw} I

C_{iw} = concentration of radionuclide i in the water used for irrigation, pCi/L

I = irrigation rate; the amount of water sprinkled on a unit area of field in one day, L/m^2-day

T_v = factor for translocation of externally deposited radionuclides to the edible parts of the vegetation (dimensionless). For simplicity, this parameter is assumed to be independent of the radionuclide and is assigned values of 1 for leafy vegetables and fresh forage and 0.1 for all other produce, including grain.

λ_i = radiological decay constant for radionuclide i, days^{-1}

λ_{Ei} = the effective removal constant for radionuclide i, days^{-1};

$\lambda_{Ei} = \lambda_i + \lambda_w$

λ_w = weathering removal constant for vegetation, days^{-1}; taken to be (0.693/14) days^{-1}

Y_v = vegetation yield, kg (wet weight)/m^2

B_{vi} = concentration factor for uptake of radionuclide i from the soil in vegetation v, pCi/kg (wet weight) per pCi/kg soil (dry)

t_b = time for buildup of radionuclides in the soil, days

t_e = time of exposure of above-ground vegetation to contamination during growing season, days

f_t = fraction of the roots in the plow layer of soil (dimensionless)

t_h = holdup time between harvest and food consumption, days

p = soil "surface density," kg (dry soil)/m^2; a value of 224 kg/m^2 is used assuming the contaminated ground is plowed to a depth of 15 cm (Napier, Kennedy and Soldat 1980)

C_{si} = concentration of radionuclide i available for plant uptake from the waste contained in the plow layer (top 15 cm of soil), pCi/m^3

0.15 = plow layer, m

f_w = fraction of the roots that penetrate the waste trenches (dimensionless)

C_{ti} = concentration of radionuclide i available for plant uptake from the subsurface waste zone, pCi/m^3

ρ = bulk soil density of subsurface waste material, kg/m^3.

The first term inside the brackets of Equation (F.2) relates to the concentration resulting from direct deposition of airborne material and irrigation on foliage during the growing season. The second term relates to the plant uptake from the soil and reflects the deposition from irrigation. The third and fourth terms account for uptake of waste material contained in the top 0.15 m of soil and below this layer, respectively. Specific values used for the parameters in Equation (F.2) are stored in data libraries associated with the code and are published in Napier et al. (1980).

The radionuclide concentration in animal products such as meat, milk, and eggs is dependent on the amount of contaminated forage or feed eaten by the animal. This concentration is described by Equation (F.3):

$$C_{ia} = S_{ia}\left[C_{if}\, Q_f + C_{iaw}\, Q_{aw}\right] \tag{F.3}$$

where C_{ia} = concentration of radionuclide i in the animal product, pCi/kg or pCi/L

S_{ia} = equilibrium transfer coefficient of radionuclide i from daily intake of the animal to the edible portion of the animal product, pCi/L (milk) per pCi/day or pCi/kg (animal product) per pCi/day

C_{if} = concentration of radionuclide i in feed or forage, pCi/kg; calculated from Equation (F.1)

Q_f = animal consumption rate of contaminated feed or forage, kg/day

C_{iaw} = concentration of radionuclide i in the water consumed by animals, pCi/L; assumed to be the same as the irrigation water, C_{iw}

Q_{aw} = consumption rate of the contaminated water by the animal, L/day.

Specific values of the parameters used in Equation (F.3) are given in Napier, Kennedy and Soldat (1980).

The nuclides ^3H and ^{14}C are treated as special cases in the calculations. The concentrations in the initial environmental media (air or water) are calculated on the basis of the specific activity of the nuclide in the naturally occurring stable element.

F.3 STANDARD HANFORD CALCULATIONAL METHODS

A set of computer programs has been developed at Hanford to calculate the dose consequences from all significant exposure pathways illustrated in Figure F.1, using the models described in Section F.2.

The evaluation of potential environmental radiation impacts is facilitated through the use of these computerized dose calculation programs. These are listed in Table F.3. Each program assesses a common set of standardized libraries which, to the extent they are

TABLE F.3. Computer Programs Used To Calculate Potential Radiation Doses from Releases During and After Waste Disposal

Program	Type of Dose	Reference
SUBDOSA	One-year air submersion dose from acute (finite cloud) or chronic (semi-infinite cloud) releases, individual and collective doses.	Strenge, Watson and Houston 1975
DACRIN	Individual and collective inhalation doses from chronic or acute releases, one-year doses, dose commitments, and accumulated doses.	Houston, Strenge and Watson 1974; Strenge 1975
PABLM	Individual and collective doses from contaminated farm products, from either air deposition or irrigation, one-year dose, dose commitment, and accumulated dose. Individual and collective doses from contaminated water and aquatic foods and aquatic recreation, one-year dose, dose commitment, and accumulated dose.	Napier, Kennedy and Soldat 1980
ALLDOS	Report generator using precalculated factors from SUBDOSA, DACRIN and PABLM. Simplifies repetitive calculations of individual and population doses.	Strenge et al. 1980
MAXI	A package of three programs to calculate individual maximum annual dose from residual radioactivity in the environment.	Napier et al. 1984
DITTY	Calculates 10,000-yr, integrated population doses from long-term releases, as from groundwater contamination.	Napier, Peloquin and Strenge 1985

available, contain Hanford-specific data. The programs and data libraries are maintained by the Hanford Dose Overview Program, with all revisions or updates documented (McCormack, Ramsdell and Napier 1984). An overall dose model QA plan is in place and followed for all code developments, revisions, and use.

F.3.1 Standard Hanford Computer Programs

The computer programs have been documented separately, and only a brief description of their application is given here.

F.3.1.1 DACRIN

This program (Houston, Strenge and Watson 1974; Strenge 1975) is used to analyze radiation doses from inhalation for Hanford operations. The program uses the model of the ICRP Task Group on Lung Dynamics (ICRP 1966) to predict radionuclide movements through the respiratory system and lung doses. Once radionuclides reach the blood stream, the doses to organs other than the lung are calculated using exponential retention functions (ICRP 1959).

DACRIN can also calculate atmospheric concentrations using the Gaussian, bivariate, normal distribution plume model. However, externally calculated dispersion factors may also be entered.

Doses calculated in DACRIN are dependent upon the values of the release time and dose time used as input. Therefore, the doses that can be calculated for both a maximally exposed individual (MI) and the regional population include a one-year dose, dose commitment and cumulative dose.

DACRIN is written in FORTRAN and typically uses about 80K of computer memory during an average 3-min run. The code is documented (Houston, Strenge and Watson 1974; Strenge 1975) and is available from PNL, the Radiation Shielding Information Center (RSIC) at Oak Ridge, and the National Energy Software Center (NESC) at Argonne.

F.3.1.2 SUBDOSA

This program (Strenge, Watson and Houston 1975) is used to calculate air submersion doses from accidental atmospheric releases of radionuclides. A space integration over the plume volume is performed. Dose results are reported for skin, male gonads, and total body. Corresponding tissue depths are 0.007, 1.0, and 5.0 cm, respectively. Doses are calculated for releases within each of several release time intervals. Up to six time intervals can be allowed, and separate radionuclide inventories and atmospheric dispersion conditions can be considered for each time interval. Normally, a one-year dose for both the maximally exposed individual and for the regional population are calculated.

SUBDOSA is written in FORTRAN and typically uses about 50K of computer memory during an average 1-min run. The code is documented (Strenge, Watson and Houston 1975) and is available from PNL or RSIC at Oak Ridge.

F.3.1.3 PABLM

The PABLM program (Napier, Kennedy and Soldat 1980) is used to calculate potential doses from environmental contamination pathways that include direct radiation from contaminated water, sediment, soil surfaces, and ingestion doses from contaminated drinking water, aquatic food products, terrestrial farm products and farm animal products. PABLM combines and enhances the pathway modeling capabilities of computer programs ARRRG and FOOD (Napier, Kennedy and Soldat 1980). It also can consider changing levels of environmental contamination with time from past or continuing deposition, and includes radioactive chain decay with daughter ingrowth. PABLM can be used to calculate dose commitments from one year of exposure and cumulative doses to either a maximally exposed individual or populations from multiple years of exposure. Some parameters included in the PABLM data libraries are specific to Hanford conditions.

PABLM is written in FORTRAN and typically uses about 90K of memory during an average 5-min run. The code is documented (Napier, Kennedy and Soldat 1980), and available from PNL, RSIC, or NESC. PABLM is widely used by the DOE sites and DOE offices involved with high-level waste repository siting. A sensitivity study was performed by Zach (1980) on a model similar to PABLM. The results are useful in providing insight concerning the dominant variables. This study, however, was only a method demonstration and not a test of the model validity.

F.3.1.4 MAXI

The MAXI program (Napier et al. 1984) is used to calculate a maximum annual dose to an individual from residual contamination after a nuclear facility has been decommissioned and returned to unrestricted use or converted from a nuclear installation. The individual can either be an office worker in a converted building, who is exposed to inhalation and direct radiation, or someone exposed to pathways as complex as those of a farmer growing crops and living on the site of a former nuclear facility. MAXI uses precalculated factors from versions of ARRRG, FOOD, ISOSHLD, and DACRIN. Exposure pathways that can be modeled include direct external exposure to contaminated soil or building surfaces, inhalation of resuspended material, and ingestion of contaminated foods and aquatic products. The time of the maximum dose rate to specific organs of reference is calculated and the annual dose for that organ is reported. Special options are available to tailor the program to simulate a variety of decommissioned facilities such as reactors, low-level waste burial grounds, or other facilities for handling nuclear material.

MAXI is written in FORTRAN and typically requires about 100K of memory during a 3-min run. Two versions of the code are documented, each with an interactive driver.

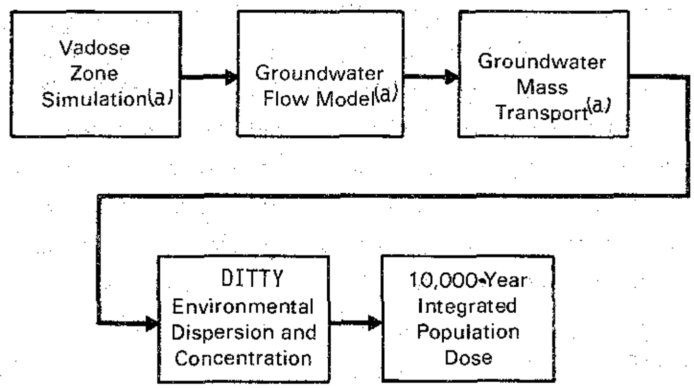
The ONSITE/MAXI1 (Napier et al. 1984) software package contains four computer codes. ONSITE is the interactive user interface that allows the end-user to simply and efficiently create and use the radiation exposure scenarios. MAXI1 is then used with the scenario information to calculate the maximum annual dose to the exposed individual from selected pathways. MAXI2 generates intermediate dose conversion factors for food pathways. These factors are stored in data files. MAXI3 calculates the data files containing intermediate dose conversion factors for aquatic pathways.

BIOPORT/MAXI1 (McKenzie et al. 1982) is a collection of computerized models designed to estimate the potential magnitude of the radiation dose to humans resulting from biotic transport processes. The BIOPORT/MAXI1 software package contains five computer codes. CREATE is the interactive computer program that allows the end-user to simply and efficiently create and evaluate biotic transport scenarios. BIOPORT simulates the redistribution of radionuclides by plant and animal processes following their intrusion into buried waste. At specified years during the biotic transport simulation, concentrations of radionuclides in the soil plow layer are determined. MAXI1 is executed with these radionuclide concentrations and a standard scenario to calculate the maximum annual dose to the maximally exposed individual from the various pathways.

The MAXI packages are presently available from PNL and NRC.

F.3.1.5 DITTY

The program estimates the time integral of collective dose over a period of up to 10,000 years for time-variant radionuclide releases to surface waters, wells, or the atmosphere. The program was initially developed to determine the collective dose from high-level waste geologic repositories resulting from groundwater pathways, but other pathways are included as well. The relationship of DITTY to the hydrogeologic models described in Appendix O is shown in Figure F.2.



(a) See Appendix O for details.

FIGURE F.2. Computer Programs for Calculating 10,000-Year Integrated Population Doses from Releases to Groundwater

Source terms of DITTY may be defined for releases to the atmosphere or to groundwater and to water wells or surface water via groundwater. The actual release rates are specified in an input file as the curies per year released for selected years following the start time of the calculation.

The time frame for the calculation is any 10,000-year period. This period is broken into 143 periods of 70 years each. The average release in each period is calculated from source-term data provided, and the total-population dose to selected organs is determined for the population present in each period. The radioactivity present during any period is the sum of material released during that period (uniformly released over 70 years) and residual material in the environment from releases in previous periods. The dose is calculated for all contributing pathways of exposure, including external exposure, inhalation, and ingestion of contaminated water and foods.

Two versions of DITTY are currently available: one for a mainframe computer, the other for an IBM personal computer. The models and solutions are identical in both cases. Minor variances between the codes occur, generally caused by minor language restrictions on the smaller machine. DITTY uses about 180K of memory. A typical problem will run 3 min on the UNIVAC and nearly 38 min on the IBM-PC. DITTY is documented (Napier, Peloquin and Strenge 1985) and is available from PNL.

F.3.1.6 ALLDOS

The computer programs used to calculate the dose to a maximumally exposed individual and to the regional population are shown in Figure F.3. The programs SUBDOSA (Strenge, Watson and Houston 1975), DACRIN (Houston, Strenge and Watson 1974; Strenge 1975), and PABLM (Napier

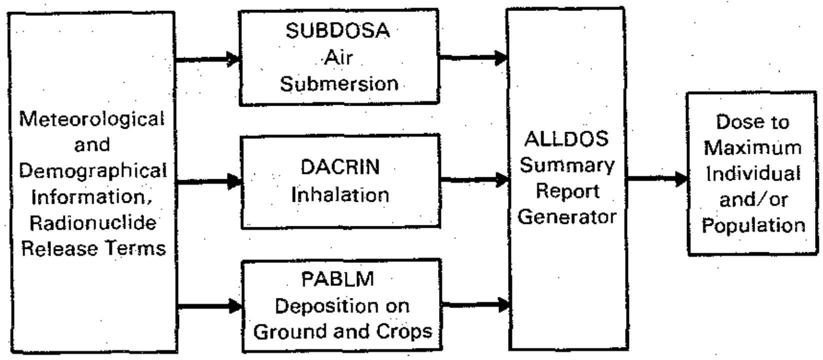


FIGURE F.3. **Computer Programs for Calculating Public Doses from Routine or Accidental Releases of Radionuclides During Operations**

et al. 1980) use information about the radionuclides released, meteorology, and population distribution to calculate air submersion, inhalation, and ingestion doses, respectively. Where many repetitive calculations are necessary for the same environmental conditions (as for the various alternatives analyzed in this EIS), it is convenient to use a fourth program, ALLDOS (Strenge et al. 1980), to summarize calculational results. This combination of computer programs was used for both the chronic (planned) and acute (accidental) releases postulated for disposal operations analyzed in this EIS.

Versions of ALLDOS are available from PNL for mainframe computers or IBM-PC installations.

F.3.2 Standard Hanford Exposure Parameters

The data used in performing dose calculations are extensive. Calculations based on effluent releases require data describing initial transport through the atmosphere or river, transfer or accumulation in terrestrial or aquatic pathways, public exposure, and dosimetry. While most of these data are contained in computer files (libraries) automatically accessed by the programs during their operation, data must also be added directly to the programs. Most of the libraries are used by more than one program, thus ensuring consistent use of the basic data for all calculations.

F.3.2.1 Population Distributions

Geographic distributions of population residing within an 80-km radius of the four operating areas are based on 1980 Bureau of Census data (Sommer, Rau and Robinson 1981). For all operational releases, the projected 1990 population within 80 km of the Hanford Meteorology Station, located midway between the 200 East and 200 West Areas, has been used. This population distribution is given in Table F.4.

For long-term releases of radionuclides to the Columbia River, estimated down-river population totals are needed. The projections of Yandon and Landstrom (1980) for the 80-km population range from about 500,000 people in the year 2000 to nearly 5,000,000 people in 10,000 years. Because the current potentially affected downriver population is about 500,000

TABLE F.4. Distribution of Population Within a 50-Mile Radius of the 200 Area Hanford Meteorology Tower by Population Grid Sector for the Year 1990 (Sommer, Rau and Robinson 1981)

Compass Direction	Number of People					
	0-16 km	16-32 km	32-48 km	48-64 km	64-80 km	Totals
N	0	202	1,320	907	2,298	4,727
NNE	0	108	790	6,448	17,482	24,828
NE	0	331	7,360	3,534	713	11,938
ENE	0	320	1,015	3,110	558	5,003
E	0	462	1,808	2,258	792	5,320
ESE	0	385	1,869	307	744	3,305
SE	0	8,664	63,866	66,306	4,094	141,930
SSE	0	2,561	16,873	3,483	6,243	29,160
S	0	1,962	1,909	251	2,114	6,236
SSW	0	1,160	6,757	787	157	8,861
SW	0	1,449	23,003	3,535	534	27,521
WSW	7	2,177	5,884	17,532	5,313	30,913
W	40	780	1,103	7,988	91,374	101,285
WNW	94	530	920	924	3,221	5,689
NW	0	652	430	499	1,467	3,048
NNW	0	289	536	1,013	5,268	7,106
Totals	141	22,032	134,443	117,882	142,372	416,870

and because no other data are available on population growth, these projections are taken to represent an upper bound of the population potentially affected by the river between Hanford and the Pacific Ocean.

F.3.2.2 Terrestrial and Aquatic Pathway Parameters

Following release and initial transport through the environment, radioactive materials may enter terrestrial or aquatic pathways that lead to public exposure. These potential pathways include consumption of fish, drinking water and foodstuffs. Input parameters describing the movement of radionuclides within potential exposure pathways include irrigation rates, growing period, holdup times, etc. These parameters are listed in Table F.5. Note that certain parameters are specific to maximum individuals and others to average individuals.

F.3.2.3 Public Exposure Parameters

Offsite radiation dose is related to the extent of public exposure to or consumption of radionuclides associated with Hanford effluents. Parameters describing assumed diet, residency and river recreation for maximum and average individuals are provided in Tables F.6 through F.8, respectively (McCormack, Ramsdell and Napier 1984).

TABLE F.5. **Values of Parameters Affecting Ingestion Pathway Exposures** (McCormack, Ramsdell and Napier 1984)

	Holdup, d[a] Maximum Individual	Holdup, d[a] Average Individual	Growing Period, d	Yield, kg/m^2	Irrigation Rate, L/m^2/month
Leafy Vegetables	1	14	90	1.5	150
Other Above-ground Vegetables	1	14	60	0.7	160
Potatoes	10	14	90	4	180
Other Root Vegetables	1	14	90	5	150
Berries	1	14	60	2.7	150
Melons	1	14	90	0.8	150
Orchard Fruit	10	14	90	1.7	0
Wheat	10	14	90	0.72	150
Other Grains	1	14	90	1.4	150
Eggs	1	18	90	0.84	150
Milk	1	4	30	1.3	200
Beef	15	34	90	0.84	140
Pork	15	34	90	0.84	140
Poultry	1	34	90	0.84	140
Fish	1	1	--	--	--
Drinking Water	1	1	--	--	--

(a) Holdup is the time between harvest and consumption.

F.3.2.4 Atmospheric Dispersion

Radioactive material released to the atmosphere becomes diluted as it is carried by the wind away from the point of release. The degree of dilution and the resultant air concentrations are predicted through the use of the Gaussian plume model (NRC 1977) and onsite measurements of atmospheric conditions.

Atmospheric dispersion data (wind speed, wind direction, and atmospheric stability) for the 200 Areas are collected at the Hanford Meteorology Station (HMS), which has been in operation since 1945. Data for the 100, 300, and 400 Areas are a composite of wind speed and direction data collected at each operating area and atmospheric stability data collected at the HMS (Stone et al. 1983).

For chronic releases, the annual average atmospheric dispersion is calculated using the sector-averaged Gaussian model and joint frequency distributions of wind speed, direction, and atmospheric stability. Values of the annual average air concentration per unit release rate ($\bar{\chi}/Q'$), in units of sec/m^3 (Ci/m^3 per Ci/sec released), calculated with this model for each operating area are available. These values of $\bar{\chi}/Q'$ have been calculated from the extended record of atmospheric data for each operating area and as such should be used for

TABLE F.6. **Dietary Parameters** (McCormack, Ramsdell and Napier 1984)

	Consumption, kg/yr	
	Maximum Individual	Average Individual
Leafy Vegetables	30	15
Other Above-ground Vegetables	30	15
Potatoes	110	100
Other Root Vegetables	72	17
Berries	30	6
Melons	40	8
Orchard Fruit	265	50
Wheat	80	72
Other Grains	8.3	7.5
Eggs	30	20
Milk	274[a,b]	230[a]
Beef	40	40
Pork	40	30
Poultry	18	8.5
Fish	40	[c]
Drinking Water	730[a,b]	438[a]

(a) Units L/yr.
(b) 330 L/yr for infant.
(c) Radiation doses are calculated based on estimated total annual catch of 15,000 kg by the total population within 80 km.

TABLE F.7. **Residency Parameters** (McCormack, Ramsdell and Napier 1984)

	Exposure, hr/day	
Parameter	Maximum Individual	Average Individual
Ground contamination	12	8
Air submersion	24	24
Inhalation[a]	24	24

(a) Inhalation Rates:
Adult--230 cm^3/sec routine; 350 cm^3/sec acute.
Infant--44 cm^3/sec.

TABLE F.8. **Recreational Activities** (McCormack, Ramsdell and Napier 1984)

Activity	Exposure, hr/yr[a] Maximum Individual	Average Individual
Shoreline	500	17
Boating	100	5
Swimming	100	10

(a) Assumes 8-hr decay time between release to river and exposure to water for maximum individual and 13 hr for average.

calculations intended to predict the potential impacts of future effluents. Tables of $\bar{\chi}/Q'$ are given for routine ground-level and elevated releases in Tables F.9 and F.10 (McCormack, Ramsdell and Napier 1984).

For acute releases, atmospheric dispersion under short-term meteorologic conditions is estimated using the sector-average model for evaluating impacts on the regional population and the centerline model for impacts on the maximally exposed individual. Dispersion estimates for assessments of postulated acute releases of effluents are based on the extended record of atmospheric data collected at the operating areas. Assessments of impacts from actual releases would be based on actual atmospheric conditions during and following the release.

Because we cannot predict precisely when a hypothetical release would occur, we conservatively assume that the release coincides with adverse atmospheric conditions. This is accomplished by calculating dispersion based on the 95th percentile atmospheric conditions derived from the recorded hourly measurements of wind speed, wind direction, and atmospheric stability. These are the conditions that predict short-term (1-hr average) air concentrations expected to be exceeded no more than 5% of the time. Doses for the maximum individual are calculated using centerline values. Population doses are calculated using sector-averaged values. These are provided in Tables F.11 through F.14 (McCormack, Ramsdell and Napier 1984).

F.3.3 Environmental Dose Code Verification

Modeling studies are relied upon to describe the potential performance of complex systems like those that define radioactive waste disposal. The major reason for conducting a modeling assessment is that real impacts upon environmental media or humans resulting from long-term release and transport cannot be measured. In addition, the low concentrations of most materials that have been released to date provide site-specific parameter values for only a few radionuclide/pathway combinations.

TABLE F.9. Annual Average Atmospheric Dispersion Parameters, $\bar{\chi}/Q'$ (sec/m^3), for Ground-Level Releases from the 200 Areas--Based on Historical Data[(a)]

Direction	Range, mi(km) 0.5(0.8)	1.5(2.4)	2.5(4.0)	3.5(5.6)	4.5(7.2)	7.5(12)	15(24)	25(40)	35(56)	45(72)
N	6.41×10^{-6}	9.81×10^{-7}	4.51×10^{-7}	2.73×10^{-7}	1.99×10^{-7}	1.02×10^{-7}	4.50×10^{-8}	2.54×10^{-8}	1.78×10^{-8}	1.35×10^{-8}
NNE	5.02×10^{-6}	7.69×10^{-7}	3.54×10^{-7}	2.14×10^{-7}	1.56×10^{-7}	8.01×10^{-8}	3.54×10^{-8}	2.00×10^{-8}	1.40×10^{-8}	1.06×10^{-8}
NE	5.84×10^{-6}	8.93×10^{-7}	4.10×10^{-7}	2.48×10^{-7}	1.81×10^{-7}	9.27×10^{-8}	4.09×10^{-8}	2.32×10^{-8}	1.62×10^{-8}	1.23×10^{-8}
ENE	9.99×10^{-6}	1.53×10^{-6}	7.02×10^{-7}	4.25×10^{-7}	3.11×10^{-7}	1.60×10^{-7}	7.08×10^{-8}	4.02×10^{-8}	2.82×10^{-8}	2.14×10^{-8}
E	2.00×10^{-5}	3.05×10^{-6}	1.41×10^{-6}	8.52×10^{-7}	6.24×10^{-7}	3.21×10^{-7}	1.43×10^{-7}	8.10×10^{-8}	5.69×10^{-8}	4.31×10^{-8}
ESE	1.92×10^{-5}	2.93×10^{-6}	1.35×10^{-6}	8.18×10^{-7}	5.98×10^{-7}	3.07×10^{-7}	1.36×10^{-7}	7.71×10^{-8}	5.40×10^{-8}	4.10×10^{-8}
SE	1.71×10^{-5}	2.62×10^{-6}	1.20×10^{-6}	7.27×10^{-7}	5.30×10^{-7}	2.71×10^{-7}	1.19×10^{-7}	6.73×10^{-8}	4.71×10^{-8}	3.56×10^{-8}
SSE	8.78×10^{-6}	1.34×10^{-6}	6.15×10^{-7}	3.72×10^{-7}	2.70×10^{-7}	1.38×10^{-7}	6.02×10^{-8}	3.39×10^{-8}	2.36×10^{-8}	1.78×10^{-8}
S	6.78×10^{-6}	1.04×10^{-6}	4.72×10^{-7}	2.86×10^{-7}	2.06×10^{-7}	1.04×10^{-7}	4.49×10^{-8}	2.50×10^{-8}	1.73×10^{-8}	1.30 x 10-8
SSW	3.76×10^{-6}	5.77×10^{-7}	2.61×10^{-7}	1.57×10^{-7}	1.13×10^{-7}	5.65×10^{-8}	2.39×10^{-8}	1.31×10^{-8}	9.02×10^{-9}	6.76×10^{-9}
SW	3.10×10^{-6}	4.76×10^{-7}	2.15×10^{-7}	1.30×10^{-7}	9.30×10^{-8}	4.67×10^{-8}	1.98×10^{-8}	1.09×10^{-8}	7.49×10^{-9}	5.61×10^{-9}
WSW	2.94×10^{-6}	4.51×10^{-7}	2.05×10^{-7}	1.24×10^{-7}	8.88×10^{-8}	4.47×10^{-8}	1.91×10^{-8}	1.05×10^{-8}	7.26×10^{-9}	5.45×10^{-9}
W	4.93×10^{-6}	6.75×10^{-7}	3.07×10^{-7}	1.86×10^{-7}	1.34×10^{-7}	6.79×10^{-8}	2.92×10^{-8}	1.63×10^{-8}	1.13×10^{-8}	8.48×10^{-9}
WNW	3.17×10^{-6}	4.86×10^{-7}	2.21×10^{-7}	1.34×10^{-7}	9.69×10^{-8}	4.92×10^{-8}	2.13×10^{-8}	1.19×10^{-8}	8.26×10^{-9}	6.23×10^{-9}
NW	5.01×10^{-6}	7.68×10^{-7}	3.51×10^{-7}	3.13×10^{-7}	1.55×10^{-7}	7.89×10^{-8}	3.45×10^{-8}	1.94×10^{-8}	1.35×10^{-8}	1.02×10^{-8}
NNW	5.03×10^{-6}	7.70×10^{-7}	3.53×10^{-7}	2.14×10^{-7}	1.56×10^{-7}	7.98×10^{-8}	3.51×10^{-8}	1.98×10^{-8}	1.39×10^{-8}	1.05×10^{-8}

(a) Data collected at the Hanford Meteorology Station from 1/76 through 1/84.

TABLE F.10. **Annual Average Atmospheric Dispersion Parameters, $\bar{\chi}/Q'$ (sec/m^3), for Elevated Releases[a] from the 200 Areas--Based on Historical Data[b]**

Direction	0.5(0.8)	1.5(2.4)	2.5(4.0)	3.5(5.6)	4.5(7.2)	7.5(12)	15(24)	25(40)	35(56)	45(72)
	Range, mi(km)									
N	5.59×10^{-8}	4.78×10^{-8}	3.80×10^{-8}	2.96×10^{-8}	2.36×10^{-8}	1.48×10^{-8}	7.32×10^{-9}	4.30×10^{-9}	3.02×10^{-9}	2.33×10^{-9}
NNE	3.82×10^{-8}	3.20×10^{-8}	2.50×10^{-8}	1.94×10^{-8}	1.55×10^{-8}	9.61×10^{-9}	4.75×10^{-9}	2.78×10^{-9}	1.94×10^{-9}	1.50×10^{-9}
NE	5.17×10^{-8}	3.75×10^{-8}	2.81×10^{-8}	2.14×10^{-8}	1.69×10^{-8}	1.03×10^{-8}	4.98×10^{-9}	2.89×10^{-9}	2.01×10^{-9}	1.55×10^{-9}
ENE	5.97×10^{-8}	4.91×10^{-8}	3.85×10^{-8}	2.99×10^{-8}	2.38×10^{-8}	1.48×10^{-8}	7.27×10^{-9}	4.26×10^{-9}	2.98×10^{-9}	2.30×10^{-9}
E	6.26×10^{-8}	7.19×10^{-8}	6.09×10^{-8}	4.91×10^{-8}	3.97×10^{-8}	2.57×10^{-8}	1.31×10^{-8}	7.83×10^{-9}	5.54×10^{-9}	4.31×10^{-9}
ESE	7.44×10^{-8}	8.62×10^{-8}	7.30×10^{-8}	5.84×10^{-8}	4.71×10^{-8}	2.99×10^{-8}	1.51×10^{-8}	8.94×10^{-9}	6.30×10^{-9}	4.89×10^{-9}
SE	1.23×10^{-7}	1.21×10^{-7}	9.55×10^{-8}	7.45×10^{-8}	5.93×10^{-8}	3.70×10^{-8}	1.83×10^{-8}	1.07×10^{-8}	7.50×10^{-9}	5.79×10^{-9}
SSE	1.15×10^{-7}	9.68×10^{-8}	7.34×10^{-8}	5.61×10^{-8}	4.43×10^{-8}	2.71×10^{-8}	1.31×10^{-8}	7.61×10^{-9}	5.30×10^{-9}	4.08×10^{-9}
S	1.54×10^{-7}	1.19×10^{-7}	8.74×10^{-8}	6.56×10^{-8}	5.12×10^{-8}	3.05×10^{-8}	1.44×10^{-8}	8.20×10^{-9}	5.66×10^{-9}	4.32×10^{-9}
SSW	1.22×10^{-7}	9.24×10^{-8}	6.37×10^{-8}	4.61×10^{-8}	3.53×10^{-8}	2.01×10^{-8}	8.99×10^{-9}	4.94×10^{-9}	3.35×10^{-9}	2.52×10^{-9}
SW	9.56×10^{-8}	6.68×10^{-8}	4.63×10^{-8}	3.37×10^{-8}	2.59×10^{-8}	1.48×10^{-8}	6.73×10^{-9}	3.74×10^{-9}	2.55×10^{-9}	1.92×10^{-9}
WSW	7.30×10^{-8}	5.93×10^{-8}	4.29×10^{-8}	3.17×10^{-8}	2.46×10^{-8}	1.43×10^{-8}	6.57×10^{-9}	3.68×10^{-9}	2.52×10^{-9}	1.91×10^{-9}
W	7.59×10^{-8}	6.71×10^{-8}	5.01×10^{-8}	3.76×10^{-8}	2.94×10^{-8}	1.73×10^{-8}	8.10×10^{-9}	4.59×10^{-9}	3.16×10^{-9}	2.40×10^{-9}
WNW	6.15×10^{-8}	5.54×10^{-8}	4.23×10^{-8}	3.21×10^{-8}	2.52×10^{-8}	1.51×10^{-8}	7.15×10^{-9}	4.09×10^{-9}	2.83×10^{-9}	2.16×10^{-9}
NW	6.75×10^{-8}	6.27×10^{-8}	4.80×10^{-8}	3.67×10^{-8}	2.89×10^{-8}	1.75×10^{-8}	8.41×10^{-9}	4.84×10^{-9}	3.37×10^{-9}	2.58×10^{-9}
NNW	5.33×10^{-8}	4.56×10^{-8}	3.63×10^{-8}	2.84×10^{-8}	2.27×10^{-8}	1.42×10^{-8}	7.07×10^{-9}	4.16×10^{-9}	2.92×10^{-9}	2.26×10^{-9}

(a) 89-m effective release height (61-m stack height and 28-m plume rise).
(b) Data collected at the Hanford Meteorology Station from 1/76 through 1/84.

TABLE F.11. 95th Percentile[a] Centerline $\bar{\chi}/Q'$ (sec/m^3) Values for Acute Ground-Level Releases from the 200 Areas[b]

Direction	Range, mi (km) 0.5(0.8)	1.5(2.4)	2.5(4.0)	3.5(5.6)	4.5(7.2)	7.5(12)	15(24)	25(40)	35(56)	45(72)
N	9.63×10^{-4}	1.61×10^{-4}	7.95×10^{-5}	5.00×10^{-5}	3.90×10^{-5}	2.27×10^{-5}	1.27×10^{-5}	8.74×10^{-6}	7.15×10^{-6}	6.03×10^{-6}
NNE	9.88×10^{-4}	1.65×10^{-4}	8.12×10^{-5}	5.12×10^{-5}	3.99×10^{-5}	2.33×10^{-85}	1.30×10^{-5}	8.95×10^{-6}	7.30×10^{-6}	6.16×10^{-6}
NE	1.03×10^{-3}	1.71×10^{-4}	8.41×10^{-5}	5.32×10^{-5}	4.14×10^{-5}	2.41×10^{-5}	1.35×10^{-5}	9.29×10^{-6}	7.56×10^{-6}	6.39×10^{-6}
ENE	8.91×10^{-4}	1.50×10^{-4}	7.46×10^{-5}	4.66×10^{-5}	3.65×10^{-5}	2.13×10^{-5}	1.19×10^{-5}	8.16×10^{-6}	6.70×10^{-6}	5.64×10^{-6}
E	9.68×10^{-4}	1.62×10^{-4}	7.99×10^{-5}	5.02×10^{-5}	3.92×10^{-5}	2.29×10^{-5}	1.28×10^{-5}	8.79×10^{-6}	7.18×10^{-6}	6.06×10^{-6}
ESE	6.88×10^{-4}	1.16×10^{-4}	5.79×10^{-5}	3.60×10^{-5}	2.83×10^{-5}	1.65×10^{-5}	9.24×10^{-6}	6.32×10^{-6}	5.19×10^{-6}	4.37×10^{-6}
SE	4.70×10^{-4}	7.59×10^{-5}	3.71×10^{-5}	2.42×10^{-5}	1.84×10^{-5}	1.09×10^{-5}	6.01×10^{-6}	4.17×10^{-6}	3.37×10^{-6}	2.85×10^{-6}
SSE	8.70×10^{-4}	1.47×10^{-4}	7.32×10^{-5}	4.56×10^{-5}	3.58×10^{-5}	2.09×10^{-5}	1.17×10^{-5}	7.99×10^{-6}	6.57×10^{-6}	5.25×10^{-6}
S	9.33×10^{-4}	1.56×10^{-4}	7.75×10^{-5}	4.85×10^{-5}	3.80×10^{-5}	2.21×10^{-5}	1.24×10^{-5}	8.50×10^{-6}	6.96×10^{-6}	5.86×10^{-6}
SSW	7.06×10^{-4}	1.18×10^{-4}	5.82×10^{-5}	3.68×10^{-5}	2.86×10^{-5}	1.67×10^{-5}	9.33×10^{-6}	6.41×10^{-6}	5.24×10^{-6}	4.42×10^{-6}
SW	7.55×10^{-4}	1.26×10^{-4}	6.27×10^{-5}	3.94×10^{-5}	3.07×10^{-5}	1.80×10^{-5}	1.00×10^{-5}	6.89×10^{-6}	5.64×10^{-6}	4.75×10^{-6}
WSW	7.66×10^{-4}	1.28×10^{-4}	6.36×10^{-5}	4.00×10^{-5}	3.12×10^{-5}	1.82×10^{-5}	1.02×10^{-5}	6.99×10^{-6}	5.73×10^{-6}	4.82×10^{-6}
W	1.18×10^{-3}	1.93×10^{-4}	9.41×10^{-5}	6.01×10^{-5}	4.65×10^{-5}	2.71×10^{-5}	1.51×10^{-5}	1.05×10^{-5}	8.47×10^{-3}	7.18×10^{-6}
WNW	1.23×10^{-3}	2.01×10^{-4}	9.78×10^{-5}	6.28×10^{-5}	4.84×10^{-5}	2.82×10^{-5}	1.57×10^{-5}	1.09×10^{-5}	8.81×10^{-6}	7.48×10^{-6}
NW	1.22×10^{-3}	2.00×10^{-4}	9.73×10^{-5}	6.23×10^{-5}	4.81×10^{-5}	2.80×10^{-5}	1.56×10^{-5}	1.09×10^{-5}	8.76×10^{-6}	7.44×10^{-6}
NNW	1.04×10^{-3}	1.73×10^{-4}	8.50×10^{-5}	5.38×10^{-5}	4.18×10^{-5}	2.44×10^{-5}	1.36×10^{-5}	9.40×10^{-6}	7.65×10^{-6}	6.47×10^{-6}

(a) One-hour average value with 5% probability of being exceeded.
(b) Data collected at the Hanford Meteorology Station from 1/76 through 1/84.

TABLE F.12. 95th Percentile[a] Centerline $\bar{\chi}/Q'$ (sec/m^3) Values for Acute Elevated[b] Releases from the 200 Areas[c]

Direction	Range, mi(km) 0.5(0.8)	1.5(2.4)	2.5(4.0)	3.5(5.6)	4.5(7.2)	7.5(12)	15(24)	25(40)	35(56)	45(72)
N	4.10×10^{-5}	2.58×10^{-5}	1.92×10^{-5}	1.48×10^{-5}	1.22×10^{-5}	9.43×10^{-6}	5.74×10^{-6}	4.77×10^{-6}	3.93×10^{-6}	3.60×10^{-6}
NNE	3.66×10^{-5}	2.44×10^{-5}	1.58×10^{-5}	1.38×10^{-5}	1.21×10^{-5}	8.49×10^{-6}	5.29×10^{-6}	4.45×10^{-6}	3.65×10^{-6}	3.37×10^{-6}
NE	3.47×10^{-5}	2.39×10^{-5}	1.47×10^{-5}	1.29×10^{-5}	1.17×10^{-5}	7.68×10^{-6}	5.09×10^{-6}	4.09×10^{-6}	3.34×10^{-6}	3.03×10^{-6}
ENE	3.12×10^{-5}	2.36×10^{-5}	1.38×10^{-5}	1.30×10^{-5}	1.19×10^{-5}	7.78×10^{-6}	4.91×10^{-6}	3.77×10^{-6}	3.07×10^{-6}	2.73×10^{-6}
E	2.81×10^{-5}	1.80×10^{-5}	1.26×10^{-5}	1.09×10^{-5}	9.28×10^{-6}	6.17×10^{-6}	4.32×10^{-6}	3.08×10^{-6}	2.50×10^{-6}	2.15×10^{-6}
ESE	2.85×10^{-5}	1.84×10^{-5}	1.26×10^{-5}	1.01×10^{-5}	8.08×10^{-6}	5.12×10^{-6}	2.78×10^{-6}	2.04×10^{-6}	1.68×10^{-6}	1.46×10^{-6}
SE	2.15×10^{-5}	1.07×10^{-5}	1.17×10^{-5}	7.95×10^{-6}	5.82×10^{-6}	3.50×10^{-6}	1.89×10^{-6}	1.52×10^{-6}	1.25×10^{-6}	1.14×10^{-6}
SSE	3.99×10^{-5}	2.45×10^{-5}	1.60×10^{-5}	1.29×10^{-5}	1.18×10^{-5}	7.55×10^{-6}	4.89×10^{-6}	3.74×10^{-6}	3.05×10^{-6}	2.70×10^{-6}
S	4.42×10^{-5}	2.48×10^{-5}	1.69×10^{-5}	1.29×10^{-5}	1.16×10^{-5}	7.18×10^{-6}	4.65×10^{-6}	3.30×10^{-6}	2.68×10^{-6}	2.29×10^{-6}
SSW	4.52×10^{-5}	2.48×10^{-5}	1.69×10^{-5}	1.24×10^{-5}	1.05×10^{-5}	6.16×10^{-6}	3.83×10^{-6}	2.75×10^{-6}	2.24×10^{-6}	1.93×10^{-6}
SW	4.56×10^{-5}	2.51×10^{-5}	1.75×10^{-5}	1.33×10^{-5}	1.12×10^{-5}	6.98×10^{-6}	4.35×10^{-6}	3.10×10^{-6}	2.52×10^{-6}	2.16×10^{-6}
WSW	4.47×10^{-5}	2.52×10^{-5}	1.76×10^{-5}	1.32×10^{-5}	1.14×10^{-5}	7.05×10^{-6}	4.48×10^{-6}	3.19×10^{-6}	2.59×10^{-6}	2.22×10^{-6}
W	4.54×10^{-5}	2.58×10^{-5}	1.90×10^{-5}	1.46×10^{-5}	1.22×10^{-5}	9.11×10^{-6}	5.40×10^{-6}	4.63×10^{-6}	3.81×10^{-6}	3.53×10^{-6}
WNW	4.62×10^{-5}	2.59×10^{-5}	1.94×10^{-5}	1.49×10^{-5}	1.22×10^{-5}	9.54×10^{-6}	5.93×10^{-6}	4.85×10^{-6}	4.00×10^{-6}	3.64×10^{-6}
NW	4.48×10^{-5}	2.55×10^{-5}	1.85×10^{-5}	1.45×10^{-5}	1.22×10^{-5}	1.02×10^{-5}	6.98×10^{-6}	5.28×10^{-6}	4.37×10^{-6}	3.84×10^{-6}
NNW	4.27×10^{-5}	2.58×10^{-5}	1.90×10^{-5}	1.47×10^{-5}	1.22×10^{-5}	9.15×10^{-6}	5.69×10^{-5}	4.74×10^{-6}	3.94×10^{-6}	3.59×10^{-6}

(a) One-hour average value with 5% probability of being exceeded.
(b) Defined as 61 m.
(c) Based on data collected at the Hanford Meteorology Station during 1982 and 1983.

9 0 1 1 7 4 1 0 9 2 9

TABLE F.13. 95th Percentile[a] Sector-Averaged $\bar{\chi}/Q'$ (sec/m^3) Values for Acute Ground-Level Releases from the 200 Areas[b]

Direction	Range, mi(km) 0.5(0.8)	1.5(2.4)	2.5(4.0)	3.5(5.6)	4.5(7.2)	7.5(12)	15(24)	25(40)	35(56)	45(72)
N	2.41×10^{-4}	3.64×10^{-5}	1.66×10^{-5}	1.02×10^{-5}	7.45×10^{-6}	3.86×10^{-6}	1.72×10^{-6}	9.88×10^{-7}	6.91×10^{-7}	5.27×10^{-7}
NNE	2.42×10^{-4}	3.65×10^{-5}	1.67×10^{-5}	1.02×10^{-5}	7.47×10^{-6}	3.87×10^{-6}	1.72×10^{-6}	9.90×10^{-7}	6.93×10^{-7}	5.28×10^{-7}
NE	2.25×10^{-4}	3.40×10^{-5}	1.55×10^{-5}	9.57×10^{-6}	6.96×10^{-6}	3.61×10^{-6}	1.61×10^{-6}	9.24×10^{-7}	6.46×10^{-7}	4.92×10^{-7}
ENE	2.00×10^{-4}	3.02×10^{-5}	1.38×10^{-5}	8.54×10^{-6}	6.19×10^{-6}	3.22×10^{-6}	1.43×10^{-6}	8.22×10^{-7}	5.75×10^{-7}	4.38×10^{-7}
E	2.01×10^{-4}	3.03×10^{-5}	1.39×10^{-5}	8.58×10^{-6}	6.22×10^{-6}	3.23×10^{-6}	1.44×10^{-6}	8.26×10^{-7}	5.77×10^{-7}	4.40×10^{-7}
ESE	1.65×10^{-4}	2.50×10^{-5}	1.15×10^{-5}	7.11×10^{-6}	5.16×10^{-6}	2.69×10^{-6}	1.20×10^{-6}	6.84×10^{-7}	4.80×10^{-7}	3.65×10^{-7}
SE	1.37×10^{-4}	2.10×10^{-5}	9.83×10^{-6}	5.91×10^{-6}	4.39×10^{-6}	2.26×10^{-6}	1.02×10^{-6}	5.76×10^{-7}	4.10×10^{-7}	3.10×10^{-7}
SSE	2.08×10^{-4}	3.14×10^{-5}	1.43×10^{-5}	8.87×10^{-6}	6.44×10^{-6}	3.35×10^{-6}	1.49×10^{-6}	8.54×10^{-7}	5.98×10^{-7}	4.55×10^{-7}
S	2.45×10^{-4}	3.70×10^{-5}	1.69×10^{-5}	1.04×10^{-5}	7.57×10^{-6}	3.92×10^{-6}	1.75×10^{-6}	1.00×10^{-6}	7.02×10^{-7}	5.35×10^{-7}
SSW	1.93×10^{-4}	2.91×10^{-5}	1.33×10^{-5}	8.26×10^{-6}	5.98×10^{-6}	3.11×10^{-6}	1.38×10^{-6}	7.94×10^{-7}	5.55×10^{-7}	4.23×10^{-7}
SW	2.17×10^{-4}	3.28×10^{-5}	1.50×10^{-5}	9.26×10^{-6}	6.73×10^{-6}	3.49×10^{-6}	1.55×10^{-6}	8.93×10^{-7}	6.24×10^{-7}	4.76×10^{-7}
WSW	2.22×10^{-4}	3.35×10^{-5}	1.53×10^{-5}	9.44×10^{-6}	6.87×10^{-6}	3.56×10^{-6}	1.59×10^{-6}	9.11×10^{-7}	6.37×10^{-7}	4.86×10^{-7}
W	2.92×10^{-4}	4.42×10^{-5}	2.01×10^{-5}	1.23×10^{-5}	9.02×10^{-6}	4.65×10^{-6}	2.07×10^{-6}	1.19×10^{-6}	8.35×10^{-7}	6.37×10^{-7}
WNW	3.09×10^{-4}	4.69×10^{-5}	2.13×10^{-5}	1.30×10^{-5}	9.55×10^{-6}	4.92×10^{-6}	2.20×10^{-6}	1.26×10^{-6}	8.85×10^{-7}	6.74×10^{-7}
NW	2.98×10^{-4}	4.51×10^{-5}	2.06×10^{-5}	1.26×10^{-5}	9.20×10^{-6}	4.74×10^{-6}	2.12×10^{-6}	1.22×10^{-6}	8.52×10^{-7}	6.50×10^{-7}
NNW	2.76×10^{-4}	4.18×10^{-5}	1.90×10^{-5}	1.17×10^{-5}	8.53×10^{-6}	4.40×10^{-6}	1.96×10^{-6}	1.13×10^{-6}	7.90×10^{-7}	6.02×10^{-7}

(a) One-hour average value with 5% probability of being exceeded.
(b) Based on data collected at the Hanford Meteorology Station during 1982 and 1983.

TABLE F.14. 95th Percentile[a] Sector-Average $\bar{\chi}/Q'$ (sec/m^3) Values for Acute Elevated Releases[b] from the 200 Areas[c]

Direction	Range, mi(km) 0.5(0.8)	1.5(2.4)	2.5(4.0)	3.5(5.6)	4.5(7.2)	7.5(12)	15(24)	25(40)	35(56)	45(72)
N	2.28×10^{-5}	1.06×10^{-5}	6.32×10^{-6}	4.52×10^{-6}	3.47×10^{-6}	2.01×10^{-6}	9.80×10^{-7}	5.92×10^{-7}	4.23×10^{-7}	3.30×10^{-7}
NNE	1.85×10^{-5}	9.95×10^{-6}	5.96×10^{-6}	4.00×10^{-6}	2.94×10^{-6}	1.85×10^{-86}	9.70×10^{-7}	5.75×10^{-7}	4.08×10^{-7}	3.15×10^{-7}
NE	1.83×10^{-5}	9.93×10^{-6}	5.84×10^{-6}	3.83×10^{-6}	2.77×10^{-6}	1.74×10^{-6}	9.01×10^{-7}	5.36×10^{-7}	3.79×10^{-7}	2.93×10^{-7}
ENE	1.61×10^{-5}	9.53×10^{-6}	5.76×10^{-6}	3.71×10^{-6}	2.64×10^{-6}	1.74×10^{-6}	9.28×10^{-7}	5.49×10^{-7}	3.87×10^{-7}	2.99×10^{-7}
E	1.39×10^{-5}	8.98×10^{-6}	4.47×10^{-6}	3.34×10^{-6}	2.49×10^{-6}	1.48×10^{-6}	5.02×10^{-7}	3.54×10^{-7}	2.53×10^{-7}	2.11×10^{-7}
ESE	1.31×10^{-5}	9.18×10^{-6}	4.57×10^{-6}	3.35×10^{-6}	2.49×10^{-6}	1.38×10^{-6}	4.77×10^{-7}	2.95×10^{-7}	1.74×10^{-7}	1.40×10^{-7}
SE	1.25×10^{-5}	6.61×10^{-6}	2.77×10^{-6}	2.90×10^{-6}	2.41×10^{-6}	1.04×10^{-6}	4.21×10^{-7}	2.04×10^{-7}	1.33×10^{-7}	1.03×10^{-7}
SSE	1.85×10^{-5}	1.06×10^{-5}	5.98×10^{-6}	4.03×10^{-6}	2.97×10^{-6}	1.74×10^{-5}	8.65×10^{-7}	5.20×10^{-7}	3.68×10^{-7}	2.86×10^{-7}
S	2.92×10^{-5}	1.06×10^{-5}	6.07×10^{-6}	4.17×10^{-6}	3.11×10^{-6}	1.74×10^{-6}	7.67×10^{-7}	4.76×10^{-7}	3.37×10^{-7}	2.66×10^{-7}
SSW	3.02×10^{-5}	1.06×10^{-5}	6.07×10^{-6}	4.17×10^{-6}	3.11×10^{-6}	1.67×10^{-6}	5.01×10^{-7}	3.53×10^{-7}	2.44×10^{-7}	2.02×10^{-7}
SW	3.05×10^{-5}	1.06×10^{-5}	6.14×10^{-6}	4.26×10^{-6}	3.20×10^{-6}	1.78×10^{-6}	7.11×10^{-7}	4.50×10^{-7}	3.19×10^{-7}	2.54×10^{-7}
WSW	2.98×10^{-5}	1.06×10^{-5}	6.15×10^{-6}	4.28×10^{-6}	3.22×10^{-6}	1.78×10^{-8}	7.31×10^{-7}	4.59×10^{-7}	3.26×10^{-7}	2.58×10^{-7}
W	2.99×10^{-5}	1.06×10^{-5}	6.30×10^{-6}	4.50×10^{-6}	3.44×10^{-6}	1.98×10^{-6}	9.76×10^{-7}	5.86×10^{-7}	4.18×10^{-7}	3.25×10^{-7}
WNW	3.07×10^{-5}	1.06×10^{-5}	6.34×10^{-6}	4.56×10^{-6}	3.50×10^{-6}	2.02×10^{-6}	9.81×10^{-7}	5.94×10^{-7}	4.25×10^{-7}	3.31×10^{-7}
NW	2.87×10^{-5}	1.06×10^{-5}	6.24×10^{-6}	4.41×10^{-6}	3.36×10^{-6}	1.97×10^{-6}	9.88×10^{-7}	6.06×10^{-7}	4.37×10^{-7}	3.42×10^{-7}
NNW	2.14×10^{-5}	1.06×10^{-5}	6.30×10^{-6}	4.50×10^{-6}	3.44×10^{-6}	1.99×10^{-6}	9.77×10^{-7}	5.87×10^{-7}	4.19×10^{-7}	3.25×10^{-7}

(a) One-hour average value with 5% probability of being exceeded.
(b) Defined as 61 m.
(c) Based on data collected at the Hanford Meteorology Station during 1982 and 1983.

Model uncertainty can best be determined by testing a model against measurements in the field under conditions similar to those the model was designed to simulate. Laboratory experiments are another potential source of comparison data if care is taken in experimental design. This process of testing predicted values against measured values is often referred to as model validation (IAEA 1981). It is impossible to validate the models used in most long-term assessments because of the complexity of the system being modeled. Sometimes parts of an overall model or submodel can be compared to limited data from another source; for example, pathway analysis models are often compared to measurements of radioactive fallout in the environment (IAEA 1984). While such exercises are useful in increasing the confidence in selecting and applying a model, they are often very incomplete. Thus care should be used so that the model validation efforts are not interpreted as being more complete than they really are. In most practical applications, models are "verified," rather than "validated." This means that their predictions are compared against results generated by similar models. The verification of a model implies that it is operating properly and gives expected results in test problems.

During the past decade, many computer codes have been developed to predict the environmental transport and subsequent impacts of radionuclide releases. These codes use various mathematical models to simulate the behavior and fate of radionuclides in environmental media by using quantitative estimates of the relationships between environmental compartments. Most of the models in use are based on the mathematical formulas originally used in the HERMES computer code (Fletcher and Dotson 1971). These include models used by EPA (Moore et al. 1979), NRC (1977), IAEA (1982), and the models used in this EIS. A recent study has compared the predictions of six internationally recognized terrestrial food-chain models, four of which are based on the HERMES-type equations, against United Nations summaries of empirical relationships between atmospheric deposition from fallout and concentrations in food of several radionuclides (Hoffman et al. 1984). Discrepancies among the model predictions varied between factors of 6 and 30. It was concluded that the differences reflected model assumptions rather than uncertainties in model parameters.

The specific computer programs used for this EIS have also received wide distribution and use. They have been compared against many other nationally used codes. Several are candidates for benchmarking studies by the NRC (Mills and Vogt 1983). These include the codes SUBDOSA and PABLM used in this EIS. Brief descriptions of some of these comparisons are given here.

F.3.3.1 Comparison of Hanford Codes for Routine Operational Releases to AIRDOS-EPA

The combined methods for calculating air submersion, inhalation, and ingestion doses using the Hanford codes KRONIC (Strenge and Watson 1973) (a finite-plume air submersion code similar to SUBDOSA), DACRIN and PABLM were compared (Aaberg and Napier 1985) to the methods used in the EPA code AIRDOS-EPA (Moore et al. 1979). The calculations were based on actual reported releases documented from Hanford facilities (Price et al. 1984). The ratios of calculated doses to various organs through the dominant pathways are shown in Table F.15.

TABLE F.15. **Ratios of Dose Commitments[a] for Maximum Individual Using Hanford/AIRDOS Models** (Aaberg and Napier 1985)

Pathway	Total Body	Lung	Thyroid	LLI[b]
Ingestion	5.1	1.3	1.2	3.1
Inhalation	1.0	3.6	0.8	0.1
External (from ground)	0.4	0.6	0.6	0.5
Air Submersion	1.2	1.6	1.5	1.4

(a) Based on actual releases from Hanford N Reactor in CY 1983 (Price 1984).
(b) LLI = lower large intenstine.

Several minor differences are apparent from the values in Table F.15. The doses calculated for food crop ingestion are somewhat higher for the Hanford PABLM code. This is partly due to a different internal dosimetry model and partly because the Hanford codes neglect atmospheric plume depletion, which tends to raise the calculated rate of deposition on crops, especially at long distances downwind (the Hanford model is thus somewhat more conservative). The external (ground deposition) doses are generally about one-half of those estimated by AIRDOS-EPA. This is because a factor of 0.5 correction for scatter from rough soil surfaces is used in the Hanford codes. This is not as conservative as the AIRDOS-EPA model, but is more representative of actual field calculations. The air submersion dose ratio in Table F.15 indicates that the Hanford model yields slightly higher doses. This is because KRONIC incorporates a finite-plume air submersion model, while AIRDOS-EPA uses an infinite-plume model. The finite-plume model accounts for the bulk of the contamination being above ground-level. At longer downwind distances than are used in this comparison, the ratio of finite to infinite plume models approaches unity. Overall, the Hanford-specific codes give results very close to those that are estimated using the EPA model for routine releases. The EPA model has been partially validated with comparisons to actual releases of ^{85}Kr at the Savannah River Plant (Fields, Miller and Cotter 1984).

F.3.3.2 Comparison of Intruder Scenario Model to NRC's 10 CFR 61 Models

In support of 10 CFR 61 the NRC issued both draft and final environmental statements (NRC 1981, 1982a). These statements describe the analysis of alternatives relating to waste forms, site design and operation, institutional controls, and administrative requirements. They also describe the radiation exposure scenario analysis used to determine near-surface disposal limits. In their analysis, the NRC defined four human intrusion scenarios. These scenarios are: 1) intruder-construction, 2) intruder-discovery, 3) intruder-agriculture, and 4) intruder-well. The disposal limits are based on a 500 mrem/yr total-body (not critical-organ) dose to the maximum-exposed individual (the intruder). The first and third scenarios are used primarily in calculating the disposal limits (NRC 1981, 1982a). For the intruder-construction scenario, an individual is assumed to excavate a basement at an abandoned disposal site. The exposure to direct penetrating radiation during this scenario controls the disposal limits for many radionuclides. For the intruder-agriculture scenario, an individual is assumed to live in the house built during the intruder-construction scenario. This individual raises part of his diet in soil that is contaminated by waste exhumed during excavation of the basement. Ingestion of radionuclides in the garden crops and inhalation of resuspended soil control the disposal limits for the remainder of the radionuclides considered in the regulation. These scenarios are conceptually similar to those described in Appendix R of this EIS.

Disposal limits are shown in Tables 1 and 2 of 10 CFR 61 for three classes of commercial wastes (NRC 1982b). Class A wastes have minimum stability requirements and low activity levels. Disposal concentrations reflect 100 years of radioactive decay that would occur during an institutional control period following site closure. Class B wastes must meet more rigorous waste-form requirements to ensure stability. Class C wastes are required to have a stable waste form and a package with higher integrity than required for Class A or B wastes. The disposal concentrations reflect 500 years of radioactive decay to account for the stability of the waste form and 5-m overburden requirement. Disposed Class C wastes are assumed to be provided ten times more protection from intrusion than that provided to disposed Class A wastes.

A recent study by Kennedy and Napier (1984) used the intruder pathway dose code MAXI, described in Section F.3 and used in Appendix R of this EIS, to re-derive the 10 CFR 61 disposal limits. The results of this effort are reproduced as Table F.16. The results obtained

TABLE F.16. Comparison of Disposal Concentrations Calculated by the Code MAXI with those in 10 CFR 61, Ci/m^3 (Kennedy and Napier 1984)

	10 CFR 61 Concentration		Concentration Calculated Using MAXI	
Radionuclide	Class A	Class C	Class A	Class C
^{14}C	0.8	8	0.8	8
^{60}Co	700	--(a)	400	--(a)
^{59}Ni	2.2	22	1	10
^{63}Ni	3.5	700	1	200
^{90}Sr+D(b)	0.04	7,000	0.03	5,000
^{99}Tc	0.3	3	3	30
^{137}Cs+D(b)	1	4,500	0.3	30,000
^{239}Pu	10(c)	100(c)	30(c)	300(c)

(a) A dash (--) indicates that no Class C limits are established (i.e., the concentration is limited only by practical considerations including the stability of the waste form, internal heat generation, and handling).
(b) +D means plus short-lived daughter.
(c) Units for ^{239}Pu are in nCi/g.

are generally within a factor of about 3 of the NRC values. The notable exception to this close agreement is the Class C disposal limit for ^{137}Cs, where the Kennedy and Napier result is about ten times the NRC value. This is caused by an error in the original Kennedy and Napier (1984) calculation. Actual MAXI results more closely parallel the NRC results. The general agreement is significant since it shows both that the dose factors and environmental models are comparable and that independently derived, dissimilar approaches can yield similar results.

The MAXI code has since been adopted by NRC for use in determining approval of proposed procedures for disposal pursuant to 10 CFR 20.302 (Napier et al. 1984).

F.3.3.3 Comparison of Long-Term Performance Assessment Codes

The Department of Energy (DOE) and the Nuclear Regulatory Commission (NRC) both employ specific models and computer codes as part of their performance assessment of potential nuclear waste repositories. The codes provide documented and traceable means to evaluate certain aspects of the repository, and the results are typically incorporated in site-selection documents, safety analysis reports, environmental impact statements, and licensing requests. Nearly all DOE offices and contractor organizations use the same program, PABLM, for environmental assessment. The NRC relies on two separate codes to independently reassess the performance of the repository. The DOE and NRC methods are compared below. In 1984, the Performance Assessment National Review Group (PANRG) addressed the adequacy of PABLM for use in assessing deep geologic repositories. The PANRG suggested improvements to PABLM, many of which are already included in the PABLM derivative code, DITTY, used in the assessments for this EIS. A comparison of the DITTY models with those used by EPA in deriving waste disposal release limits is also given below.

F.3.3.4 Comparison of PABLM with NRC Codes

In preparing nuclear waste repository site-selection documentation, safety analysis reports, environmental impact statements, and licensing requests, DOE uses certain numerical codes and computer programs to assess the potential performance of the repository. NRC uses other codes to independently reassess the DOE choices. Both approaches consist of three parts: 1) environmental transport and distribution of contamination, 2) human exposure to the contamination, and 3) human radionuclide dosimetry. The methods used for the human exposure and human dosimetry are essentially the same for NRC and DOE. Only in portions of the environmental transport do the two methods differ significantly: the NRC method uses a code, PATH1, developed by Sandia National Laboratories (SNL), which allows consideration of widespread, low-level contamination in multiple "zones," while the DOE method, to date, has considered only individual environmental "zones" (Dove 1983). The SNL/NRC approach does require additional outside hydrology/sediment transport models as a data source; however, DOE also has many codes available, and if they were used in conjunction with the present DOE methodology, the DOE and SNL/NRC approaches would be essentially indistinguishable.

The codes currently proposed or being used for the environmental consequence analysis portions of the repository performance assessment by NRC and DOE are shown in Table F.17. Essentially all of these various offices and outside contractors use the same basic codes for

TABLE F.17. **Computer Codes Proposed or Used for Environmental Consequence Analysis in Nuclear Waste Repository Performance Assessment**

Contractor,[a] Project, and Site	Surface-Water Transport	Environmental Accumulation	Human Uptake	Human Dosimetry	Human Effects
ONWI: Salt Domes					
Mississippi	--[b]	PABLM	PABLM	PABLM	--
Texas	--	PABLM	PABLM	PABLM	--
Louisiana	--	PABLM	PABLM	PABLM	--
ONWI: Bedded Salt					
Utah	--	PABLM	PABLM	PABLM	--
Texas	--	PABLM	PABLM	PABLM	--
Michigan	--	PABLM	PABLM	PABLM	--
OCRD: Crystalline Rock	--	--	--	--	--
OWI	--	PABLM	PABLM	PABLM	--
BWIP: Basalt	--	PABLM/DITTY	PABLM/DITTY	PABLM/DITTY	--
INTE	--	PABLM	PABLM	PABLM	--
AEGIS PNL					
Paradox salt	--	--	--	--	--
Permian salt	--	--	--	--	--
Generic salt	--	ARRRG/FOOD	ARRRG/FOOD	ARRRG/FOOD	--
Swedish granite	--	--	--	--	--
Gulf Coast salt	--	PABLM	PABLM	PABLM	--
Columbia basalt	--	--	--	--	--
NNWSI: Tuff	--	--	--	--	--
SNL/NRC	PATH1/Other	PATH1	PATH1/DOSHEM	DOSHEM	DOSHEM

(a) AEGIS = Assessment of the Effectiveness of Geologic Isolation Systems
BWIP = Basalt Waste Isolation Project
INTE = Intera Environmental Consultants Inc.
NNWSI = Nevada Nuclear Waste Storage Investigation
NRC = Nuclear Regulatory Commission
ONWI = Office of Nuclear Waste Isolation
OCRD = Office of Crystalline Repository Development
OWI = Office of Waste Isolation
PNL = Pacific Northwest Laboratory
SNL = Sandia National Laboratories

(b) A dash (--) indicates that no specific models are currently in use.

environmental consequence analysis. The dominant computer program is the code PABLM (Napier, Kennedy and Soldat 1980). This program was originally developed and documented by the AEGIS [then the Waste Isolation Safety Assessment Program (WISAP)] program at Pacific Northwest Laboratory (PNL) for the Office of Nuclear Waste Isolation (ONWI). ONWI now conducts the performance assessments for all candidate salt sites, both bedded and domed salt, using the PABLM code. ONWI states that, "The PABLM code represents the most up-to-date combination of detailed and broad capabilities for dose-to-man function of all the codes reviewed. The flexibility of the PABLM code is good with numerous user options, and it is applicable to a variety of radionuclide release conditions" (ONWI 1983). PABLM incorporates the capabilities of the two earlier computer programs--ARRRG and FOOD (Napier, Kennedy and Soldat 1980), which were originally developed for evaluating nuclear power reactor effluents (Soldat, Robinson and Baker 1974) and used by AEGIS in early generic assessments. The AEGIS codes were transmitted to Intera Environmental Consultants Inc. (INTE) under a technology transfer agreement. The PABLM code is being used to assess the tuff geology at the Nevada Test Site (NTS) because of the generally high quality of its documentation and because it is easy to use.[a] The Basalt Waste Isolation Project (BWIP) also identifies PABLM as the environmental assessment code of preference (BWIP 1983). BWIP personnel are also using factors calculated using the DITTY computer program (Napier, Peloquin and Strenge 1985) in preliminary assessments.[b] DITTY is a version of PABLM used in this EIS; it is described in Section F.3.

While DOE and NRC employ different codes, their approaches are similar in that each consists of three components:

- description of environmental transport and distribution of contamination
- estimation of human exposure to contamination
- calculation of human-radionuclide dosimetry.

Both environmental consequence methods begin with the time-dependent discharge of radionuclides to the biosphere. PABLM considers individual environmental zones (physical locations downstream of release points) while the PATH1 code allows the consideration of widespread low-level contamination in multiple zones.

For both approaches, once the water and soil radionuclide concentrations are known, concentration ratios are used to determine the concentration in foods. The food concentrations are then used with input consumption rates to determine human intake of radionuclides, from which the doses are calculated. The present DOE approach is to stop at individual and population doses. The NRC approach goes one step further and applies a dose-to-risk conversion factor to obtain estimates of the risk of health effects for individuals.

F.3.3.5 National Academy of Sciences Use of PABLM

A major study by the National Academy of Sciences and National Research Council (NAS/NRC) on the geologic isolation systems for geologic waste disposal (NAS/NRC 1983) based many of its conclusions on potential radiation doses to future individuals from projected waste releases. The projected doses were calculated using the PABLM code applied to well-water and surface-water scenarios similar to those used in this EIS. PABLM was selected because it is a general code for modeling environmental pathway transport and dosimetry and because it has been used extensively in modeling doses from repository releases (Cloninger, Cole and Washburn 1980; Thompson, Dove and Krupka 1984).

(a) Personal communication from J. P. Brannon, Sandia National Laboratories, to B. A. Napier, Pacific Northwest Laboratory, June 1983.
(b) Personal communication from J. C. Sonnichsen, Basalt Waste Isolation Project, to B. A. Napier, Pacific Northwest Laboratory, March 1983.

F.3.3.6 The Performance Assessment National Review Group (PANRG) Recommendations

The PANRG was convened by the DOE Office of Civilian Radioactive Waste Management (OCRWM) to review proposed performance assessment methods for the national geologic repository program. The PANRG report summarizes that although the EPA standard and NRC regulations do not specifically require the calculation of radiation dose or risk, the PANRG believes that such calculational capability should be performed for a time period beyond 10,000 years (PANRG 1984). Existing codes (e.g., PABLM), with some modifications, are believed to be usable for this purpose.

F.3.3.7 Comparison of PABLM to Atomic Energy of Canada, Limited (AECL) Models

The PANRG also made specific suggestions to improve the dosimetry modeling. These suggestions included 1) adoption of the newer ICRP-30 human dosimetry model, with some reservations, 2) inclusion of provisions to deal with resuspension of deposited radioactivity from soil into air, 3) inclusion of provisions to consider removal of radioactivity from the soil root zone over time via downward migration, 4) updating selected parameters, 5) adoption of a less conservative surface-water transport model, and 6) correction of an inappropriate model for environmental behavior of ^{14}C (PANRG 1984).

The development of the DITTY code to enhance PABLM anticipated most of these suggestions. Provisions for resuspension and soil weathering, newer parameters, and a new ^{14}C model have been included in DITTY (Napier, Peloquin and Strenge 1985). The older dosimetry and surface-water models have been retained for simplicity and conservatism at present.

In an appendix to the "Study of the Isolation System for Geologic Disposal of Radioactive Wastes" (NAS/NRC 1983), individual radiation doses per unit release to the environment calculated with the PABLM model were compared to results from the Canadian National Fuel Waste Management Program, as reported by Wuschke et al. (1981). Only those pathways to man initiated by contaminated surface water were compared. The comparison is shown in Table F.18. Because somewhat different approaches are taken by PNL and AECL, the results are not completely consistent. The results are of the same general order-of-magnitude, however, for most radionuclides.

F.3.3.8 Comparison of DITTY and EPA Long-Term Environmental Dosimetry Models

As part of its program to develop environmental standards for disposal of high-level-radioactive wastes (EPA 1985b), the EPA estimated population health risks over a 10,000-year period after disposal in mined geologic repositories. The mathematical models used to calculate environmental dose commitments and health effects are reported in EPA-520/5-85-026 (Smith, Fowler and Golden 1985). This report also identifies the data used and gives the estimates used to prepare 40 CFR 191. The data used in the EPA calculations are designed to allow calculations for a representative generic waste disposal site. For the comparison with the results of the DITTY model used in this EIS (described in Section F.3.3), the important parameters used in the EPA model defining a basalt site are described, and the EPA model results are compared with those generated by the DITTY model.

TABLE F.18. Average Annual Dose to an Individual per Unit Concentration of Radioactivity in Water as Calculated Using PNL and AECL Models (NAS/NRC 1983)

Radionuclide	Average Annual Dose per Unit Concentration ($Sv\text{-}m^3/Bq\text{-}yr$) PABLM (PNL)	Wuschke et al.[(a)] (AECL)
^{79}Se	1.6×10^{-7}	7.9×10^{-9}
^{99}Tc	7.0×10^{-10}	6.1×10^{-10}
^{129}I	2.0×10^{-8}	9.1×10^{-11}
^{135}Cs	5.3×10^{-8}	1.6×10^{-8}
^{210}Pb	7.7×10^{-6}(b)	4.9×10^{-6}
^{225}Ra	1.4×10^{-6}	1.7×10^{-7}
^{226}Ra	2.4×10^{-6}(c)	4.1×10^{-6}
^{229}Th	5.6×10^{-7}	6.4×10^{-6}
^{230}Th	8.0×10^{-8}	3.4×10^{-4}
^{233}U	3.8×10^{-8}	4.8×10^{-8}
^{234}U	3.8×10^{-8}	4.8×10^{-8}
^{238}U	2.9×10^{-8}	4.2×10^{-8}
^{237}Np	1.3×10^{-5}(d)	6.5×10^{-6}
^{239}Pu	9.8×10^{-9}	6.1×10^{-8}

NOTE: All data rounded to two significant digits.
(a) Fifty-year committed effective dose equivalent from water and food taken in during first year; derived from data of Wuschke et al. (1981).
(b) Increased by a factor of 4.4 above data of B. A. Napier (Battelle Pacific Northwest Laboratory, personal communication to T. H. Pigford, 1982) to allow for ICRP-30 corrections (Runkle and Soldat 1982).
(c) Reduced by factor of 90 below data of B. A. Napier (Battelle Pacific Northwest Laboratory, personal communication to T. H. Pigford, 1982) to allow for ICRP-30 corrections (Runkle and Soldat 1982).
(d) Increased by a factor of 200 above data of B. A. Napier (Battelle Pacific Northwest Laboratory, personal communication to T. H. Pigford, 1982) to allow for ICRP-30 corrections (Runkle and Soldat 1982).

The models and approach of the EPA differ significantly from those used by DITTY. Neither the EPA model nor DITTY can be described as being more "sophisticated" than the other because both attempt to project into admittedly imprecise futures.

For the purposes of the EPA rulemaking, Smith, Fowler and Golden (1985) evaluated the potential impacts of radionuclide releases to surface waters (rivers), oceans, land surface (through intrusions), and those due to violent interactions (e.g., volcanos, meteorites).

The river releases have the highest impact per unit release and so are those that control the EPA regulations. Thus they are analyzed here in some detail. Five exposure pathways are used to define the surface-water release impacts: drinking water, fish ingestion, food-crop ingestion, inhalation of resuspended material, and external contamination. Each pathway has a basic equation used to estimate the dose per unit release (S/Q, man-rem/curie released).

Drinking water: $S/Q = I_w D_{nop} P_R/R$
Fish ingestion: $S/Q = CF_{np} P_{FF} I_f D_{nop}/R$
Food crop ingestion: $S/Q = RI_{np} D_{nop} CP_p f_p f_R$
Inhalation of resuspended material: $S/Q = RF\ PD_p I_B D_{nop} f_R$ (function of time)
External contamination: $S/Q = f_R PD_p D_{nop}$ SOF (function of time)

where I_w = individual water ingestion rate, L/yr
D_{nop} = dose factor for nuclide n, organ o, and pathway p, units of rem/Ci ingested, rem/Ci inhaled, or rem/yr per Ci/m^2 for surface contamination.
P_R = number of people drinking water, persons
R = river flow rate, L/yr
CF_{np} = bioaccumulation factor for nuclide n in pathway p, Ci/kg per Ci/L
P_{FF} = population eating freshwater fish, persons
I_f = fish consumption rate, kg/yr per person
RI_{np} = intake rate per unit deposition of nuclide n in food pathway p, as calculated using methods similar to AIRDOS-EPA, Ci intake per Ci/m^2 deposited
CP_p = number of people who can be fed per unit area of crops, persons/m^2
f_p = fraction of land used for food crop p, dimensionless
f_R = fraction of river flow used for irrigation, dimensionless
RF = resuspension factor, m^{-1}
PD_p = population density for pathway p, persons/m^2
I_B = individual breathing rate, m^3/sec
SOF = household shielding and occupancy factor, dimensionless.

The functions of time in the equations above define the buildup and decay of surface contamination and are incidental to the following analysis, because similar methods are used by both the EPA model and DITTY.

For each pathway equation, one set of parameters can be defined as being "site-specific," that is, that realistic values for Hanford may be specific rather than generic values. For drinking water, this is the ratio P_R/R, the ratio of the number of people drinking river water to the total river flow. The value EPA uses is 3.3 x 10^{-7}. Using the projected average downriver population and a Columbia River flow rate of about 10^{14} L/yr, a Hanford value of 2 x 10^{-8} can be derived. Thus the Hanford value for this pathway is 6% of that used by EPA for their generic analysis because the Columbia River has a very large flow.

The site-specific correction for the fish consumption pathway can be incorporated in the ratio $P_{FF}I_f/R$, the ratio of the product of the number of people eating river-caught fish times consumption to the river flow rate. EPA uses a world-average value of 3.3×10^{-7} man-kg/L. Columbia River sport fishing yields only about 15,000 kg/yr of fish in the Hanford area (Price et al. 1984), for an average ingestion rate of only about 0.04 kg/person. Conservatively, assuming ten times this average for the projected down-river population gives about 7×10^{-9} man-kg/L, which is 2% of the EPA value.

The variables in the food crop ingestion equation that can be modified for Hanford releases are $f_R CP_p$, the fraction of river flow that is used for irrigation times the agricultural productivity. EPA uses a value of 0.1 for f_R, which is appropriate for small western rivers, but is much too large for the Columbia River below Hanford. While much of the area upstream of and around Hanford is irrigated with Columbia River water, only a small area below Hanford is suitable for or requires irrigation. This area is now heavily irrigated, using about 1.3% of the river flow (ERDA 1975). Accounting for the potential for increased irrigation in this area, the EPA value for fraction of land irrigated, and for the large river flow, a value of f_R can be derived of only about 0.02, without major diversion projects. The number of people who can be fed per unit area, CP_p, is estimated by EPA at about 0.004 person/m^2. Approximating this either by averaging the parameters for yield and consumption, or by dividing the assumed irrigated area by the projected population, results in a value of 0.002 person/m^2. The ratio of the EPA value for the factor $f_R CP_p$ to the value used in this EIS is thus 0.08.

For inhalation of resuspended material from irrigated soils, the parameters $f_R PD_p$ can be derived for Hanford-specific analyses. As described above, f_R is 0.1 for the EPA analysis and about 0.02 for the Hanford area. The EPA uses a value of 6.67×10^{-5} person/m^2 based on world averages. If the projected population downriver of Hanford is assumed to live in a 30-km-wide strip along the river, the population density is about 1×10^{-4} person/m^2, somewhat higher than the EPA value. Combining these gives a ratio of Hanford values to EPA value of 0.3 for the factors $f_R PD_p$.

Doses from external exposure, like inhalation, are dependent on the area irrigated and the number of people exposed. The parameters $f_R PD_p$ apply here also. The ratio for the two external exposure scenarios is then 0.3.

The EPA Background Information Document for 40 CFR 191 (EPA 1985a) presents a table of the pathway contributions to the total calculated values of health effects per unit release. That table is reproduced here as Table F.19. The individual pathways are summed to get the total. If the individual pathways are modified using Hanford-specific parameters, the results are as given in Table F.20. Compared to the Hanford-specific values calculated using the EPA model are the Hanford-specific results calculated using the DITTY model. The results can be seen to correspond closely.

A few notable exceptions to the modeling agreement can be observed in Table F.20. The newer ICRP-30 dosimetry used by EPA as illustrated in Table F.2 accounts for the differences in the values for the strontium and neptunium isotopes. The ^{14}C specific activity model used by EPA is the same one reviewed by the Performance Assessment National Review Group (PANRG) as used in PABLM (Section F.3), which is described as providing "a gross overestimate of ^{14}C

TABLE F.19. Fatal Cancers per Curie Released to a River, Estimated Using the EPA Model (Smith, Fowler, and Golden 1985)

		Ingestion					Inhalation	External Dose	
Radionuclide	Total	Drinking Water	Freshwater Fish	Surface Crops	Milk	Beef	Resuspended Material	Ground Contamination	Air Submersion
^{14}C	5.83×10^{-2}	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
^{59}Ni	4.78×10^{-5}	4.91×10^{-6}	1.25×10^{-6}	3.94×10^{-5}	4.72×10^{-7}	1.83×10^{-8}	3.25×10^{-10}	3.17×10^{-10}	1.11×10^{-15}
^{90}Sr	2.26×10^{-2}	3.72×10^{-3}	1.04×10^{-4}	1.75×10^{-2}	1.19×10^{-3}	4.59×10^{-6}	4.05×10^{-9}	0.00×10^{0}	0.00×10^{0}
^{93}Zr	1.59×10^{-4}	1.66×10^{-5}	1.41×10^{-7}	1.28×10^{-4}	4.05×10^{-7}	5.23×10^{-6}	6.58×10^{-8}	1.45×10^{-7}	4.86×10^{-14}
^{99}Tc	3.68×10^{-4}	7.02×10^{-5}	7.70×10^{-6}	2.02×10^{-4}	8.38×10^{-5}	1.38×10^{-6}	4.67×10^{-11}	0.00×10^{0}	1.80×10^{-19}
^{126}Sn	1.25×10^{-2}	2.67×10^{-4}	2.04×10^{-3}	5.37×10^{-4}	2.42×10^{-5}	3.75×10^{-5}	6.47×10^{-8}	7.55×10^{-3}	1.14×10^{-10}
^{129}I	8.09×10^{-2}	3.15×10^{-3}	2.65×10^{-4}	6.75×10^{-2}	9.68×10^{-3}	1.31×10^{-4}	3.68×10^{-8}	5.41×10^{-6}	6.86×10^{-13}
^{135}Cs	7.76×10^{-3}	2.38×10^{-4}	7.89×10^{-4}	6.10×10^{-3}	5.71×10^{-4}	3.16×10^{-5}	5.38×10^{-9}	0.00×10^{0}	0.00×10^{0}
^{137}Cs	1.07×10^{-2}	1.62×10^{-3}	5.37×10^{-3}	2.53×10^{-3}	8.42×10^{-4}	4.65×10^{-5}	1.33×10^{-9}	3.19×10^{-4}	4.45×10^{-12}
^{151}Sm	9.78×10^{-6}	4.52×10^{-6}	2.88×10^{-7}	4.53×10^{-6}	6.13×10^{-9}	2.97×10^{-8}	2.14×10^{-9}	0.00×10^{0}	1.31×10^{-17}
^{210}Pb	1.25×10^{-1}	5.40×10^{-2}	1.38×10^{-2}	4.93×10^{-2}	9.26×10^{-4}	2.16×10^{-5}	3.45×10^{-7}	9.60×10^{-8}	6.13×10^{-15}
^{226}Ra	1.68×10^{-1}	6.41×10^{-2}	8.18×10^{-3}	7.78×10^{-2}	2.41×10^{-3}	6.03×10^{-5}	8.91×10^{-6}	1.00×10^{-2}	1.56×10^{-10}
^{238}U	2.08×10^{-2}	6.32×10^{-3}	1.61×10^{-4}	1.38×10^{-2}	2.96×10^{-4}	1.91×10^{-6}	4.09×10^{-6}	2.65×10^{-5}	1.88×10^{-12}
^{237}Np	8.66×10^{-2}	2.43×10^{-2}	3.10×10^{-2}	2.41×10^{-2}	1.83×10^{-5}	7.08×10^{-6}	3.40×10^{-6}	4.83×10^{-5}	1.55×10^{-12}
^{238}Pu	4.27×10^{-2}	2.43×10^{-2}	4.96×10^{-4}	1.75×10^{-2}	1.57×10^{-7}	6.10×10^{-8}	1.14×10^{-5}	1.74×10^{-9}	1.60×10^{-15}
^{239}Pu	5.20×10^{-2}	2.61×10^{-2}	5.33×10^{-4}	2.28×10^{-2}	1.85×10^{-7}	7.18×10^{-8}	3.14×10^{-4}	2.21×10^{-8}	4.26×10^{-14}
^{240}Pu	5.03×10^{-2}	2.60×10^{-2}	5.31×10^{-4}	2.16×10^{-2}	1.80×10^{-7}	6.99×10^{-8}	2.75×10^{-4}	3.97×10^{-8}	3.55×10^{-14}
^{241}Pu	2.18×10^{-3}	1.25×10^{-3}	2.55×10^{-5}	8.94×10^{-4}	8.10×10^{-9}	3.14×10^{-9}	8.73×10^{-8}	9.46×10^{-9}	1.68×10^{-15}
^{242}Pu	5.01×10^{-2}	2.48×10^{-2}	5.07×10^{-4}	2.23×10^{-2}	1.78×10^{-7}	6.90×10^{-8}	3.13×10^{-4}	3.95×10^{-8}	3.62×10^{-14}
^{241}Am	5.80×10^{-2}	2.70×10^{-2}	5.59×10^{-3}	2.16×10^{-2}	7.63×10^{-7}	1.29×10^{-7}	3.85×10^{-5}	6.22×10^{-6}	1.10×10^{-12}
^{243}Am	6.81×10^{-2}	2.69×10^{-2}	5.56×10^{-3}	2.40×10^{-2}	8.28×10^{-7}	1.41×10^{-7}	7.92×10^{-5}	7.08×10^{-4}	2.93×10^{-11}

TABLE F.20. **Comparison of DITTY and EPA Values of Number of Fatal Cancers per Curie Released**

Nuclide	Standard EPA	Hanford-Specific EPA	Hanford-Specific DITTY	EPA Release to Ocean
^{241}Am	0.06	0.004	0.001	0.004
^{243}Am	0.07	0.004	0.03	0.01
^{14}C	0.06	N/A[(a)]	0.000007[(b)]	N/A[(a)]
^{135}Cs	0.008	0.0006	0.0004	0.00003
^{137}Cs	0.01	0.0006	0.001	0.000004
^{129}I	0.08	0.008	0.0005	0.0001
^{237}Np	0.09	0.004	0.0006[(c)]	0.007
^{238}Pu	0.04	0.003	0.0001	0.0004
^{239}Pu	0.05	0.004	0.002	0.002
^{242}Pu	0.05	0.004	0.001	0.002
^{226}Ra	0.2	0.02	0.04	0.005
^{99}Tc	0.0004	0.00004	0.00004	0.000003
^{126}Sn	0.1	0.003	0.01	0.002
^{151}Sm	0.00001	0.0000007	0.00004	0.0000004
^{90}Sr	0.02	0.002	0.02	0.00008
^{238}U	0.02	0.002	0.0005	0.0002

(a) Not specifically addressed by EPA.
(b) DITTY incorporates a revised ^{14}C model that more realistically reflects crop uptake of carbon from contaminated water.
(c) Adoption of ICRP-30 dosimetry would raise this value by a factor of nearly 10^2 (gut uptake).

in the biosphere, and, thus, in human intake" (PANRG 1984). The ^{14}C model in DITTY has been modified to account for carbon uptake by plants from air, and the doses are reduced by about three orders of magnitude. The low value for ^{151}Sm developed by the EPA discounts any external dose from surface-deposited material, which is included in DITTY.

The last column of Table F.20 presents the EPA values for radionuclide releases to oceans. For the mobile radionuclides ^{99}Tc and ^{129}I, the contribution from worldwide distribution of contamination in the ocean from the river releases is only a small increment to the total, even using the Hanford Site parameters. Therefore, to the degree of accuracy of the calculations, the integrated population doses along the Columbia River are a good approximation of the entire impact of releases from Hanford.

F.4 REFERENCES

Aaberg, R. L., and B. A. Napier. 1985. Hanford Dose Overview Program: Comparison of AIRDOS-EPA and Hanford Site Dose Codes. PNL-5633, Pacific Northwest Laboratory, Richland, Washington.

Adams, N. 1981. "Dependence on Age at Intake of Committed Dose Equivalents from Radionuclides." Phys. Med. Biol. 26(6):1019-1034.

Atomic Energy Commission (AEC). 1968. Meteorology and Atomic Energy, ed. D. H. Slade. Air Resources Laboratories, Washington, D.C.

Baker, D. A., G. R. Hoenes and J. K. Soldat. 1976. FOOD--An Interactive Code to Calculate Internal Radiation Doses from Contaminated Food Products. BNWL-SA-5523, Pacific Northwest Laboratory, Richland, Washington.

Basalt Waste Isolation Project (BWIP). 1983. Basalt Waste Isolation Project Performance Assessment Plan. SD-BWI-PAP-001, Rev. 1, Rockwell Hanford Operations, Richland, Washington.

Cloninger, M. O., C. R. Cole and J. F. Washburn. 1980. An Analysis on the Use of Engineered Barriers for Geologic Isolation of Spent Fuel in a Reference Salt Site Repository. PNL-3356, Pacific Northwest Laboratory, Richland, Washington.

Cryer, M. A., and K. F. Baverstock. 1972. "Biological Half-Life of ^{137}Cs in Man." Health Physics. 23:394-395.

Dove, F. H. 1983. Modeling Long-Term Aspects of Nuclear Waste Disposal: The AEGIS Experience. PNL-SA-10694, Pacific Northwest Laboratory, Richland, Washington.

Dunning, D. E., and G. Schwarz. 1981. "Variability of Human Thyroid Characteristics and Estimates of Dose from Ingested ^{131}I." Health Physics. 40:661-675.

Energy Research and Development Administration (ERDA). 1975. Final Environmental Statement on Waste Management Operations, Hanford Reservation. 2 vols. ERDA-1538, Washington, D.C.

Engel, R. L., J. Greenborg and M. M. Hendrickson. 1966. ISOSHLD--A Computer Code for General Purpose Isotope Shielding Analysis. BNWL-236, Pacific Northwest Laboratory, Richland, Washington.

Environmental Protection Agency (EPA). 1985a. High-Level and Transuranic Radioactive Wastes, Background Information Document for Final Rule. EPA 520/1-85-023, Washington, D.C.

Environmental Protection Agency (EPA). 1985b. "U.S. Environmental Protection Agency, 40 CFR 191, Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High Level and Transuranic Radioactive Wastes; Final Rule." Federal Register 50(182):38066, September 19, 1985.

Fields, D. E., C. W. Miller and S. J. Cotter. 1984. "Validation of the AIRDOS-EPA Computer Code by Simulating Intermediate Range Transport of ^{85}Kr from the Savannah River Plant." Atmospheric Environment 18:2029-2036.

Fletcher, J. F., and W. L. Dotson, comp. 1971. HERMES--A Digital Computer Code for Estimating Regional Radiological Effects from the Nuclear Power Industry. HEDL-TME-71-168, Hanford Engineering Development Laboratory, Richland, Washington.

Hoenes, G. R., and J. K. Soldat. 1977. Age-Specific Radiation Dose Commitment Factors for a One-Year Chronic Intake. NUREG-0172, prepared for Nuclear Regulatory Commission, Washington, D.C.

Hoffman, F. O., U. Bergstrom, C. Gyllander and A-B Wilkens. 1984. "Comparison of Predictions from Internationally Recognized Assessment Models for the Transfer of Selected Radionuclides Through Terrestrial Food Chains." Nuclear Safety 25:533-546.

Houston, J. R., D. L. Strenge and E. C. Watson. 1974. DACRIN: A Computer Program for Calculating Organ Dose from Acute or Chronic Radionuclide Inhalation. BNWL-B-389, Pacific Northwest Laboratory, Richland, Washington (reissued 1976).

International Atomic Energy Agency (IAEA). 1981. "Safety Assessment for the Underground Disposal of Radioactive Wastes." Safety Series No. 56, Vienna, Austria.

International Atomic Energy Agency (IAEA). 1982. "Generic Models and Parameters for Assessing the Environmental Transfer of Radionuclides from Routine Releases: Exposures of Critical Groups." Safety Series No. 57, Vienna, Austria.

International Atomic Energy Agency (IAEA). 1984. "Safety Analysis Metholologies for Radioactive Waste Repositories in Shallow Ground." Safety Series No. 64, Vienna, Austria.

International Commission on Radiation Units and Measurements (ICRU). 1980. Radiation Quantities and Units. ICRU Report 33, Washington, D.C.

International Commission on Radiological Protection (ICRP). 1959. "Report of ICRP Committee II on Radiological Protection." ICRP Publication 2, Pergamon Press, New York.

International Commission on Radiological Protection (ICRP). 1966. "Deposition and Retention Models for Internal Dosimetry of the Human Respiratory Tract." Health Physics 12:173.

International Commission on Radiological Protection (ICRP). 1975. "Report of the Task Group on Reference Man." ICRP Publication 23, Pergamon Press, Oxford.

International Commission on Radiological Protection (ICRP). 1977. "Recommendations of the International Commission on Radiological Protection." ICRP Publication 26, Annals of the ICRP, 2, No. 3.

International Commission on Radiological Protection (ICRP). 1978. "Radionuclide Release into the Environment Assessment of Doses to Man." ICRP Publication 29, Annals of the ICRP, 2, No. 2.

International Commission on Radiological Protection (ICRP). ICRP Publication 30, Limits for Intakes of Radionuclides by Workers:
Part 1 (1979) Annals of the ICRP, 2, No. 3/4:1-62.
Supplement to Part 1 (1979) Annals of the ICRP, 3:84, 202, 236, 414, 456.
Supplement to Part 2 (1981) Annals of the ICRP, 5:195.
Supplement A to Part 3 (1982) Annals of the ICRP, 7:17.

Kennedy, W. E., and B. A. Napier. 1984. "An Independent Derivation of the 10 CFR 61 Commercial LLW Disposal Limits." Transactions of the American Nuclear Society 46:145-146.

Leggett, R. W., K. W. Eckerman and L. R. Williams. 1982. "A Case Study in Age-Dependent Dosimetric Modeling." Health Physics 43:307-322.

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. 1982. Relevance of Biotic Pathways to the Long-Term Regulation of Nuclear Waste Disposal. NUREG/CR-2675 (PNL-4241), Vol. II, Nuclear Regulatory Commission, Washington, D.C.

Medical Research Council. 1975. Criteria for Controlling Radiation Doses to the Public After Accidental Escape of Radioactive Material. HMSO, London.

Mills, M., and D. Vogt. 1983. A Summary of Computer Codes for Radiological Assessment. NUREG/CR-3209, prepared by Teknekron Research, Inc., for the Nuclear Regulatory Commission, Washington, D.C.

Moore, R. E., C. F. Baes III, L. M. McDowell-Boyer, A. P. Watson, F. O. Hoffman, J. C. Pleasant and C. W. Miller. 1979. AIRDOS-EPA: A Computerized Methodology for Estimating Environmental Concentrations and Dose to Man from Airborne Releases of Radionuclides. ORNL-5532, Prepared for Office of Radiation Programs, Environmental Protection Agency, Washington, D.C.

Napier, B. A., et al. 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., 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 and D. L. Strenge. 1986. DITTY--A Computer Program for Calculating Population Dose Integrated Over Ten Thousand Years. PNL-4456, Pacific Northwest Laboratory, Richland, Washington.

National Academy of Sciences-National Research Council (NAS-NRC). 1983. A Study of the Isolation System for Geologic Disposal of Radioactive Wastes. Waste Isolation Systems Panel, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1977. Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I. Regulatory Guide 1.109, Rev. 1, Washington, D.C.

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

Nuclear Regulatory Commission (NRC). 1982a. Final Environmental Impact Statement on 10 CFR Part 61 "Licensing Requirements for Land Disposal of Radioactive Waste." NUREG-0945, Washington, D.C.

Nuclear Regulatory Commission (NRC). 1982b. "10 CFR Parts 2, 19, 20, 21, 30, 40, 51, 61, 70, 73, and 170: Licensing Requirements for Land Disposal of Radioactive Waste." Federal Register 47:248.

Office of Nuclear Waste Isolation (ONWI). 1983. Preliminary Performance Assessment Plant for a Nuclear Waste Repository in Salt. BMI-ONWI-545, Battelle Memorial Institute, Columbus, Ohio.

Papworth, D. G., and J. Vennart. 1973. "Retention of Sr-90 in Human Bone at Different Ages and the Resulting Radiation Doses." Phys. Med. Biol. 18(2):169-186.

Performance Assessment National Review Group (PANRG). 1985. Performance Assessment National Review Group, Final Report. Roy F. Weston, Inc., Rockville, Maryland.

Price, K. R., J. M. V. Carlile, R. L. Dirkes and M. S. Trevathan. 1984. Environmental Surveillance at Hanford for CY 1983. PNL-5038, Pacific Northwest Laboratory, Richland, Washington.

Runkle, G. E., and J. K. Soldat. 1982. "Comparison of ICRP-2 and ICRP-30 for Estimating the Dose and Adverse Health Effects from Potential Radionuclide Releases from a Geologic Waste Repository." In Proceedings of the Waste Management 1982 Symposium. SAND81-2163C, Sandia National Laboratories, Albuquerque, New Mexico.

Smith, J. M., T. W. Fowler and A. S. Golden. 1985. Environmental Pathway Models for Estimating Population Health Effects from Disposal of High-Level Radioactive Waste in Geologic Repositories. EPA 520/5-85-026, Montgomery, Alabama.

Soldat, J. K. 1971. Modeling of Environmental Pathways and Radiation Doses from Nuclear Facilities. BNWL-SA-3939, Pacific Northwest Laboratory, Richland, Washington.

Soldat, J. K. 1976. "Radiation Doses from Iodine-129 in the Environment." Health Physics 30:61-70.

Soldat, J. K., N. M. Robinson and D. A. Baker. 1974. Models and Computer Codes for Evaluating Environmental Radiation Doses. BNWL-1754, Pacific Northwest Laboratory, Richland, Washington.

Sommer, D. J., R. G. Rau and D. C. Robinson. 1981. Population Estimates for the Areas Within a 50-Mile Radius of Four Reference Points on the Hanford Site. PNL-4010, Pacific Northwest Laboratory, Richland, Washington.

Stather, J. W., J. R. Greenhalgh and N. Adams. 1983. The Metabolism of Iodine in Children and Adults. Radiological Protection Bulletin 54, National Radiological Protection Board, Harwell, United Kingdom.

Stone, W. A., J. M. Thorp, O. P. Gifford and D. J. Hoitink. 1983. Climatological Summary for the Hanford Area. PNL-4622, Pacific Northwest Laboratory, Richland, Washington.

Strenge, D. L. 1975. DACRIN--Modification of Gastrointestinal Tract Dose. BNWL-B-399, SUPP., Pacific Northwest Laboratory, Richland, Washington.

Strenge, D. L., B. A. Napier, R. A. Peloquin and M. G. Zimmerman. 1980. ALLDOS--A Computer Program for Calculation of Radiation Doses from Airborne and Waterborne Releases. PNL-3524, Pacific Northwest Laboratory, Richland, Washington.

Strenge, D. L. and E. C. Watson. 1973. KRONIC--A Computer Program for Calculating Annual Average External Doses from Chronic Atmospheric Releases of Radionuclides. BNWL-B-264, Pacific Northwest Laboratory, Richland, Washington.

Strenge, D. L., E. C. Watson and J. R. Houston. 1975. SUBDOSA--A Computer Program for Calculating External Doses from Accidental Atmospheric Releases of Radionuclides. BNWL-B-351, Pacific Northwest Laboratory, Richland, Washington.

Thompson, F. L., F. H. Dove and K. M. Krupka. 1984. Preliminary Upper-Bound Consequence Analysis for a Waste Repository at Yucca Mountain, Nevada. SAND83-7475, Sandia National Laboratories, Albuquerque, New Mexico.

United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR). 1977. Sources and Effects of Ionizing Radiation. United Nations, New York.

Wuschke, D. M., K. K. Mehta, D. W. Dormuth, T. Andres, G. R. Sherman, E. L. J. Rosinger, B. W. Goodwin, J. A. K. Reid and R. B. Lyon. 1981. Environmental and Safety Assessment Studies for Nuclear Fuel Waste Management, Vol. 3: Post-Closure Assessment. TR-127-3, Atomic Energy of Canada Ltd., Ottawa.

Yandon, K. E., and D. K. Landstrom. 1980. Supplemental Report on Population Estimates for Hanford High-Level Defense Waste Draft Programmatic Environmental Impact Statement. PNL-3128, Pacific Northwest Laboratory, Richland, Washington.

Zach, R. 1980. Sensitivity Analysis of the Terrestrial Food Chain Model FOOD III. AECL-6794, Whiteshell Nuclear Research Establishment, Pinawa, Manitoba.

THIS PAGE INTENTIONALLY
LEFT BLANK

APPENDIX G

METHOD FOR CALCULATING NONRADIOLOGICAL INJURIES AND ILLNESSES AND NONRADIOLOGICAL FATALITIES

THIS PAGE INTENTIONALLY
LEFT BLANK

APPENDIX G

METHOD FOR CALCULATING NONRADIOLOGICAL INJURIES AND ILLNESSES AND NONRADIOLOGICAL FATALITIES

This appendix describes the method used to estimate postulated nonradiological injuries and illnesses and nonradiological fatalities associated with each alternative analyzed in this EIS. (The method for calculating radiological health effects is described in Appendix F.) These calculated injuries/illnesses and fatalities, which include both the construction and operational period, are used as input to Appendix L, to Table 2 of the Executive Summary, and to Tables 3.4, 5.4, and 5.23 of this EIS.

The postulated occurrences are based on an estimate of manpower requirements and occupational accident rates of major industry groups and of DOE and its contractors. All calculations follow the same basic formulas:

$$\text{number of fatalities} = (\text{occupational fatality rate}) \times (\text{manpower required})$$
$$\text{number of injuries and illnesses} = (\text{occupational incidence rate}) \times (\text{manpower required})$$

An incidence rate is defined as the number of recordable cases of injuries and illnesses per 100 worker-years of work (200,000 worker-hours). Other categories used are defined by the Occupational Safety and Health Administration (OSHA) as follows (National Safety Council 1985):

Occupational injury is any injury such as a cut, fracture, sprain, amputation, etc., which results from a work accident or from an exposure involving a single incident in the work environment.

Occupational illness of an employee is any abnormal condition or disorder, other than one resulting from an occupational injury, caused by exposure to environmental factors associated with employment. It includes acute and chronic illnesses or disease which may be caused by inhalation, absorption, ingestion, or direct contact.

Lost workdays are those days which the employee would have worked but could not because of occupational injury or illness. The number of lost workdays should not include the day of injury or onset of illness. The number of days includes all days (consecutive or not) on which, because of injury or illness: 1) the employee would have worked but could not, or 2) the employee was assigned to a temporary job, or 3) the employee worked at a permanent job less than full time, or 4) the employee worked at a permanently assigned job but could not perform all duties normally connected with it.

Recordable cases are those involving an occupational injury or occupational illness, including deaths. Not recordable are first aid cases which involve one-time treatment and subsequent observation of minor scratches, cuts, burns, splinters, etc., which do not ordinarily require medical care, even though such treatment is provided by a physician or registered professional personnel.

Nonfatal cases without lost workdays are cases of occupational injury or illness which did not involve fatalities or lost workdays but did result in: 1) transfer to another job or termination of employment, or 2) medical treatment, other than first aid, or 3) diagnosis of occupational illness, or 4) loss of consciousness, or 5) restriction of work or motion.

Postulated incidents for each alternative are based on the type of work activity. Work activities can be grouped into three major categories: 1) onsite waste handling and monitoring, 2) transportation, and 3) repository emplacement.

The first activity, waste handling and monitoring, occurs within all the alternatives. Postulated incidents associated with onsite waste handling and monitoring are calculated by multiplying manpower estimates (Rockwell 1985) by actual incidence rates for DOE and its contractors for operational workers. The DOE incidence rates are shown in Table G.1.

TABLE G.1. DOE and Contractor Incidence Rates, 1976-80 Average (DOE 1982)

	Lost Workday Cases Cases per 100 Worker-Years
Injuries	1.1
Illnesses	0.018
Total nonfatal cases	1.1
Fatalities	0.0045

Postulated incidents associated with transportation are listed for each waste type in Appendix I, Table I.10.

Postulated incidents associated with repository emplacement are based on manpower estimates for each repository type. Repository manpower is taken from DOE (1979, 1980a) for fuel-reprocessing-waste repositories in basalt for the onsite and granite for the offsite case. Manpower for the Waste Isolation Pilot Plant (WIPP) repository is taken from DOE (1980b). The values are listed in Table G.2.

For each waste type, repository manpower is taken from Table G.2 and then prorated by the fraction of the given repository that each waste class would occupy. These fractions are based on final waste volumes and repository loading requirements. The prorated manpower is then multiplied by occupational incidence rates shown in Table G.3. The underground mining incidence rates are from the Mine Safety and Health Administration (Department of Labor 1982) and are an average for all noncoal underground mines, including metal, nonmetal and stone.

For comparison, incidence rates for general construction are 14.8 injuries and illnesses (six of these are lost workday cases) and 0.039 fatalities. These construction incidence

TABLE G.2. Manpower Requirements for Repository Construction and Operations

	Repository Manpower, Worker-Years		
	Onsite	Offsite	WIPP
Construction			
Manual	21,500	18,000	1,800
Nonmanual	4,500	4,000	700
	26,000	22,000	1,500
Operations			
General operation	15,000	16,600	7,100
Security and remote control	3,100	4,100	1,100
Underground	2,900	3,300	3,500
	21,000	24,000	11,700

TABLE G.3. Incidence Rates Used for Repository Construction and Operation Activities

Work Group	Lost Workday Cases Cases per 100 Worker-Years
DOE and Contractors	
Injuries and illnesses	1.1
Fatalities	0.0045
Underground Mining	
Injuries and illnesses	8.37
Fatalities	0.09

rates are from the Bureau of Labor Statistics (National Safety Council 1985). Incidence rates for DOE and its contractors (DOE 1982) are those shown previously in Table G.1.

Table G.4 summarizes the incidence rates used for each alternative. For alternatives that include repository emplacement, the calculated injuries, illnesses and fatalities, as reported in Appendix L and in Tables 3.4, 5.4 and 5.23, include those associated with onsite waste handling and transportation and repository construction and operation.

TABLE G.4. Injury/Illness and Fatality Rates Used for Each Alternative
(Incidents per 200,000 worker-hours)

Disposal Alternative	Injury/Illness Incidence Rate (lost workday cases)	Fatality Incidence Rate
Geologic Disposal		
Waste Processing and Stabilization	1.1	0.0045
Transportation	See Appendix I	
Repository Emplacement		
Repository construction (manual)	8.37	0.09
Operations and nonmanual construction	1.1	0.0045
In-Place Stabilization and Disposal	1.1	0.0045
Reference (Combination)		
Same breakdown as for Geologic Disposal		
No Disposal Action		
Continued Storage	1.1	0.0045

G.1 REFERENCES

Department of Energy (DOE). 1979. Technology for Commercial Radioactive Waste Management. DOE/ET-0028, Washington, D.C.

Department of Energy (DOE). 1980a. Final Environmental Impact Statement--Management of Commercially Generated Radioactive Waste. 3 vols. DOE/EIS-0046F, Washington, D.C.

Department of Energy (DOE). 1980b. Final Environmental Impact Statement--Waste Isolation Pilot Plant. DOE/EIS-0026, Assistant Secretary for Defense Programs, Washington, D.C.

Department of Energy (DOE). 1982. Environment, Safety, and Health at DOE Facilities, Annual Report, Calendar Year 1981. DOE/EP-0024/1, Washington, D.C.

Department of Labor. 1982. Injury Experience in Metallic Mineral Mining, 1981. Mine Safety and Health Administration, Denver, Colorado.

National Safety Council. 1985. Accident Facts. Chicago, Illinois.

Rockwell Hanford Operations. 1985. Hanford Defense Waste Disposal Alternatives: Engineering Support Data for the HDW-EIS. RHO-RE-ST-30 P, Richland, Washington.

APPENDIX H

RADIATION DOSES TO THE PUBLIC FROM OPERATIONAL ACCIDENTS

THIS PAGE INTENTIONALLY
LEFT BLANK

APPENDIX H

RADIATION DOSES TO THE PUBLIC FROM OPERATIONAL ACCIDENTS

This appendix was written to support the estimates of public dose, cited in Section 5 of this EIS, from operational accidents during waste processing. That section of the EIS discussed the postulated impacts and potential environmental consequences resulting from implementation of the four waste disposal alternatives. These alternatives, outlined in Chapter 3, are geologic disposal of the defense wastes, in-place stabilization and disposal, the reference alternative, and no disposal action. Occupational doses were not considered as part of this accident analysis. The information needed to provide a realistic estimate of dose to workers during an accident, such as shielding, distance, exposure time, was not available since many of these facilities have yet to be built. Any occupational doses generated as part of this analysis would be highly speculative, and as a consequence have been omitted from the EIS. In general, facilities will be designed to limit individual occupational exposure from accidents to 1 rem.

This appendix summarizes the accidents that were estimated to result in the greatest offsite radiological impact. A complete description of all accidents evaluated as part of the analysis of operational accidents is provided in PNL-5356, Potential Radiological Impacts of Upper-Bound Operational Accidents for Proposed Disposal Alternatives for Hanford Defense Waste.

H.1 SUMMARY OF UPPER-BOUND OPERATIONAL ACCIDENTS

The accident scenarios described in this appendix were developed by using information on the design of the waste processing facilities and extrapolating from other industrial facilities with similar features. It is possible to postulate accident situations for each phase of waste retrieval, handling, or disposal. Only a minor portion of these, however, have the potential to release radioactive material to uncontrolled areas with subsequent exposure of the general public. Many of the facilities and processes that will be used to dispose of the waste and that were considered as part of this accident analysis are yet to be constructed. The accident scenarios developed for them were based on a best-estimate of their future design; good engineering practices in their design and construction will make most of these accidents unlikely. Thus no attempt has been made to quantify the probability of any accidents described here. Instead the analyses in the following sections are believed to provide reasonable, credible, and conservative estimates of the maximum radionuclide releases that could occur during the processing of the waste. Impacts of operational accidents for tanks constructed in the future are expected to be within (less than) those presented in this analysis.

Several accidents were postulated and examined for each waste-handling operation. Because of the numerous scenarios considered, only a summary of the most significant can be presented in this appendix. For each waste type and disposal action option there was one controlling accident which, if it occurred, would result in the maximum airborne release of material and hence cause the greatest radiological impact. These "upper-bound" accidents and their resultant doses are listed in Tables H.1 through H.8.

The dose estimates listed in Tables H.1 through H.8 give the maximally exposed individual dose and the population dose. For each of these categories the first-year total-body and critical-organ dose[(a)] and the 70-year total-body and critical-organ dose commitment were calculated.

The highest total-body dose to a maximally exposed individual from any of the waste disposal alternatives was calculated to be 0.2 rem in the first year and 3 rem over a 70-year period. This dose is based on a summation of the ingestion and inhalation pathways, calculated separately to maximize the dose estimate (and thus provide an upper-bound estimate of potential dose). The dose received by an actual individual in such an accident would most probably be much lower. The annual, or first year, dose of 0.2 rem is below the DOE guideline[(b)] (DOE 1986) of 0.5 rem/yr to a member of the population. It is also equivalent to approximately twice the annual average background radiation dose received by a resident of the Tri-Cities from naturally occurring sources of radiation, (Price et al. 1984). The accident resulting in the greatest public dose was the explosion of the single-shell tank wastes during retrieval or handling operations. It has been postulated that a layer containing ferro- or ferricyanide precipitates might be present in the single-shell tank wastes. Under certain conditions, this material could react explosively with nitrates present in the waste. If ferrocyanide precipitates are present, the potential for an explosion does exist. However, the presence of this material in quantities sufficient to produce a large explosion is still a subject of some debate. A recent PNL report (Martin 1985) suggests that the explosion is highly unlikely.

The federal government does not currently set limits for the maximum dose that can be received by a population as a whole. Consequently, one cannot compare the population dose to a specific DOE limit. It is possible, however, to compare the estimated accidental dose to that routinely received by the same group of individuals from natural sources of radiation. About 140,000 persons were presumed to be exposed from the postulated upper-bound accidental releases. Their first-year dose was estimated to be 500 man-rem. This same group of individuals receives about 0.1 rem apiece each year from natural sources of radiation; this calculates to 1.4×10^4 man-rem, or nearly 30 times the maximum estimated dose they might receive from an accident during processing of the wastes for disposal.

H.2 TECHNICAL APPROACH AND METHODS

The following approach was used in the accident analysis and subsequent dose evaluation was to: 1) identify potential accidents and release mechanisms for each disposal/handling process, 2) determine accidents that could breach the radionuclide containment systems and

(a) That organ receiving the greatest dose in the time period considered.
(b) Vaughan, W. A. 1985. "Radiation Standards for Protection of the Public in the Vicinity of DOE Facilities." Department of Energy memorandum, August 5, 1985.

TABLE H.1. Summary of Upper-Bound Operational Accidents, Atmospheric Releases, and Maximum Individual Radiation Doses Postulated for Geologic Disposal

Waste Class	Description of Upper-Bound Accident	Respirable Release, g	Total-Body Dose, rem: 1st Yr Dose	Total-Body Dose, rem: 70 Yr Dose Commitment	Critical-Organ Dose, rem: 1st Yr Dose	Critical-Organ Dose, rem: 70 Yr Dose Commitment
Existing Tank Waste	Explosion of ferrocyanide precipitates in single-shell waste during mechanical retrieval operations	1.3×10^{4}	2×10^{-1}	3	2, Lungs	2×10^{1}, Bone
Future Tank Waste	Pressurized release of liquid waste due to failure of a diversion valve during hydraulic retrieval operations	4.5×10^{3}	9×10^{-2}	9×10^{-1}	1, Lungs	8, Bone
Strontium/Cesium Capsules	Rupture of a strontium capsule by improper handling during retrieval operations	5.5×10^{-6}	2×10^{-7}	3×10^{-6}	2×10^{-6}, Lungs, Bone	3×10^{-5}, Bone
TRU-Contaminated Soil Sites	Deflagration of contaminated material due to process malfunction in slagging pyrolysis incinerator	5×10^{1}	5×10^{-7}	2×10^{-5}	5×10^{-5}, Lungs	5×10^{-4}, Bone
Pre-1970 TRU Solid Waste	Deflagration of contaminated material due to process malfunction in slagging pyrolysis incinerator	5×10^{1}	5×10^{-6}	1×10^{-4}	2×10^{-4}, Lungs	2×10^{-3}, Bone
Retrievably Stored and Newly Generated TRU	Pressurized release from ruptured waste drum due to buildup of radiolytic gases	2×10^{3}	1×10^{-3}	5×10^{-2}	1×10^{-1}, Lungs	1, Bone

TABLE H.2. Summary of Upper-Bound Operational Accidents, Atmospheric Releases, and Population Radiation Doses Postulated for Geologic Disposal

Waste Class	Description of Upper-Bound Accident	Respirable Release, g	Total-Body Dose, man-rem: 1st Yr Dose	Total-Body Dose, man-rem: 70 Yr Dose Commitment	Critical-Organ Dose, man-rem: 1st Yr Dose	Critical-Organ Dose, man-rem: 70 Yr Dose Commitment
Existing Tank Waste	Explosion of ferrocyanide precipitates in single-shell waste during mechanical retrieval operations	1.3×10^4	4×10^2	7×10^3	4×10^3, Lungs, Bone	6×10^4, Bone
Future Tank Waste	Pressurized release of liquid waste due to failure of a diversion valve during hydraulic retrieval operations	4.5×10^3	3×10^2	2×10^3	2×10^3, Lungs	2×10^4, Bone
Strontium/Cesium Capsules	Rupture of a strontium capsule by improper handling during retrieval operations	5.5×10^{-6}	6×10^{-4}	1×10^{-2}	5×10^{-3}, Lungs, Bone	8×10^{-2}, Bone
TRU-Contaminated Soil Sites	Deflagration of contaminated material due to process malfunction in slagging pyrolysis incinerator	5×10^1	1×10^{-3}	4×10^{-2}	1×10^{-1}, Lungs	1, Bone
Pre-1970 TRU Solid Waste	Deflagration of contaminated material due to process malfunction in slagging pyrolysis incinerator	5×10^1	1×10^{-2}	3×10^{-1}	5×10^{-1}, Lungs	5, Bone
Retrievably Stored and Newly Generated TRU	Pressurized release from ruptured waste drum due to buildup of radiolytic gases	2×10^3	3	1×10^2	3×10^2, Lungs	2×10^3, Bone

TABLE H.3. Summary of Upper-Bound Operational Accidents, Atmospheric Releases, and Maximum Individual Radiation Doses Postulated for In-Place Stabilization and Disposal

Waste Class	Description of Upper-Bound Accident	Respirable Release, g	Total-Body Dose, rem		Critical-Organ Dose, rem	
			1st Yr Dose	70 Yr Dose Commitment	1st Yr Dose	70 Yr Dose Commitment
Existing Tank Waste	Explosion of ferrocyanide precipitates in single-shell waste during stabilization operations	1.3×10^{4}	2×10^{-1}	3	2, Lungs	2×10^{1}, Bone
Future Tank Waste	Pressurized release of liquid waste due to failure of a diversion valve during hydraulic retrieval operations	4.5×10^{3}	9×10^{-2}	9×10^{-1}	1, Lungs	8, Bone
Strontium/Cesium Capsules	Shearing of a strontium capsule by improper handling during disposal operations	2.2×10^{-3}	3×10^{-4}	4×10^{-3}	3×10^{-3}, Lungs	4×10^{-2}, Bone
TRU-Contaminated Soil Sites	Collapse of voids in soil site during subsidence-control operations	2.6	2×10^{-8}	9×10^{-7}	2×10^{-6}, Lungs	2×10^{-5}, Bone
Pre-1970 TRU Solid Waste	Collapse of void space at waste site during subsidence-control operations	2.6	3×10^{-7}	7×10^{-6}	1×10^{-5}, Lungs	1×10^{-4}, Bone
Retrievably Stored and Newly Generated TRU	Breach of waste container during package-disposal operations	1×10^{3}	2×10^{-3}	4×10^{-2}	1×10^{-1}, Lungs	8×10^{-1}, Bone

TABLE H.4. Summary of Upper-Bound Operational Accidents, Atmospheric Releases, and Population Radiation Doses Postulated for In-Place Stabilization and Disposal

Waste Class	Description of Upper-Bound Accident	Respirable Release, g	Total-Body Dose, man-rem: 1st Yr Dose	Total-Body Dose, man-rem: 70 Yr Dose Commitment	Critical-Organ Dose, man-rem: 1st Yr Dose	Critical-Organ Dose, man-rem: 70 Yr Dose Commitment
Existing Tank Waste	Explosion of ferrocyanide precipitates in single-shell waste during stabilization operations	1.3×10^4	4×10^2	7×10^3	4×10^3, Lungs, Bone	6×10^4, Bone
Future Tank Waste	Pressurized release of liquid waste due to failure of a diversion valve during hydraulic retrieval operations	4.5×10^3	3×10^2	2×10^3	2×10^3, Lungs	2×10^4, Bone
Strontium/Cesium Capsules	Shearing of a strontium capsule by improper handling during disposal operations	2.2×10^{-3}	6×10^{-1}	1×10^1	6, Bone, Lungs	9×10^1, Bone
TRU-Contaminated Soil Sites	Collapse of voids in soil site during subsidence-control operations	2.6	5×10^{-5}	2×10^{-3}	5×10^{-3}, Lungs	5×10^{-2}, Bone
Pre-1970 TRU Solid Waste	Collapse of void space at waste site during subsidence-control operations	2.6	6×10^{-4}	2×10^{-2}	3×10^{-2}, Lungs	2×10^{-1}, Bone
Retrievably Stored and Newly Generated TRU	Breach of waste container during package-disposal operations	1×10^3	5	8×10^1	3×10^2, Lungs	2×10^3, Bone

TABLE H.5. Summary of Upper-Bound Operational Accidents, Atmospheric Releases, and Maximum Individual Radiation Doses Postulated for the Reference Alternative

Waste Class	Description of Upper-Bound Accident	Respirable Release, g	Total-Body Dose, rem: 1st Yr Dose	Total-Body Dose, rem: 70 Yr Dose Commitment	Critical-Organ Dose, rem: 1st Yr Dose	Critical-Organ Dose, rem: 70 Yr Dose Commitment
Existing Tank Waste	Explosion of ferrocyanide precipitates in single-shell waste during stabilization operations	1.3×10^4	2×10^{-1}	3	2, Lungs	2×10^1, Bone
Future Tank Waste	Pressurized release of liquid waste due to failure of a diversion valve during hydraulic retrieval operations	4.5×10^3	9×10^{-2}	9×10^{-1}	1, Lungs	8, Bone
Strontium/Cesium Capsules	Rupture of a strontium capsule by improper handling during retrieval operations	5.5×10^{-6}	2×10^{-7}	3×10^{-6}	2×10^{-6}, Lungs, Bone	3×10^{-5}, Bone
TRU-Contaminated Soil Sites	Collapse of voids in soil site during subsidence-control operations	2.6	2×10^{-8}	9×10^{-7}	2×10^{-6}, Lungs	2×10^{-5}, Bone
Pre-1970 TRU Solid Waste	Collapse of void space at waste site during subsidence-control operations	2.6	3×10^{-7}	7×10^{-6}	1×10^{-5}, Lungs	1×10^{-4}, Bone
Retrievably Stored and Newly Generated TRU	Pressurized release from ruptured waste drum due to buildup of radiolytic gases	2×10^3	2×10^{-3}	6×10^{-2}	2×10^{-1}, Lungs	1, Bone

TABLE H.6. Summary of Upper-Bound Operational Accidents, Atmospheric Releases, and Population Radiation Doses Postulated for the Reference Alternative

Waste Class	Description of Upper-Bound Accident	Respirable Release, g	Total-Body Dose, man-rem: 1st Yr Dose	Total-Body Dose, man-rem: 70 Yr Dose Commitment	Critical-Organ Dose, man-rem: 1st Yr Dose	Critical-Organ Dose, man-rem: 70 Yr Dose Commitment
Existing Tank Waste	Explosion of ferrocyanide precipitates in single-shell waste during stabilization operations	1.3×10^4	4×10^2	7×10^3	4×10^3, Lungs, Bone	6×10^4, Bone
Future Tank Waste	Pressurized release of liquid waste due to failure of a diversion valve during hydraulic retrieval operations	4.5×10^3	3×10^2	2×10^3	2×10^3, Lungs	2×10^4, Bone
Strontium/Cesium Capsules	Rupture of a strontium capsule by improper handling during retrieval operations	5.5×10^{-6}	6×10^{-4}	1×10^{-2}	5×10^{-3}, Lungs, Bone	8×10^{-2}, Bone
TRU-Contaminated Soil Sites	Collapse of voids in soil site during subsidence-control operations	2.6	5×10^{-5}	2×10^{-3}	5×10^{-3}, Lungs	5×10^{-2}, Bone
Pre-1970 TRU Solid Waste	Collapse of void space at waste site during subsidence-control operations	2.6	6×10^{-4}	2×10^{-2}	3×10^{-2}, Lungs	2×10^{-1}, Bone
Retrievably Stored and Newly Generated TRU	Pressurized release from ruptured waste-drum due to buildup of radiolytic gases	2×10^3	4	1×10^2	4×10^2	3×10^3

TABLE H.7. Summary of Upper-Bound Operational Accidents, Atmospheric Releases, and Maximum Individual Radiation Doses Postulated for No Disposal Action

Waste Class	Description of Upper-Bound Accident	Respirable Release, g	Total-Body Dose, rem		Critical-Organ Dose, rem	
			1st Yr Dose	70 Yr Dose Commitment	1st Yr Dose	70 Yr Dose Commitment
Existing Tank Waste	Pressurized release of liquid waste due to failure of a diversion valve during hydraulic retrieval operations	4.5×10^{3}	6×10^{-2}	9×10^{-1}	7×10^{-1}, Lungs	8, Bone
Future Tank Waste	Pressurized release of liquid waste due to failure of a diversion valve during hydraulic retrieval operations	4.5×10^{3}	9×10^{-2}	9×10^{-1}	1, Lungs	8, Bone
Strontium/Cesium Capsules	Rupture of a strontium capsule by improper handling during retrieval operations	5.5×10^{-6}	2×10^{-7}	3×10^{-6}	2×10^{-6}, Lungs, Bone	3×10^{-5}, Bone
TRU-Contaminated Soil Sites	Collapse of void in soil site during site-stabilization activities	2.6	2×10^{-8}	9×10^{-7}	2×10^{-6}, Lungs	2×10^{-5}, Bone
Pre-1970 TRU Solid Waste	Collapse of void space at waste site during site-stabilization activities	2.6	3×10^{-7}	7×10^{-6}	1×10^{-5}, Lungs	1×10^{-4}, Bone
Retrievably Stored and Newly Generated TRU	Collapse of void space at waste site during site-stabilization activities	2.6	5×10^{-6}	7×10^{-5}	3×10^{-4}, Lungs	1×10^{-3}, Bone

TABLE H.8. Summary of Upper-Bound Operational Accidents, Atmospheric Releases, and Population Radiation Doses Postulated for No Disposal Action

Waste Class	Description of Upper-Bound Accident	Respirable Release, g	Total-Body Dose, man-rem		Critical-Organ Dose, man-rem	
			1st Yr Dose	70 Yr Dose Commitment	1st Yr Dose	70 Yr Dose Commitment
Existing Tank Waste	Pressurized release of liquid waste due to failure of a diversion valve during hydraulic retrieval operations	4.5×10^3	1×10^2	2×10^3	2×10^3, Lungs	2×10^4, Bone
Future Tank Waste	Pressurized release of liquid waste due to failure of a diversion valve during hydraulic retrieval operations	4.5×10^3	3×10^2	2×10^3	2×10^3, Lungs	2×10^4, Bone
Strontium/Cesium Capsules	Rupture of a strontium capsule by improper handling during retrieval operations	5.5×10^{-6}	6×10^{-4}	1×10^{-2}	5×10^{-3}, Lungs, Bone	8×10^{-2}, Bone
TRU-Contaminated Soil Sites	Collapse of void in soil site during site-stabilization activities	2.6	5×10^{-5}	2×10^{-3}	5×10^{-3}, Lungs	5×10^{-2}, Bone
Pre-1970 TRU Solid Waste	Collapse of void space at waste site during site-stabilization activities	2.6	6×10^{-4}	2×10^{-2}	3×10^{-2}, Lungs	2×10^{-1}, Bone
Retrievably Stored and Newly Generated TRU	Collapse of void space at waste site during site-stabilization activities	2.6	1×10^{-2}	2×10^{-1}	7×10^{-1}, Lungs	3, Bone

provide a pathway of escape for the radionuclides to the biosphere, 3) estimate the fraction of radionuclides released, 4) calculate doses from the estimated releases using established models as described in Appendix F, and 5) consider significant mitigating factors. The following sections describe the key assumptions used in developing the accident scenarios, the radionuclide inventories used in the accident analysis, and general considerations used in determining release fractions and performing the dose calculations.

H.2.1 Key Assumptions

The following key assumptions were made in the analysis of operational accidents:

1. All facilities, processes, and operations are or will be designed, constructed, and used in a manner consistent with prudent and proven practices.
2. The processes, facilities, operations, radionuclide inventories, and waste forms are those described in the engineering support data (Rockwell 1985).
3. The upper-bound accident identified as having the greatest potential radiological consequences for a given operation conservatively bounds all other credible accidents that likely could occur during that operation.

H.2.2 Radionuclide Inventories

The radionuclide inventories used in the analysis of operational accidents are based on those shown in Appendix A of this EIS. They were converted from quantity to concentration to make them suitable for use in the dose calculations. Conversions were done using known waste volumes or densities to provide the proper units. For additional information on the conversion factors used see PNL-5356 (Mishima et al. 1986).

H.2.3 Downwind Transport and Dose Assessment Methods

The radiological impact on the general public from one of these accidents is dependent on the quantity of material that escapes from a processing or waste facility and becomes airborne. The estimates of fractional airborne releases for each of the accidents described in this appendix were based on previously published data that examined common industrial accidents including fires, explosions, container ruptures, etc. to provide realistic estimates.

Population and maximum-individual dose estimates were calculated for each accident scenario postulated to result in significant release. Occupational doses were not addressed because of the unavailability of facility-specific information (such as manpower requirements, shielding, distance from the source, etc.) essential to analysis of occupational dose. The assumptions, models, and input parameters required for the calculation of maximum individual and population dose are described below.

Many different accident scenarios were developed as part of this effort. Only those resulting in a release of radioactive material to the offsite environment were considered in the dose analysis. The duration of a release during an accident can significantly affect the radiological consequences of the event. In this study all releases were postulated to be of short duration (less than an hour). Even with a short-term (also known as acute) release, there are many ways the radionuclides can continue to expose the population long after the release has been terminated. For example, in a typical accident scenario, a cloud (or plume) of contaminated material is postulated to be released. As the plume travels off site, members of the public may be irradiated by the radionuclides contained in the cloud passing overhead. If they inhale some of the radioactive material from the cloud as it passes, they can receive an additional exposure. If some of the radioactive material deposits on plants or on the ground, long-term exposure to people residing in the area can result. The standard method for evaluating the radiological impact of a release is to estimate the dose to the "maximally exposed individual" (the single person receiving the highest dose from the release) and to the entire exposed population as a whole. The doses are reported in rem for the maximum individual and man-rem for the population. The doses calculated for the analysis of operational accidents included the first-year dose and the 70-year dose commitment. The 70-year dose commitment is calculated based on one-year exposure to the material in the environment. For additional discussion of this topic, see Appendix F, Section F.1.3.

The computer programs used to calculate doses to the maximally exposed individual and to the regional population are discussed in Appendix F. The computer program, ALLDOS (Strenge et al. 1980), was used to summarize the results of the calculations. ALLDOS uses precalculated dose conversion factors, developed through application of other dose programs, to generate dose commitments to a maximum individual and the population in the region of the release site. The code was developed for calculating radiation doses from postulated releases of aged radioactive wastes. These radionuclides are long-lived with decay half-lives of several weeks or longer. Therefore, radioactive decay in transit from the release point to the location of exposure in the environment is not considered.

The dose calculations rely on the use of meteorological data to estimate the manner in which radioactive material would most likely disperse following an accidental release to the atmosphere. For short-term accidental releases the meteorological parameter used in the dose calculations is the value of air concentration of radionuclides per unit release that is not exceeded more than 5% of the time; it is referred to as E/Q, with units of sec/m^3. Typically the results of the meteorological efforts are tabulated and reported as $\bar{\chi}/Q'$, or Ci/m^3 per Ci/sec of release. The value of $\bar{\chi}/Q'$ can be converted to E/Q when the length of release is known or can be estimated. Values of $\bar{\chi}/Q'$ used in these calculations were based on data given in PNL-3777 Rev 1 (McCormack, Ramsdell and Napier 1984).

Demographic data also play an important role in the calculation of radiation dose. It is the combination of meteorological and demographical information that indicates which population group will receive the highest exposure from radioactive releases. In the case of accidental releases from the 200 Areas of the Hanford Site, the population projected to receive the greatest exposure lives 16 to 80 km southeast of the waste site. The population data used in this assessment came from Population Estimates for the Areas Within a 50-Mile Radius of Four Reference Points on the Hanford Site (Sommer, Rau and Robinson 1981) and is for the 1990 projected population. Meteorological and population data used in the dose calculations are shown in Table H.9.

TABLE H.9. **Population Values and Sector Averaged $\bar{\chi}/Q'$s Used in the Assessment of Radiation Dose**

Distance, km	Population Size	Ground Level χ/Q'	Elevated χ/Q'
0-16	0		
16-32	8,664	1.02×10^{-6}	4.21×10^{-7}
32-48	62,866	5.76×10^{-7}	2.04×10^{-7}
48-65	66,306	4.10×10^{-7}	1.33×10^{-7}
65-80	4,094	3.10×10^{-7}	1.03×10^{-7}
Population Weighted χ/Q'		7.35×10^{-2}	2.54×10^{-2}

For the maximally exposed individual, the 95th percentile center-line $\bar{\chi}/Q'$ values provided in Hanford Dose Overview Program: Standardized Methods and Data for Hanford Environmental Dose Calculations (McCormack, Ramsdell and Napier 1984) were used. The following assumptions were used to determine the location of the maximally exposed individual for accidental releases. For purposes of inhalation and submersion dose calculations, the maximally exposed individual was assumed to be positioned on Highway 240, 8.8 km south of the 200 Areas; but for ingestion dose calculations this person is presumed to live on a farm in Franklin County 19 km east of the 200 Areas. The values used in the calculations are shown in Table H.10.

TABLE H.10. **Maximum Individual, 95th Percentile Centerline $\bar{\chi}/Q'$ Values**

Pathway	Location	Elevation: Ground	Elevation: 60 m
Inhalation	8 km S	3.40×10^{-5}	1.05×10^{-5}
Ingestion	19 km E	1.50×10^{-5}	4.90×10^{-6}

Data required for the dose programs include dietary and recreational preferences and habits in the general population, as well as agricultural practices in the general region. The standard Hanford terrestrial pathway data used as part of the dose calculations are given in Appendix F. Standardized input for Hanford environmental documentation is summarized in recent publications (Napier 1981; McCormack, Ramsdell and Napier 1984).

H.3 DESCRIPTION OF ACCIDENTS FOR THE GEOLOGIC DISPOSAL ALTERNATIVE

Several operations are required to process each of the six waste forms for geologic disposal. The processing of the waste for disposal is depicted in Table H.11. Twenty-six separate dose calculations were performed to analyze the potential radiological impact from the

TABLE H.11. Waste-Processing Steps for Geologic Disposal

Existing Tank Waste	Newly Generated and Future Tank Waste	Strontium and Cesium Capsules	TRU-Contaminated Soil Sites	Pre-1970 Buried Suspect TRU-Contaminated Solid Waste	Retrievably Stored and Newly Generated TRU Wastes
Retrieve at least 95% of single-shell tank contents and 99.95% of double-shell tank contents.	Retrieve 99.95% of wastes in double-shell tanks. Separate strontium, cesium, technetium, and TRU elements for vitrification and geologic disposal.	Package capsules in canisters for repository disposal.	Recover TRU waste to 100-nCi/g level.	Recover wastes defined as TRU.	Retrieve TRU waste and sort and package it to WIPP waste acceptance criteria.
Separate strontium, cesium, technetium, and TRU elements for vitrification and repository disposal. Send residual low-activity wastes as grout to near-surface vaults or to existing tanks.	Immobilize the residual waste as grout and dispose of in near-surface vaults or in existing tanks.	Send canisters to geologic repository for disposal.	Treat, package, and send waste to Waste Isolation Pilot Plant (WIPP).	Treat, package, and send waste to WIPP.	Package and treat waste as required, and send it to WIPP.
Fill tanks sufficiently with appropriate material to limit subsidence, and seal all accessible penetrations.	Fill tanks sufficiently with appropriate material to limit subsidence, and seal all accessible penetrations.		Backfill excavated sites and dispose of as low-level waste sites.	Backfill excavated sites and dispose of as low-level waste sites.	Dispose of residual waste as low-level waste.
Leave soil contaminated from tank leaks in place.	Cover grout disposal waste sites and tanks with the protective barrier, emplace markers, and record.				
Cover emptied tank farms and grout disposal sites with the protective barrier, emplace markers, and record.					

disposal of the Hanford Defense Wastes under the geologic disposal alternative. This section contains a brief description of the waste processing operations used and the facility involved during the upper-bound accident.

H.3.1 Existing Tank Wastes

During geologic disposal, the contents of the existing tank wastes are retrieved mechanically or through hydraulic sluicing. The strontium, cesium, technetium, and transuranic elements are separated for vitrification, and the residual wastes are grouted and disposed of in near-surface vaults. The empty tanks are then filled, sealed, and the tank farms are covered with soil and with the protective barrier and marker system.

Throughout the disposal operation, the potential for accidental release of radioactive material exists. The accident with the greatest release of radioactive material was postulated to occur during mechanical retrieval of the salt cake and sludge from the single-shell tank wastes. The single-shell tanks have received a variety of waste streams during their active lifetime, and there is the potential for explosive mixtures to be produced through reactions of the tank contents. The salt cake is composed of many salts; among them is sodium nitrate, a powerful oxidizer. If ferrocyanide precipitates are also present in the waste, the cyanide and nitrate ions could, with the proper configuration and amount of material, react to cause an explosion during handling of the waste. The explosion would have sufficient energy to breach the filters on the tank and release radionuclides as aerosols directly to the atmosphere. A recent PNL report (Martin 1985) shows that such an explosion is highly unlikely.

Steindler and Seefeldt (1980) developed a method to predict aerosol production from a detonation. This method was used to estimate a release of 4.98×10^8 g of aerosol from 2,000 m^3 of salt cake. The respirable fraction of material, 10 μm aerodynamic equivalent diameter and smaller, was calculated to be 1.3×10^4 g. This value was used in the calculation of offsite dose impacts.

Other accident scenarios considered as part of the analysis of operational accidents for disposal of the existing tank wastes included: suspension of contaminated soil during sampling; waste spills during retrieval or handling of the waste; loss of high-efficiency particulate air filtration; loss of services or power; pressurized release during hydraulic sluicing of the double-shell tank wastes; pipe breaks; hydrogen explosions; fires; leaks; and explosions.

H.3.2 Future Tank Wastes

The disposal method for future tank waste under the geologic disposal alternative would be to remove as much of the waste as practicable to a geologic repository. The processing of the future tank wastes would be integrated with the processing of the existing tank waste contents, and many of the remaining operations will be the same. These wastes, which are stored in double-shell tanks, would be retrieved through hydraulic sluicing, the same process as is used for the existing double-shell tank waste.

The upper-bound accident developed for the future tank wastes under the geologic disposal alternative involves a release of waste during hydraulic retrieval of the tank contents for processing. The operation involves recovery and transfer of the radioactive liquid using multistage pumps, deep-well turbine pumps, and shielded piping. Sludges are removed using a sluicer composed of a high-pressure water supply system and a nozzle-aiming mechanism. The slurry and liquid are transferred to waste-processing facilities or other tanks. A pressurized release of the liquid waste is postulated as the upper-bound release event. The scenario of the accident is as follows: recycled liquid is pumped to a manifold where it will be directed back to be used with the sluicer. Failure of the diversion valve in the manifold results in the backflow of waste solution into an unenclosed area with the spray release of liquid. The waste in the slurry is assumed to be as insoluble particles, with a concentration of about 25% by volume; this is similar to slurries of coal or gravel that are pumped (Perry 1984). Using a pumping rate of 0.2 m^3/min (Rockwell 1980) and a waste density of 1.8×10^6 g/m^3, the quantity of waste pumped would be 9×10^4 g/min. Not all the liquid would become airborne, but with a nominal wind speed of 7.6 mph (Stone, Jenne and Thorp 1972) about 5% of the released material could become airborne (Sutter 1980). Moderate wind speeds were chosen to maximize the combination of resuspension and airborne concentration of the radionuclides transported off site during the accident. The total postulated release is then 4.5×10^3 g/min. A 1-min ground-level release is postulated, which would result in an estimated airborne release of 4.5×10^3 g.

Other accidents reviewed as part of the analysis for the future tank wastes included filter failure during solid/liquid separation; a fire of the ion exchange resin; the spraying of contaminated liquid from a process line; and loss of filters during immobilization of the waste in glass.

H.3.3 Strontium and Cesium Capsules

The strontium and cesium capsules are to be stored in the Waste Encapsulation and Storage Facility until a repository is available and then removed for geologic disposal. During this time they will be periodically inspected; storage of capsules requires cooling water, makeup water, ventilation and maintenance of the facility operating systems.

The accident analysis was based on the design described by Braden et al. (1971) in the safety analysis report for the Waste Encapsulation and Storage Facility. The operation and equipment design and usage are assumed in keeping with sound, prudent nuclear practices. The upper-bound airborne release event postulated to occur with the strontium and cesium capsules under the geologic disposal alternative was the rupture of one capsule. Waste canisters can be breached before they are encapsulated if they are dropped during handling operations (Hayward and Jensen 1980). This study has shown that once encapsulated, the canisters will not rupture even if dropped onto concrete from a height of up to 6 m, or when struck by heavy falling objects such as cell covers. However, if the encapsulated canisters become degraded during extended storage, rupture during handling is possible. This analysis assumed that a capsule ruptured upon impact and released some of the contents. Current literature provides no means to estimate the fraction released and made airborne during such an event. It was assumed, however, that the high pressure that would have to be generated within the capsule to create sufficient energy to rupture it would produce a finely sized particulate release. Studies of the release of fine depleted uranium dioxide powder under 50-psi pressure showed that about 1% of the airborne release was less than 10 μm aerodynamic equivalent diameter (Sutter 1983). This value was used to estimate the respirable-sized fraction available for release. The accident is postulated to occur within a nuclear-grade facility with a filtration system having two stages of high-efficiency particulate air filters. The respirable fraction of material released to the atmosphere of the building is estimated to be 2.5×10^{-9} of the source. The theoretical densities of the salt forms are 3.98 and 4.24 g/cm^3, but the capsule contents are compacted to 75% of this value, and consequently a value of 3 g/cm^3 was used in the calculations. The mass of the source is estimated to be 2.2×10^3 g, the respirable fraction is 2.2×10^1 g, and the atmospheric release is calculated to be 5.5×10^{-6} g.

Other accidents considered as part of this analysis included: a capsule dropped in the water basin; hydrogen accumulation and explosion; loss of filtration during routine processing activities; fire in the facility; capsule failure in the storage basin; and loss of services or power.

H.3.4 TRU-Contaminated Soil Sites

The TRU-contaminated soil sites consist of cribs, trenches, ponds, ditches, reverse wells, French drains, and other areas that have had liquids discharged to them. Under the geologic disposal alternative the radioactive contaminants of the TRU liquid disposal sites would be retrieved through use of a large mobile waste retrieval facility. The retrieved contaminated soil would be transported to a facility to convert it to a chemically inert, physically stable, basalt-like slag that would meet repository requirements for ultimate geologic disposal of immobilized waste forms.

The upper-bound accident postulated for the geologic disposal alternative of the TRU-liquid soil sites is fire and explosion that would occur during processing of the waste at the facility. While the facility design has not yet been selected, certain aspects of the process structure can be anticipated. For purposes of the accident analysis, a slagging pyrolysis incinerator was assumed to be used; the incinerator was presumed to consist of two main components, a gasifier and a secondary combustion chamber. The gasifier would have three zones: drying, pyrolysis, and combustion. The secondary chamber would complete the combustion of the off gas, which is then cooled and filtered. The slag generated by the process would be poured into molds, assayed, and prepared for transport to the geologic repository. The upper-bound accident involves a fire and explosion of contaminated materials in the fuel-rich gasifier portion of the incinerator. Explosions can result from the ignition of clouds of rich fuel mixtures (Orr 1966). It is presumed that a process malfunction allows carbon monoxide to reach the drying section of the unit, where it reacts rapidly with the air introduced with the waste. The mixture deflagrates, resulting in failure of the upper structure portion of the gasifier due to overpressurization. The facility is breached in the explosion.

The explosion generates a 1,000 m^3 cloud filled with radioactive particles; Mishima (1975) estimated that explosion-generated clouds can attain a quasi-stable concentration of particles of 100 mg/m^3. The particles in the cloud are considered to have a size distribution typical of particles in the secondary combustion chamber off gas, with 50% in the fraction 10 micrometer aerodynamic equivalent diameter and less, as suggested by the work of Christian et al. (1978) and Kirstein et al. (1979). The total release of respirable particles is 50 g. Only a portion of the release will be TRU-contaminated soil; for purposes of this analysis, however, it is assumed that all of the respirable-sized release is TRU-contaminated soil.

Other accidents considered as part of this analysis included: a battery-generated hydrogen explosion during waste recovery; spills of contaminated soil; a fire in the mobile waste retrieval facility; filter failure in the facility; criticality in the slagging pyrolysis incinerator facility; and a slag spill as the molten waste was cast.

H.3.5 Pre-1970 Buried Suspect TRU-contaminated Solid Waste

Between 1944 and 1970, TRU-contaminated waste, including laboratory supplies, clothing, tools, etc., was packaged and buried in specially constructed "alpha" trenches. For this EIS, the site was defined as a TRU solid waste burial ground if the concentration of the contents of containers at that location was estimated to exceed 100 nCi TRU/g. Under the geologic disposal alternative, buried TRU solid waste would be retrieved and processed using procedures similar to those proposed for the TRU-contaminated soil sites. Some additional equipment would be required to process the waste for disposal because packaged wastes are involved. This additional equipment would include waste sizers that would perform sawing, shearing, hammering and bending operations.

The postulated upper-bound accident is the same as that described for the TRU-contaminated soil sites; only the inventory of radionuclides available for release differs.

The other accidents considered as part of this analysis are the same as those described for the TRU-contaminated soil site, except for the additional accidents that account for the extra processing steps: leaks from a breached waste drum; a pressurized release from a waste drum; the spread of surface contamination; a fire in a waste container; and release of waste from a dropped container.

H.3.6 Retrievably Stored and Newly Generated TRU

TRU waste generated since 1970 has been stored so that it can be retrieved. If the surface dose rate exceeded 200 mR/yr, the waste was classified as remote-handled and stored in caissons or packaged for direct shipment off site. If the TRU waste was unsuitable for storage on asphalt pads or in caissons because of size, chemical composition, security requirements, or surface radiation, it was packaged in metal or reinforced wooden boxes or concrete and stored in an "alpha" trench. Under the geologic disposal alternative, the remote-handled TRU in caissons would be mechanically retrieved using an airtight, double-shelled structure installed over the caissons. The waste is retrieved using a grappler housing with a telescoping articulated boom. A conveyor system is used to transfer remote-handled casks containing retrieved waste.

Waste placed in retrievable storage trenches and above-ground buildings is known as contact-handled TRU. It is stored free of external contamination and packaged to maintain integrity for a minimum of 20 years. It is packaged so that the waste can be retrieved in an open environment without generating airborne release of radioactivity (Rockwell 1985). The soil overburden will be removed using conventional equipment and/or hand digging as required. Once the overburden is removed, the packaged waste will be removed by a forklift or crane.

The upper-bound accident is postulated to be the explosive release of contact-handled TRU waste due to the buildup of radiolytic gases. The waste is assumed to be stored in a 55-gal drum and have a density of 0.96 g/cm^3. The container is assumed to release its contents at a pressure of 50 psi. Based on experimental studies with depleted uranium dioxide powder (used as a TRU surrogate), 1% of the contents is estimated to become airborne at this release pressure (Sutter 1983, Table B.1). With a source of 2×10^5 g, the estimated airborne release for this event is 2×10^3 g.

Other accidents considered as part of this analysis included: a pressurized release from a remote-handled TRU waste container; spills of material from ruptured or breached packages of remote or contact TRU containers; spread of surface contamination from waste packages; punctures of contact TRU containers during retrieval/handling operations; fire in a waste container; equipment failure; a range fire during retrieval operations; explosion in the slagging pyrolysis incinerator facility during processing of the remote-handled TRU; and a handling accident at the processing facility that includes spill of the waste.

H.4 DESCRIPTION OF UPPER-BOUND ACCIDENTS FOR THE IN-PLACE STABILIZATION AND DISPOSAL ALTERNATIVE

Several operations are required to process each of the six waste forms as part of the in-place stabilization and disposal alternative, as depicted in Table H.12. Twenty separate dose calculations were performed to analyze the potential radiological impact from the disposal of the Hanford defense wastes under the in-place stabilization and disposal alternative. This section contains a brief description of the waste-processing operations used during the upper-bound accident.

H.4.1 Existing Tank Wastes

During in-place disposal, the contents of existing tank wastes are either dried to achieve stability or retrieved through hydraulic sluicing. The single-shell tank wastes would be dried. Residual liquor and other liquid waste from the double-shell tanks would be retrieved hydraulically. Wastes with high concentrations of organic complexes would be treated to destroy or remove the organics and then converted to a cementitious grout. The waste would be processed through use of a transportable grout facility, and the grouted wastes would be disposed of in near-surface vaults. The tank-dome voids above the waste in

TABLE H.12. Waste-Processing Steps for the In-Place Stabilization Alternative

Existing Tank Waste	Newly Generated and Future Tank Waste	Strontium and Cesium Capsules	TRU-Contaminated Soil Sites	Pre-1970 Buried Suspect TRU-Contaminated Solid Waste	Retrievably Stored and Newly Generated TRU Wastes
Leave waste in single-shell tanks. Retrieve 99.95% of waste from double-shell tanks.	Retrieve 99.95% of wastes in double-shell tanks. Separate and encapsulate cesium, and dispose of as outlined for the strontium and cesium capsules, right.	Package capsules in canisters for near-surface disposal onsite.	Wastes remain disposed of in place.	Wastes remain disposed of in place.	Dispose of stored TRU waste in place. (First remove building-stored waste to burial ground.)
↓	↓	↓	↓	↓	↓
Treat double-shell tank waste as necessary to immobilize in grout; place grout in near-surface vaults or in existing tanks.	Immobilize the residual waste as grout and dispose of in near-surface vaults or in existing tanks.	Place canisters in near-surface caissons after cooling.	Fill voids with grout to limit subsidence.	Compact in-place, disposed wastes as needed. Fill voids in old TRU caissons and other sites with grout to control subsidence.	Compact waste, fill voids with grout as needed for subsidence control, provide the protective barrier, emplace markers, and record.
↓	↓	↓	↓	↓	
Fill tanks sufficiently with appropriate material to limit subsidence, and seal all accessible penetrations.	Fill tanks sufficiently with appropriate material to limit subsidence, and seal all accessible penetrations.	Cover canister disposal area with the protective barrier, emplace markers, and record.	Cover all sites with the protective barrier, emplace markers, and record.	Cover all TRU buried solid waste sites with the protective barrier, emplace markers, and record.	
↓	↓				
Leave soil contaminated from tank leaks in place.	Cover grout disposal waste sites and tanks with the protective barrier, emplace markers, and record.				
↓					
Cover tank farms and grout disposal sites with the protective barrier, emplace markers, and record.					

both the single-shell and double-shell tanks would be filled and sealed, and the tank farms would be covered with soil and the protective barrier and marker system.

Throughout the disposal operation, the potential for accidental release of radioactive material exists. The accident with the greatest release of radioactive material was postulated to occur during drying of the salt cake and sludge from the single-shell tank wastes. This accident and its consequences are the same as described in Section H.3.1 for the geologic disposal alternative.

Other accident scenarios considered as part of the analysis of operational accidents for disposal of the existing tank wastes included: loss of filtration; loss of service power; equipment failure; pressurized release during hydraulic sluicing of the double-shell tank wastes; pipe breaks; failure of the air bubble dome; leaks; and explosions.

H.4.2 Future Tank Wastes

Under the in-place stabilization and disposal alternative, the future tank waste will be disposed of by removing as much of it as practicable by hydraulic sluicing. The upper-bound accident developed for this is the same as that described for geologic disposal of future tank waste and involves a release of waste during hydraulic retrieval of the tank contents for processing. The releases and radiological consequences of the event are the same as those described in Section H.3.2.

Other accidents reviewed as part of the analysis for the future tank wastes included: pressurized release of liquid due to a stuck diversion valve; slurry spill during hydraulic retrieval; loss of services or power; loss of filtration; cesium ion-exchange fire during processing of the waste for cesium removal; filter fire during processing; and rupture of an ion-exchange column during waste processing.

H.4.3 Strontium and Cesium Capsules

The strontium and cesium capsules are to be stored in the Waste Encapsulation and Storage Facility for 20 to 40 years. At that time the heat generated by the capsules will be low enough to permit passive cooling of the encapsulated waste and disposal in a drywell storage facility. Canisters will be transported to the drywell disposal area by a shielded-cask transporter that also lowers the casks into the drywell and discharges sand into the space above the canister to fill the drywell.

The upper-bound accident postulated for disposal of the canisters under the in-place stabilization and disposal alternative involves misalignment of the capsule in the drywell. It is postulated that the transporter does not correctly align the waste capsule in the drywell and moves with the capsule still partially in the transporter. The transporter shears the capsule and causes subdivision and dispersion of the particles generated. The operation is performed in the open without any enclosure over the drywell, and any material released would be directly to the atmosphere.

The dispersion value given in DOE (1982) was a release fraction of 1×10^{-6} for a cover block drop. The Mishima et al. (1986 p. 5.32) value for a dropped shipping container was

1×10^{-5}. Since the area affected is less than either the container or cover block drop, the smaller value of 1×10^{-6} is used. As noted in Section H.3.3, one capsule contains 2.2×10^{3} g, so the release is 2.2×10^{-3} g.

Other accidents considered as part of this analysis included: capsule rupture; hydrogen explosion; loss of filtration during retrieval; fire; loss of services or power during retrieval and packaging; and package element failure during drywell storage.

H.4.4 TRU-Contaminated Soil Sites

Grouting is the only disposal operation associated with the in-place stabilization and disposal alternative for the TRU-contaminated soil sites. The waste sites are to be surveyed to determine radiation and contamination levels and to determine subsidence potential. Subsidence control involves the completion of a geophysical survey of the sites to identify those (typically cribs) that have a high potential for subsidence problems. These sites are stabilized by injecting a cementitious grout into the soil. The injection equipment includes mixing tanks, proportioning transfer pumps, hoses, and pneumatic drills. After grout injection, a construction crew must trim vents and feed piping with power saws while under a tent-like containment structure. The equipment and personnel are transported by heavy-duty trucks. Grout will be injected into some sites, such as French drains, cribs, settling tanks, and reverse wells, for subsidence control. Other sites, such as abandoned ponds, trenches, and ditches, will be filled before covering. Following grout injection, the protective barrier and marker system would be applied to the site (Rockwell 1985).

The upper-bound accident postulated for this disposal option is collapse of a void initiated by site-stabilization equipment. The release is based on work by Murphy and Holter (1980); they postulated that an earthmover could be engulfed in a 90-m^3 void space, which would disturb 45 m^3 of waste for an hour. In the event postulated by Murphy and Holter, an atmospheric fractional release of 3.6×10^{-5} was used. Using a soil density of 1.8 g/cm^3, the 45 m^3 of disturbed waste represents a quantity of 8.1×10^{7} g, and the calculated release from the accident would be 2.9×10^{3} g. Only a small portion of Hanford soil 0.088% (Sutter 1980), will be in the respirable size range, which amounts to a release of 2.6 g of respirable material.

Other accidents considered as part of this analysis included: accidental ejection of the grout during injection activities; excavation of contaminated soil; fire; and thermal reaction of the waste with the grout.

H.4.5 Pre-1970 Buried Suspect TRU-Contaminated Solid Waste

The pre-1970 buried suspect TRU-contaminated solid waste burial grounds will be stabilized as required. Caissons containing TRU waste would be immobilized in place by filling them with grout or other stable fillers. The area would then be covered and marked. Sites other than caissons that contain TRU wastes would be subject to subsidence-control measures. One method proposed is to inject piles into identified waste zones by using a diesel-powered vibratory hammer/extractor attached to a vibratory crane. Piles would be driven through the waste zone and then withdrawn; contaminated piles would be redriven to grade and left in place. Finally, a protective barrier and marker system would be placed over the site.

waste zone and then withdrawn; contaminated piles would be redriven to grade and left in place. Finally, a protective barrier and marker system would be placed over the site.

The postulated upper-bound accident is the same as that described for the TRU-contaminated soil sites; only the inventory of radionuclides available for release differs.

The other accidents considered as part of this analysis included: penetration of waste during injection of a pile into the site for stabilization; range fire at the site; excavation of contaminated soil during stabilization activities; and criticality due to changes in fissile geometry during subsidence operations.

H.4.6 Retrievably Stored and Newly Generated TRU Waste

TRU waste generated since 1970 has been stored so that it can be retrieved. If the surface dose rate exceeded 200 mR/hr, the waste was classified as remote-handled (RH) and stored in caissons or packaged for direct shipment off site. If the TRU waste was unsuitable for asphalt pad or caisson storage because of size, chemical composition, security requirements, or surface radiation, it was packaged in reinforced wooden boxes, concrete or metal boxes and stored in an "alpha" trench. Under the in-place stabilization and disposal alternative, any TRU solid waste packages stored in above-grade facilities would be buried. All retrievably stored TRU solid waste would be treated the same as pre-1970 solid waste and would undergo subsidence control measures and barrier placement described in the previous section.

The upper-bound accident postulated for the waste involves the breach of a waste container during burial operations. If the accident occurs when the container is in the open, a large release of material could occur. Based on the experimental work of Sutter (1980), 5% of the spilled waste could be entrained if the ambient winds are at the Hanford average of 7.6 mph (Stone, Jenne and Thorp 1972). As noted previously, moderate wind speeds were chosen to maximize the combination of resuspension and airborne concentration of the radionuclides transported off site during the accident. The waste, in 55-gal drums, with a density of 0.96 g/cm, has a total mass of 1×10^5 g. The total amount released to the atmosphere is calculated to be 1×10^3 g.

Other accidents considered as part of this analysis included: void space collapse during subsidence control operations; punctures of waste containers; fire; and equipment impacting the waste.

H.5 DESCRIPTION OF OPERATIONAL ACCIDENTS FOR THE REFERENCE DISPOSAL ALTERNATIVE

The reference alternative would combine aspects of the geologic disposal and in-place stabilization and disposal alternatives to provide a third approach to the disposal of the wastes. The processing of the waste for disposal is depicted in Table H.13. (Twenty-seven separate dose calculations were performed to analyze the potential radiological impact from the disposal of the Hanford defense wastes under the reference disposal alternative.) This section briefly describes the waste processing operations used and the facility involved during the upper-bound accident. Those accidents that have already been presented elsewhere in the appendix will not be discussed again; however, they are listed in Tables H.5 and H.6 along with the other upper-bound operational releases.

TABLE H.13. Waste-Processing Steps for the Reference Disposal Alternative

Existing Tank Waste	Newly Generated and Future Tank Waste	Strontium and Cesium Capsules	TRU-Contaminated Soil Sites	Pre-1970 Buried Suspect TRU-Contaminated Solid Waste	Retrievably Stored and Newly Generated TRU Wastes
Leave waste in single-shell tanks. Retrieve 99.95% of waste from double-shell tanks.	Retrieve 99.95% of wastes in double-shell tanks. Separate high-activity and TRU waste for vitrification and geologic disposal.	Package capsules in canisters for repository disposal.	Wastes remain disposed of in place.	Most wastes remain disposed of in place. Retrieve TRU wastes not on 200 Area plateau.	Retrieve TRU waste and sort and package it to WIPP waste acceptance criteria.
↓	↓	↓	↓	↓	↓
Separate the TRU and high-activity fraction of the double-shell tank wastes, immobilize as glass, and dispose of in a repository. Immobilize the low-activity fraction of double-shell tank waste as grout, and dispose of in near-surface vaults.	Immobilize the residual waste as grout and dispose of in near-surface vaults or in existing tanks.	Send canisters to geologic repository for disposal.	Fill voids with grout to limit subsidence.	Compact in-place, disposed wastes as needed. Fill voids in old TRU caissons and other sites with grout to control subsidence.	Package and treat waste as required, and send it to WIPP.
↓	↓		↓	↓	↓
Fill tanks sufficiently with appropriate material to limit subsidence, and seal all accessible penetrations.	Fill tanks sufficiently with appropriate material to limit subsidence, and seal all accessible penetrations.		Cover all sites with the protective barrier, emplace markers, and record.	Cover all TRU buried solid waste sites with the protective barrier, emplace markers, and record.	Dispose of residual waste as low-level waste.
↓	↓			↓	
Leave soil contaminated from tank leaks in place.	Cover grout disposal waste sites and tanks with the protective barrier, emplace markers, and record.			As required, package and treat the retrieved waste and send it to WIPP.	
↓					
Cover tank farms and grout disposal sites with the protective barrier, emplace markers, and record.					

H.5.1 Existing Tank Wastes

The reference disposal alternative calls for disposing of the single-shell tank wastes in a manner identical to that described for the in-place stabilization and disposal operation. The single-shell tank wastes will be dried. This disposal method and the upper-bound accident for the reference disposal alternative are the same as stated in Section H.4.1. Operations for disposal of the contents of the double-shell tanks are similar to those discussed in Section H.3.1. The waste disposal steps include hydraulic retrieval, sludge-washing, high-level and TRU separation, vitrification, and repository disposal.

Other accident scenarios considered as part of this analysis of operational accidents for disposal of the existing tank wastes are the same as those discussed in previous sections.

H.5.2 Future Tank Wastes

The reference alternative disposal option for future tank waste involves geologic disposal of the high-activity waste. Hydraulic sluicing would be used to empty the waste tanks. The waste would be processed to separate the solids and liquids, and the cesium would be removed from the neutralized current acid waste before grouting. Strontium and TRU elements from solid/liquid separation operations would be contained primarily in the waste sludge. The cesium concentrate and the sludge would be vitrified. Accidental releases from these operations were examined in Section H.3.2. The upper-bound accident is the same as described in that section.

H.5.3 Strontium and Cesium Capsules

The strontium and cesium capsules would be disposed of in the manner described under the geologic disposal alternative. The upper-bound accident and the other accidents considered as part of the analysis are the same as those described in Section H.3.3.

H.5.4 TRU-Contaminated Soil Sites

The TRU-contaminated soil sites would, under the reference alternative, be disposed of in the same manner described for the in-place stabilization and disposal alternative. The upper-bound accident and the other operational accidents analyzed as part of this effort are the same as those described in Section H.4.4.

H.5.5 Pre-1970 Buried Suspect TRU-Contaminated Solid Waste

The pre-1970 buried TRU solid waste sites would, under the reference alternative, be disposed of in the same manner described for the in-place stabilization and disposal alternative. The upper-bound accident and the other operational accidents analyzed as part of this effort are the same as those described in Section H.4.5.

H.5.6 Retrievably Stored and Newly Generated TRU Waste

For the reference alternative, retrievably stored and newly generated TRU waste would be sent to a geologic repository. The waste would be processed in the same way as in the geologic disposal alternative, except for the remote-handled TRU waste. Since only the remote-handled TRU waste would be processed, a smaller facility would be used. The waste-processing facility proposed for the geologic disposal alternative was sized to accommodate TRU-contaminated soil sites and pre-1970 TRU solid waste burial grounds.

The remote-handled TRU waste would be processed in a facility that provides remote handling, and contains hot cells for size reduction, immobilization, and packaging. A remote-handled waste retrieval and packaging facility would include specific processes required to immobilize and package the waste (Rockwell 1985). However, the immobilization process is not identified, so no releases can be developed for this operation. The upper-bound accident for the reference alternative disposal of newly generated TRU remains the same as described in Section H.3.6.

H.6 OPERATIONAL ACCIDENTS FOR THE PREFERRED ALTERNATIVE

Under the preferred alternative near-term disposal of existing and newly generated and future double-shell tank waste, strontium and cesium capsules and retrievably stored and newly generated TRU waste, would be disposed of in a geologic repository according to the operations described for the reference alternative. Operational accidents for the implementation of the preferred alternative for a given waste class would be the same as those described earlier for the reference alternative. Disposal decisions have been deferred on the remainder of the waste classes, and they will remain as stored (single-shell tank waste) or disposed of without further enhanced long-term protection until completion of further development and evaluation.

H.7 DESCRIPTION OF OPERATIONAL ACCIDENTS FOR THE NO DISPOSAL ACTION ALTERNATIVE

Under the no disposal action, the wastes are placed in continued storage; this alternative does not implement a long-term solution for permanent disposal of the radioactive wastes. The wastes would continue to be stored essentially as they are now for the indefinite future. The waste-handling operations would include storage, necessary remedial actions, and waste surveillance. The waste processing steps for this alternative are shown in Table H.14; they involve the double-shell wastes (both existing and future tank wastes), the strontium and cesium capsules and all TRU wastes. With the exception of the existing tank wastes and the retrievably stored and newly generated TRU, upper-bound accidents have been described in Sections H.3.2 and H.3.3. For the existing tank wastes, the ferrocyanide explosion is no longer postulated to occur, as the single-shell tank wastes are left undisturbed. The dominant accident for this waste class then becomes the pressurized release from hydraulic removal of the existing double-shell tank wastes. For the retrievably stored and newly generated TRU, the collapse of a void space, similar to that described for the other TRU sites, is postulated to occur during subsidence control.

TABLE H.14. Waste-Processing Steps for the No Disposal Action Alternative

Existing Tank Waste	Newly Generated and Future Tank Waste	Strontium and Cesium Capsules	TRU-Contaminated Soil Sites	Pre-1970 Buried Suspect TRU-Contaminated Solid Waste	Retrievably Stored and Newly Generated TRU Wastes
Leave waste in single-shell tanks; retank double-shell tank waste every 50 yr.	Retank double-shell waste every 50 yr.	Package capsules in canisters as necessary for onsite dry storage.	Leave sites as disposed of.	Leave sites as disposed of.	Leave waste as stored.
↓	↓	↓	↓	↓	↓
Monitor and maintain tanks, filling single-shell tank domes and unused old double-shell tank domes as required to prevent collapse and maintain surface.	Monitor and maintain tanks, filling unused, old double-shell tanks as required to prevent collapse.	Store canisters in near-surface caissons.	Monitor sites and continue site maintenance.	Monitor site and continue site maintenance.	Monitor waste and continue to maintain.
↓		↓			
Leave soil contaminated from tank leaks in place.		Monitor caissons and continue caisson maintenance.			

H.8 REFERENCES

Braden, D. E., C. B. Foster, J. R. Houston, G. R. Kiel, R. A. Watrous and R. A. Zinsli. 1971. Safety Analysis Report, Waste Encapsulation and Storage Facilities. ARH-1986, Atlantic Richfield Hanford Company, Richland, Washington.

Christian, J. D., R. C. Girton, B. E. Kirstein and D. T. Pence. 1978. R&D for an Off-Gas Treatment System for a Slagging Pyrolysis Radioactive Waste Incinerator. Final Report for Phase II, SAI 78-904-LJ, Science Applications, Inc., La Jolla, California.

Department of Energy (DOE). 1982. Final Environmental Impact Statement--Defense Waste Processing Facility, Savannah River Plant, Aiken, South Carolina. DOE/EIS-0082-Final, Office of Defense Waste and Byproducts Management, Washington, D.C.

Department of Energy (DOE). 1986. Environment, Safety, and Health Program for Department of Energy Operations. DOE Order 5480.1B, Chapter XI, Operational and Environmental Safety Division, Washington, D.C.

Hayward, W. M., and R. J. Jensen. 1980. Environmental Aspects of Long-Term Management Alternatives for High-Level Defense Waste at the Hanford Site. RHO-LD-140, Rockwell Hanford Operations, Richland, Washington.

Kirstein, B. E., W. J. Paplawsky, D. T. Pence, B. D. Snow and M. E. Sparth. 1979. R&D for an Off-Gas Treatment System for a Slagging Pyrolysis Radioactive Waste Incinerator. SAI-132-79-764-LJ, Science Applications, Inc., San Diego, California.

Martin, E. C. 1985. Complexant Stability Investigation. Task 2. Organic Complexants. PNL-5453, Pacific Northwest Laboratory, Richland, Washington.

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.

Mishima, J. 1975. "Data Useful in the Evaluation of Airborne Plutonium from Postulated Accident Scenarios." Appendix F in Considerations in the Assessment of the Consequence of Effluents from Mixed Oxide Fuel Fabrication Plants. BNWL-1697, Pacific Northwest Laboratory, Richland, Washington.

Mishima, J., S. L. Sutter, K. A. Hawley, C. E. Jenkins and B. A. Napier. 1986. Potential Radiological Impacts of Upper-Bound Operational Accidents During Proposed Waste Disposal Alternatives for Hanford Defense Waste. PNL-5356, Pacific Northwest Laboratory, Richland, Washington.

Murphy, E. S., and G. M. Holter. 1980. Technology, Safety and Costs of Decommissioning a Reference Low-Level Waste Burial Ground. NUREG/CR-0570, Pacific Northwest Laboratory, Richland, Washington.

Napier, B. A. 1981. Standardized Input for Hanford Environmental Impact Statements, Part I. PNL-3509 PT 1, Pacific Northwest Laboratory, Richland, Washington.

Orr, C., Jr. 1966. Particulate Technology. McMillan Company, New York.

Perry, J. H., ed. 1984. Chemical Engineers' Handbook. 6th Edition, McGraw-Hill Book Company, New York.

Price, K. R., J. M. V. Carlile, R. L. Dirkes and M. S. Trevathan. 1984. Environmental Surveillance at Hanford for CY 1983. PNL-5038, Pacific Northwest Laboratory, Richland, Washington.

Rockwell Hanford Operations. 1980. Technical Aspects of Long-Term Management Alternatives for High-Level Defense Waste at the Hanford Site. RHO-LD-141, Richland, Washington.

Rockwell Hanford Operations. 1985. Hanford Defense Waste Alternatives: Engineering Support Data for the HDW-EIS. RHO-RE-ST-30 P, Richland, Washington.

Sommer, D. J., R. G. Rau and D. C. Robinson. 1981. Population Estimates for the Areas Within a 50-Mile Radius of Four Reference Points on the Hanford Site. PNL-4010, Pacific Northwest Laboratory, Richland, Washington.

Steindler, M. J., and W. B. Seefeldt. 1980. "A Method for Estimating the Challenge to an Air Cleaning System Resulting from an Accidental Explosive Event." In Proceedings of the 16th DOE Nuclear Air Cleaning Conference. CONF-801038, Vol. 2, National Technical Information Service, Springfield, Virginia.

Stone, W. A., D. E. Jenne and J. M. Thorp. 1972. Climatography of the Hanford Area. BNWL-1605, Pacific Northwest Laboratory, Richland, Washington.

Strenge, D. L., B. A. Napier, R. A. Peloquin and M. G. Zimmerman. 1980. ALLDOS--A Computer Program for Calculation of Radiation Doses from Airborne and Waterborne Releases. PNL-3524, Pacific Northwest Laboratory, Richland, Washington.

Sutter, S. L. 1980. Potential Airborne Release from Soil Working Operations in a Contaminated Area. PNL-3498, Pacific Northwest Laboratory, Richland, Washington.

Sutter, S. L. 1983. Aerosols Generated by Releases of Pressurized Powders and Solutions in Static Air. NUREG/CR-3093 (PNL-4566), Nuclear Regulatory Commission, Washington, D.C.

THIS PAGE INTENTIONALLY
LEFT BLANK

APPENDIX I

ANALYSIS OF IMPACTS FOR TRANSPORTATION OF HANFORD DEFENSE WASTE

THIS PAGE INTENTIONALLY
LEFT BLANK

APPENDIX I

ANALYSIS OF IMPACTS FOR TRANSPORTATION OF HANFORD DEFENSE WASTE

This appendix presents an analysis of impacts associated with transportation of Hanford high-level defense wastes to a basalt repository at Hanford and to a hypothetical repository location off site and transportation of transuranic waste to the Waste Isolation Pilot Plant (WIPP) in New Mexico. Also described here are: 1) the regulations governing transport activities and the organizations responsible for them, 2) the volume of waste and the numbers of shipments to be transported that are associated with the alternatives which use repositories, 3) the packagings and packaging systems used for the wastes, 4) the cost of transporting wastes, 5) the radiological and nonradiological effects of transporting wastes under both normal and accident conditions, and 6) emergency response provisions.

The nonradiological effects of transporting Hanford defense wastes are used in Appendix L, where the nonradiological impacts associated with each waste management alternative are tabulated and summed. Radiological impacts of transportation are used in Chapter 5 in calculation of the total radiological impacts associated with each alternative. Transportation costs are also used in Chapter 5 to calculate total waste disposal system costs.

I.1 APPLICABLE REGULATIONS AND RESPONSIBLE ORGANIZATIONS

The transportation of Hanford defense wastes to an offsite repository would comply with the regulations and orders promulgated by one or more of four Federal agencies: 1) the Department of Transportation (DOT), 2) the Department of Energy (DOE), 3) the Nuclear Regulatory Commission (NRC), and 4) the Interstate Commerce Commission (ICC). These agencies have developed comprehensive regulations covering the performance of the shipping packagings, vehicle safety, routing of shipments, physical protection, and economics. The DOE has also developed applicable transportation requirements, which are set forth in DOE 5480.3 (DOE 1985). The following sections briefly discuss the regulations and organizations responsible for the safe transport of radioactive wastes in the United States.

I.1.1 Applicable Regulations

Regulations for the safe transportation of radioactive materials are designed to protect the public from the potential consequences of loss or dispersal of radioactive materials during transit as well as from routine (nonaccident) radiation doses. These regulations ensure safety through standards for packaging, handling, and routing of shipments. Specific regulations that apply to offsite shipments of Hanford defense wastes are found in the Code of Federal Regulations (CFR) under the following headings:

49 CFR 107 Rule-making Procedures for the Materials Transportation Bureau

49 CFR 171 General Information, Regulations, and Definitions

49 CFR 172 Hazardous Materials Table and Hazardous Materials Communications Regulations

49 CFR 173 Shippers--General Requirements for Shipments and Packagings

49 CFR 174 Carriage by Rail

49 CFR 177 Carriage by Public Highway

49 CFR 178 Shipping Container Specifications

The following subsections present key elements of the regulations pertaining to shipment of Hanford defense wastes.

I.1.1.1 Packaging

Packaging, as used in this report, is defined as the shipping container for radioactive material. Properly designed, manufactured, and prepared packaging is the primary means for ensuring the safe transport of radioactive materials. Consequently, most of the regulations are concerned with packaging standards.

DOT regulations that apply to shipments of Hanford defense wastes are contained in 49 CFR Part 173. These regulations seek to enhance safety through three key elements: 1) containment of radioactive material, with allowances for heat dissipation if required, 2) shielding from radiation emitted by the material, and 3) prevention of nuclear criticality in fissile materials. These aspects of DOT regulations are addressed in the remainder of this subsection. Regulations allow radioactive materials to be shipped in different types of packagings, depending on the total radioactive hazard presented by the material within the package. Of interest to this study are the Type A and the more durable Type B packagings. All DOE packagings must meet, as a minimum, the design requirements described in 49 CFR 173, Sections 411 and 412. Type B packagings must additionally meet the design requirements for Type B packages specified in 49 CFR 173.413. These Type B design requirements are found in United States Nuclear Regulatory Commission, Rules and Regulations, 10 CFR 71, Subpart E. In addition, all DOE packagings must meet the testing requirements specified in 49 CFR 173.465 for Type A packages and 49 CFR 173.467 for Type B packages. Type B packaging tests are found in 10 CFR 71, Subpart F. As a result, the DOE design and testing criteria meet the same packaging standards as the NRC.

Radioactive materials exceeding the limits for Type A packagings can be shipped only in Type B packagings. These packagings are extremely accident-resistant and must be used for all shipments of high-level (HLW) and transuranic (TRU) wastes from the Hanford Site. Any Type B packaging design placed in service must be certified by either the DOE or NRC to the design and testing standards of the NRC. The DOE may use NRC-certified packagings or may certify their own packagings for Hanford defense wastes only if the packagings satisfy the design and testing standards of the NRC as required by 49 CFR 173.413 and 49 CFR 173.467. In addition to meeting the standards for a Type A packaging, a Type B packaging must be designed to withstand severe hypothetical accident conditions that demonstrate resistance to impact, puncture, fire, and water immersion (10 CFR 71.73). To be acceptable, the Type B packaging must release no radioactivity except for limited amounts of contaminated coolant and gases.

Also, there can be no external radiation dose rate exceeding one rem per hour at one meter from the external surface of the packaging [10 CFR 71.51(a)(2)]. Surface contamination of packagings is limited to specified levels. The method for determining amounts of surface contamination is specified in 49 CFR 173.443.

Radiation allowed to escape from a packaging must be below specified limits that minimize the exposure of the handling personnel and general public. Packages handled only by the shipper and receiver (i.e., shipped in exclusive-use or sole-use vehicles) must be designed so that the following radiation limits are not exceeded (49 CFR 173.441) during normal transport activities:

- 1,000 mR/hr at 1 m from the exterior of the package (in a closed transport vehicle only)
- 200 mR/hr at any point on the external surface of the car or vehicle (in a closed transport vehicle only)
- 10 mR/hr at any point 2 m from the vertical planes projected by the outer lateral surfaces of the car or vehicle; or if the load is transported in an open transport vehicle, at any point 2 m from the vertical planes projected from the outer edges of the vehicle
- 2 mR/hr in any normally occupied position in the car or vehicle; this provision does not apply to private motor carriers under certain conditions.

Criticality standards for packages containing fissile materials are found in 49 CFR 173.451 through 173.459. Packagings used to ship fissile materials must be designed to prevent criticality. The number of such packages shipped together is also limited. Some quantities and forms of fissile materials cannot be made critical under credible conditions and are exempted from special fissile-material requirements.

I.1.1.2 Vehicle Safety

No additional or special vehicle regulations are imposed on carriers of radioactive materials beyond those required for carriers of any hazardous material. Truck safety is governed by the Bureau of Motor Carrier Safety of the DOT, which imposes vehicle-safety standards on all truck carriers (49 CFR 325.386 through 325.398). Along with other functions, the Bureau conducts unannounced wayside inspections of all truck-carrier vehicles and drivers. Several states, including Washington and Oregon, also have truck inspection programs. During the inspection, the condition and loading of the vehicle and the drivers' documents are checked.

Rail cars and trucks carrying HLW and TRU wastes must be placarded in accordance with 49 CFR 172 subpart F. To ensure that cars are in safe condition, DOT regulations in 49 CFR 174 specify that each placarded rail car and each adjacent railcar be inspected by an authorized representative of the carrier company or DOT at each required inspection point. Inspection includes visual examination for obvious defects of the running gear and any leakage of contents.

I.1.1.3 Routing

The DOT's routing regulations, 49 CFR 177.825 (Docket HM-164), were published January 19, 1981, and became effective February 1, 1982. Objectives are to reduce impacts of transporting radioactive wastes, to establish consistent and uniform requirements for route selection, and to identify the role of state and local governments in the routing of radioactive materials. The regulations attempt to reduce potential hazards by avoiding populous areas and minimizing transit times. A carrier or any person operating a motor vehicle carrying a "highway-route-controlled quantity" of radioactive materials is required by Docket HM-164 to use the interstate highway system except when moving from origin to interstate or interstate to destination. Other "preferred highways" may be designated by any state to replace or supplement the interstate highway system. Under its authority, however, to regulate interstate transportation safety, the DOT can prohibit state and local bans and restrictions as "undue restraint of interstate commerce."

All regulations announced by state and local governments have to be consistent with the provisions of Docket HM-164 or they will be preempted. The DOT holds that conflicting requirements among jurisdictions may be unduly restrictive and may increase risks by directing shipments to highways having higher accident rates. State and local requirements will be preempted by Docket HM-164 if they:

- completely prohibit travel between any two points served by highway
- prohibit use of an interstate highway, including prohibition of travel based on time of day, without designation of an equivalent preferred highway as a substitute
- require use of a preferred highway except in accordance with the provisions of the regulation
- require prenotification of state and/or local authorities
- require special personnel, equipment or escort.

The Second Circuit Court of Appeals recently ruled that federal DOT routing regulations were valid, and therefore pre-empted a conflicting ordinance enacted by New York City banning shipment of irradiated reactor fuel through the City on interstate highways. This ruling is expected to set precedent for preemption of a number of state and local ordinances inconsistent with DOT regulations.

The DOT regulation requires carriers to use routes selected to minimize transit time and radiological risk. Carriers transporting Hanford defense wastes will be required to travel on interstate circumferential or bypass routes, if available, to avoid populous areas. Carriers may use interstate or preferred highways that pass through urban areas only if circumferential routes are not available.

No additional regulations are currently proposed for rail transport. Routes are fixed by rail locations, and urban areas cannot be readily bypassed. Thus rail transport of Hanford defense wastes will be similar to that of other non-Hanford loads routinely carried, including hazardous nonradioactive cargoes.

Use of interstate highways will be required for transport of radioactive waste wherever possible. Exceptions are routes between interstate highway access points and points-of-origin and destination. Alternative routes may be proposed by states involved.

Actual transportation routes to the WIPP site from Hanford have not been established. Typical rail and truck routes between the two sites were presented in the final environmental impact statement for the Waste Isolation Pilot Plant (DOE 1980). Under existing DOE procedures, the designated carrier of radioactive material is permitted to select from routes identified as "preferred" by the DOT or the states. Because other geologic repositories have not been selected, specific routes for Hanford wastes to a deep geologic repository have not been established.

Impacts associated with shipments of radioactive materials are generally the radiation doses from the passing radiation source to individuals near the route. A total population dose is obtained by summing the individual doses for the entire exposed group. Little change from route-specific population doses versus generic-route population doses would be expected; both avoid population centers and would likely result in similar values. Individual doses for routine shipments are small (about 2 mrem/yr if the individual were a bystander in a rail yard standing 20 m from the waste cask for every shipment) and could be reduced or avoided by heeding warnings on shipment placards and moving farther away from the radioactive cargo.

Procedures for emergency response are required as discussed in Section I.8. It is the policy of DOE, upon request from any State, Federal, or local authorities, NRC licenses, private organizations or commercial carriers, to provide radiological assistance teams to support State or local authorities as required.

I.1.2 Responsible Organizations

Shipments of Hanford defense wastes to the hypothetical repository locations will involve the four agencies listed in Section I.1. The DOT, NRC, and DOE deal primarily with safety, while the ICC regulations are related to the economics of transportation. Since this report deals primarily with safety, the regulatory function of the ICC will not be discussed.

Some overlaps exist in the responsibilities of the DOT, NRC, and DOE. The DOT has primary responsibility for safety in transporting all hazardous materials, including nuclear materials. The DOE has the authority to design and certify its own packagings to NRC packaging standards for use by government shippers. The NRC, as the regulator of the commercial nuclear industry, is responsible for regulating the Type B packagings used by commercial shippers. Where overlap exists, Memoranda of Understanding (MOU) have been issued between the agencies to define areas of responsibility. For example, the DOE has the authority, granted by a 1973 MOU between the DOT and the Atomic Energy Commission, predecessor to the DOE (38 FR 8486), to certify its own Type B packagings to be used by DOE or its contractors, provided the packagings comply with DOT and corresponding NRC design and test criteria. DOE is currently in the process of procuring a TRUPACT Type B shipping container, which will have NRC certification for use to transport Hanford TRU wastes. DOE is also designing a Type B shipping cask to transport the solidified tank wastes and possibly strontium/cesium capsules. It is not clear at this time whether or not DOE will seek an NRC Certificate of Compliance for this shipping cask.

The DOE, through its management directives and contractual agreements, protects the public health and safety by imposing standards on its offsite transportation activities in accordance with the DOT regulations.

The DOT specifies and enforces regulations intended to guarantee that hazardous materials are properly classified, described, packaged, labeled, placarded, and in proper condition for shipment. The DOT is responsible for enforcing vehicle safety standards, setting allowable radiation levels, and requiring the use of tamper-indicating seals on loaded packagings. The DOT is also responsible for highway routing of radioactive materials (see Section I.1.1.3), and specifies criteria governing the location of radioactive cargoes relative to other materials being shipped. For rail shipment, additional DOT criteria cover the location of cars carrying radioactive cargo in relation to other placarded railcars, the engine, and/or the caboose.

I.2 PACKAGINGS FOR TRANSPORTING HANFORD DEFENSE WASTES

This section describes the shipping containers that would be used to transport processed Hanford defense wastes to a geologic repository. (Such transportation, of course, would not be required for in-place stabilization and disposal or continued storage.) The general waste classifications considered include strontium and cesium capsules, vitrified high-level wastes, and transuranic wastes, all described in detail in Chapter 3 of this EIS. Table I.1 summarizes descriptions of the packagings used to transport these wastes and the number of shipments for each waste type. Additional details are presented in the remainder of this subsection.

I.2.1 Strontium and Cesium Wastes

As stated in Chapter 3, $^{90}SrF_2$ and $^{137}CsCl$ are presently stored in water-filled basins in the Waste Encapsulation Storage Facility (WESF).

In the geologic disposal alternative, strontium and cesium capsules would remain in continued storage in the Waste Encapsulation Storage Facility basins until a repository is available. For this analysis, it is assumed that the capsules would be loaded into canisters for shipment to the repository. The strontium and cesium inventory and radionuclide content data for the year 1995 are used in this appendix (Rockwell 1985).

The shipping canisters will be constructed of 0.3-m-dia carbon steel pipe and will be 2.7 m long. For geologic disposal, an average of three strontium or five cesium capsules

TABLE I.1. **Summary of Packagings and Shipments**[a]

Waste Type	Transport Mode Onsite	Transport Mode Offsite	Shipment Capacity Onsite	Shipment Capacity Offsite	Number of Shipments[b] Onsite	Number of Shipments[b] Offsite
Existing Tank Wastes						
Geologic Disposal	Rail	Rail	7 Canisters[c]		2,829	2,829
Reference Alternative	Rail	Rail	7 Canisters		68	68
Future Tank Wastes						
Geologic Disposal	Rail	Rail	5 Canisters[c]		662	662
Reference Alternative	Rail	Rail	5 Canisters[c]		119	119
Sr/Cs Canisters	Truck	Rail	1 Canister	9 Canisters	509	57
Pre-1970 Buried TRU Solid Wastes	NA	Rail	NA	72 Drums[d]	NA	570
TRU-Contam. Soil Sites	NA	Rail	NA	72 Drums[d]	NA	178
Retrievably Stored TRU Wastes	NA	Truck	NA	36 Drums[e]	NA	1,040
Newly Generated TRU Wastes			NA			
Geologic Disposal	NA	Truck	NA	36 Drums[e]	NA	1,560
Reference Alternative	NA	Truck	NA	36 Drums[e]	NA	1,800

(a) Onsite refers to onsite shipments to a basalt repository; offsite refers to shipments to a hypothetical repository 4,800 km away (for tank wastes and Sr/Cs canisters) or to WIPP (for TRU wastes).
(b) Numbers of shipments were calculated by dividing the numbers of containers produced in each alternative (see Rockwell 1985, Table B-2-1) by the shipment capacities.
(c) Same shipping cask used for offsite shipments. See Figure I.1.
(d) Five railcars are transported per shipment. Total: 360 drums/shipment.
(e) These shipments use the TRUPACT shipping container, which, for purposes of impact calculations, was assumed to hold 36 drums. If the capacity is 24 drums as recent design indicates, the number of shipments would be higher (i.e., 1.5 times). Transportation impacts would also be higher. See Figure I.2.

will be placed in each canister based on allowable canister heat loads. Based on these loading factors, an estimated 509 canisters will be available for shipment (Rockwell 1985).

Repositories at Hanford and elsewhere were considered in this analysis. Transport to a repository at Hanford would be via truck, with one canister carried per shipment. Transport to a repository elsewhere would be via train, with nine canisters carried per shipment. Table I.2 presents estimated radionuclide inventories per shipment of strontium and cesium canisters.

I.2.2 Tank Wastes

The processing of tank waste will produce a borosilicate glass waste form that will be transported to a geologic repository either at Hanford or elsewhere. The glass will be

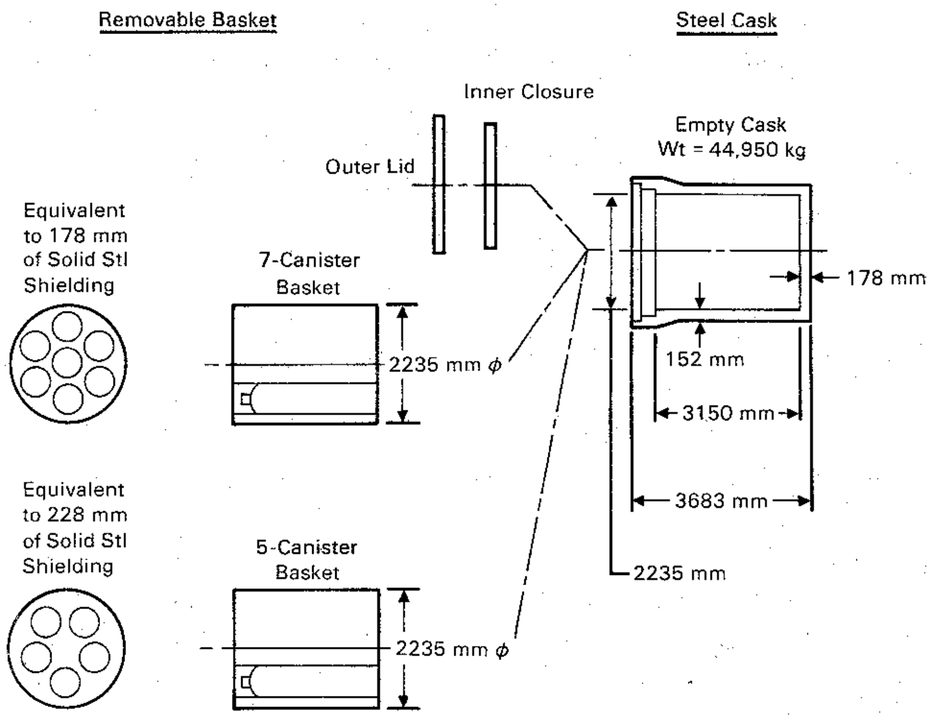


FIGURE I.1. Railroad Cask

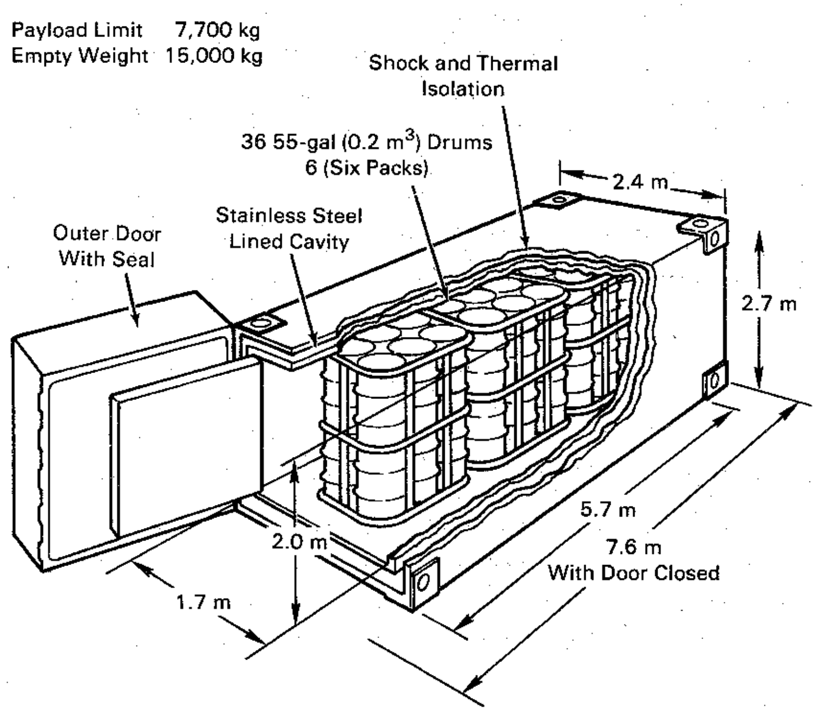


FIGURE I.2. TRUPACT-1 Model Concept

TABLE I.2. Radionuclide Content of Shipments of Strontium and Cesium Canisters[a]

Waste Type/ Nuclide	Radionuclide Content, Curies per Shipment	
	Onsite Truck Shipments	Offsite Rail Shipments
Cs canisters/^{137}Cs	1.7×10^5	1.5×10^6
Sr canisters/^{90}Sr	9.8×10^4	8.6×10^5

(a) See Chapter 3 and Appendix A for detailed waste descriptions.

contained in narrow-mouth, high-integrity, carbon steel canisters holding about 0.62 m^3 of glass. Existing tank waste processed for the geologic disposal alternative will result in about 19,800 canisters of waste glass, while processing of new tank waste will produce about 3,310 canisters. In the reference alternative, 473 canisters will be generated from existing tank waste and 595 canisters from new tank waste. Of these, 463 canisters will be produced from processing of neutralized current acid waste (NCAW) and 132 canisters from processing of Plutonium Finishing Plant (PFP) waste.

For this analysis it is assumed that vitrified waste will be shipped to a repository by rail using a specially designed shipping cask like that shown in Figure I.1. These casks can transport seven canisters of existing tank waste and five canisters of new tank waste or Plutonium Finishing Plant waste. One cask will be used for each shipment. These casks are at the conceptual-design stage and have not been fabricated. Radionuclide inventories for these wastes are shown in Tables I.3 and I.4.

TABLE I.3. Radionuclide Content of Rail Shipments of Existing Tank Wastes[a]

Isotope	Radionuclide Content, Curies per Shipment	
	Geologic Disposal	Reference Alternative
^{90}Sr	2.1×10^4	2.8×10^5
^{99}Tc	1.4×10^1	0
^{137}Cs	7.0×10^3	0
^{151}Sm	3.5×10^2	4.2×10^3
^{237}Np	2.1×10^{-2}	7.0×10^{-1}
^{239}Pu, ^{240}Pu	9.1	1.7
^{241}Pu	2.1×10^1	7.0
^{241}Am	1.4×10^1	4.2×10^2

(a) See Chapter 3 and Appendix A for detailed waste descriptions.

TABLE I.4. Radionuclide Content of Rail Shipments of Future Tank Wastes[(a)]

Isotope	Radionuclide Content, Curies per Shipment: Geologic Disposal	Reference Alternative[(b)]: NCAW	Reference Alternative[(b)]: PFP
^{90}Sr	2.1×10^4	4.4×10^5	0
^{99}Tc	7.5	0	0
^{137}Cs	1.5×10^4	5.0×10^5	0
^{151}Sm	2.9×10^2	4.2×10^3	0
^{239}Pu	6.5	3.5×10^1	9.0×10^1
^{241}Am	9.0×10^1	3.2×10^3	7.0×10^2

(a) See Chapter 3 and Appendix A for detailed waste descriptions.
(b) Two types of vitrified HLW are produced in the reference alternative: 1) from the PUREX Plant and 2) from Plutonium Finishing Plant wastes.

I.2.3 Transuranic Wastes

TRU wastes for shipment, in the geologic disposal alternative or the reference alternative, will result from the processing of TRU-contaminated soils, retrievably stored and newly generated TRU wastes, and pre-1970 buried TRU solid wastes if retrieved. All of these wastes are assumed to be transported to a repository (assumed for calculation purposes to be WIPP) in the case of geologic disposal; only the retrievably stored and newly generated TRU wastes will be transported in the reference alternative.

I.2.3.1 TRU-Contaminated Soil Sites

A total of 33,000 m^3 of TRU-contaminated soils would be retrieved and processed in the geologic disposal alternative. Processing in the Slagging Pyrolysis Incinerator (SPI) would result in a total of 11,000 m^3 of slag to be packaged in Type A or lesser quantities in DOT-specification 17C (55-gal) steel drums and transported to a geologic repository at WIPP.

Transportation of cast slag from the Slagging Pyrolysis Incinerator facility to the TRU waste repository at WIPP will be by train, with 72 drums per rail car and five cars per shipment. Radionuclide inventories per shipment of these wastes are shown in Table I.5.

I.2.3.2 Pre-1970 Buried Suspect TRU-Contaminated Solid Wastes

Unsegregated pre-1970 solid wastes are packaged in several configurations, including 55-gal drums, concrete boxes, and caissons. An estimated 120,000 m^3 of wastes in this category will be retrieved and processed in the Slagging Pyrolysis Incinerator facility, to produce 44,800 m^3 of slag for repository disposal (Rockwell 1985). All of these wastes are assumed to be cast into 55-gal drums before shipment to either an onsite or offsite repository.

TABLE I.5. Radionuclide Content of Truck and Rail Shipments of Wastes from TRU-Contaminated Soil Sites

Isotope	Radionuclide Content, Curies per Shipment, Offsite Rail[a]
^{90}Sr	1.2
^{137}Cs	6.2×10^{-1}
^{239}Pu	3.4
^{241}Pu	2.0
^{241}Am	1.0

(a) Values shown give the radionuclide content of each rail car; five rail cars are shipped in each train.

In this analysis, it is assumed that transport to the offsite waste repository (assumed for calculative purposes to be the WIPP) will be by rail. The cast slag from the Slagging Pyrolysis Incinerator facility will be packaged and loaded into rail cars and shipped about 2,400 km to the WIPP. Each rail car will be loaded with 72 drums, and five rail cars will be transported per shipment. Radionuclide inventories per shipment of these wastes are shown in Table I.6.

TABLE I.6. Radionuclide Content of Truck and Rail Shipments of Pre-1970 Buried TRU Wastes

Isotope	Radionuclide Content, Curies per Shipment, Offsite Rail[a]
^{14}C	3.3×10^{-4}
^{90}Sr	7.9
^{137}Cs	7.9
^{239}Pu	6.9
^{241}Pu	3.9
^{241}Am	2.0

(a) Values shown give the radionuclide content of each rail car; five rail cars are shipped per train.

I.2.3.3 Retrievably Stored and Newly Generated TRU Waste

TRU waste that is currently in retrievable storage has been packaged in several different configurations including:

- 55-gal (210 L) metal drums
- 30-gal (114 L) metal drums
- 1-gal (3.8 L) metal cans (alpha caisson storage only)
- M-III metal bins
- Concrete boxes
- Metal boxes
- Fiberglass-reinforced plywood (FRP) boxes
- Concrete casks
- Specially designed containers.

New regulations will mandate that this waste be repackaged or overpacked before shipment. Newly generated TRU waste will be packaged only in 55-gal drums or in metal boxes. Boxes constructed of fiberglass-reinforced plywood are no longer acceptable storage containers.

Most TRU waste in storage is contact-handled (CH) waste with external dose rates of less than 200 mrem/hr. Also included in the inventory is remote-handled (RH) TRU waste (i.e., caisson waste) contained in 1-gal metal cans. A total of 21.7 m^3 of caisson waste was generated from 1970 through FY 1983. Some CH and RH retrievably stored TRU waste will require processing for the geologic disposal alternative, to be acceptable at WIPP.

Newly generated TRU waste, both CH and RH, will be received starting in FY 1984. A total of 1.2×10^4 m^3 of CH waste and 34 m^3 of RH waste (including 4.0 m^3 of fuel hull waste from the Fast Flux Test Facility) will be received through 1996. Some of the CH and RH newly generated waste will also require processing for the geologic disposal alternative, to be acceptable at WIPP.

Preparation of retrievably stored TRU wastes for geologic disposal will require retrieving, sorting and, possibly, further processing of the wastes. After retrieval, sorting will be necessary to verify that the TRU waste package meets the >100 nCi/g criteria, as well as the waste-form and packaging criteria for the WIPP repository. If the package does not meet these criteria, the waste will be processed before shipment to WIPP.

TRU wastes for shipment to a geologic repository will be packaged in 55-gal drums, which may be banded together in six-packs (see Figure I.2). Transport to the repository will be by truck hauling a TRUPACT packaging system containing a minimum of 24 waste drums (assumed for calculative purposes to be 36 drums). The TRUPACT is illustrated in Figure I.2. Radionuclide inventories per shipment of these wastes are shown in Table I.7.

I.3 METHODS FOR CALCULATING RADIOLOGICAL AND NONRADIOLOGICAL IMPACTS

This section discusses methods used to calculate the radiological and nonradiological impacts of transporting Hanford defense wastes. Round-trip distances are used for each of the three repositories involved. These are 20 km for an onsite basalt repository; up to 9,600 km for a repository in another medium; and for TRU wastes, 4,800 km for the WIPP. A discussion of why this analysis is conservative is presented in the Analytical Methodology section at the front of this volume.

TABLE I.7. Radionuclide Content of Shipments of Retrievably Stored and Newly Generated TRU Wastes

Isotope	Radionuclide Content, Curies per Shipment: Retrievably Stored TRU	Newly Generated: Geologic Disposal	Newly Generated: Reference Alternative
^{90}Sr	2.9×10^{1}	1.7×10^{1}	1.5×10^{1}
^{137}Cs	3.0×10^{1}	1.8×10^{1}	1.7×10^{1}
^{237}Np	7.8×10^{-5}	0	0
$^{239,240}Pu$	2.3×10^{1}	1.6×10^{1}	1.5×10^{1}
^{241}Pu	5.8×10^{1}	7.7×10^{1}	6.7×10^{1}
^{241}Am	1.4	2.2	2.1

I.3.1 Radiological Impacts

The radiological impacts of transporting wastes are calculated for both normal and accident conditions. In normal (incident-free) transport, the package of radioactive material arrives at its destination without releasing its contents. The accident analysis considers the potential release of radioactive material from the package and its associated impacts. Impacts from accidents during transport may or may not occur. Risks are presented in this analysis as expected impacts (i.e., consequences times accident rates).

I.3.1.1 Impacts Resulting from Normal Transport

Radiological impacts during normal transport involve dose to bystanders from radiation emitted by radioactive material packages as the shipment passes by. Even though radiation shields are incorporated into packaging designs (if required by regulations), some radiation penetrates the package and exposes the nearby population to an extremely low dose rate. After the shipment has passed, no further exposure occurs.

The groups exposed to radiation include crew members of trains, truck drivers, those who directly handle waste packages, and the general public--bystanders at truck stops and rail sidings, persons living or working along a route, and nearby travelers (moving in the same and opposite directions). The RADTRAN III computer code was used to calculate exposures to these population groups (Taylor and Daniel 1982; Madsen et al. 1983; Madsen et al. 1986). The RADTRAN III normal population exposure models are illustrated in Figure I.3.

In the population exposure model, the assessment of population dose assumes the packaging or shipping cask is a point source of radiation. The point-source approximation is acceptable for distances between the receptor and source of more than two source-characteristic lengths.[a] At shorter distances, the point-source approximation is conservative; i.e., the doses calculated tend to be higher than those likely to occur.

(a) Source-characteristic length is equal to the largest physical dimension (length, diameter, etc.) of the source.

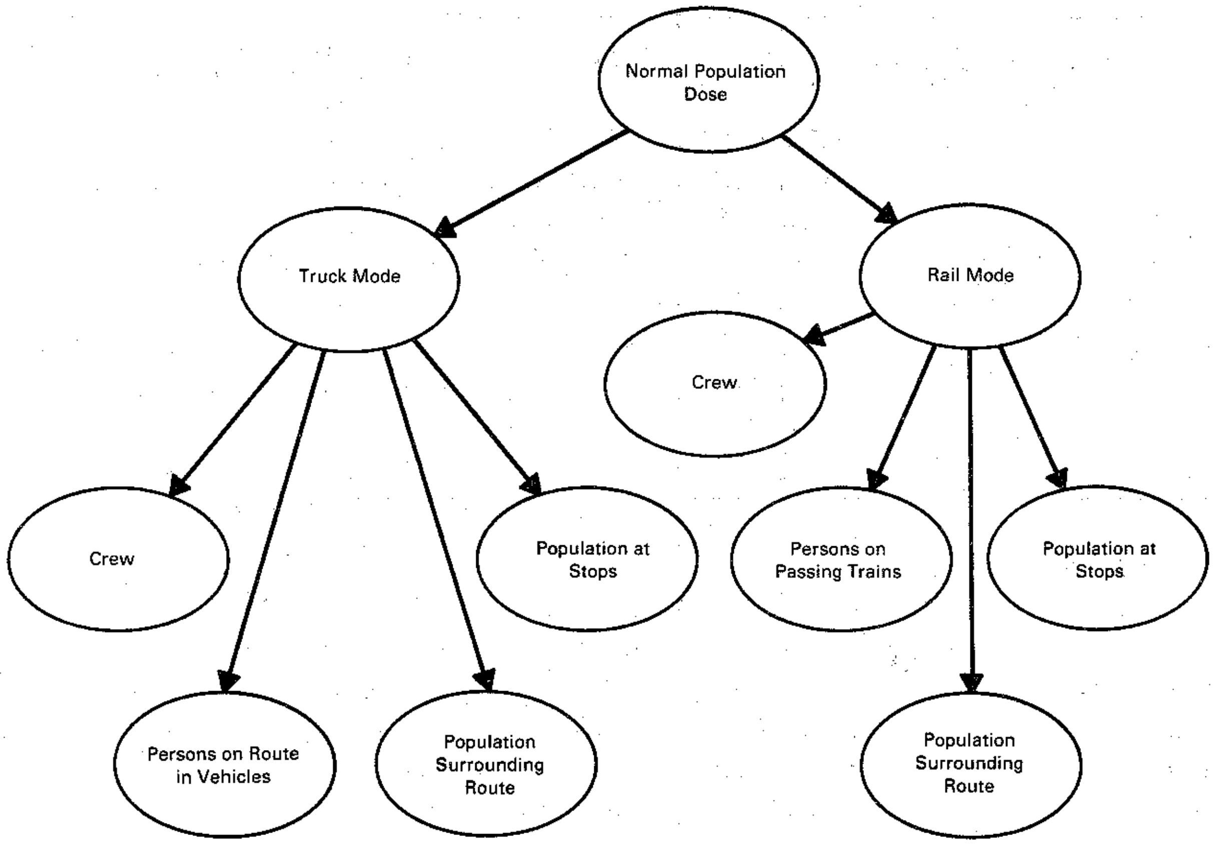


FIGURE I.3. Population Dose Models for Normal Transport Included in RADTRAN III

The basic equation used to calculate the dose rate (D) from a point source, assuming only attenuation and buildup in air and ignoring scatter from the ground, is:

$$D(r) = \frac{QK\ B(r)\ \exp\ (-\mu r)}{r^2}$$

where $D(r)$ = dose rate at distance r, R/hr
$B(r)$ = dose buildup factor for an isotropic source
K = dose rate factor for a unit source strength, R/hr-Ci at one meter
μ = linear attenuation coefficient, m^{-1}
r = distance from the source, m
Q = source strength, Ci.

The equations used to calculate exposures differ among population groups and transport modes (i.e., truck and rail), but their basis in the point-source assumption is the same. Derivations of the various equations are discussed in detail by Taylor and Daniel (1982).

I.3.1.2 Impacts from Accidents Involving Radioactive Wastes

The RADTRAN III computer code was used also to calculate the impacts that result from transport accidents. As previously discussed, the impacts associated with potential transport accidents are expressed as risks. For this appendix, risk is defined as the product of the probability of occurrence of an accident involving radioactive materials times the consequences of an accident. Consequences can be expressed in terms of the health effects from a release of radionuclides from the packagings or the exposure of persons to radiation that could result from damaged package shielding.

RADTRAN III evaluates radiological impact for four pathways: groundshine, cloudshine, food ingestion, and inhalation (see Figure I.4). Cloudshine is the external exposure to radiation from a cloud of radioactive material. Groundshine is external exposure that results from radionuclides deposited on the ground. Inhalation is the exposure pathway to radiation that results from inhaling radioactive materials. Ingestion is the exposure pathway of the population from food that has become contaminated with radioactive material

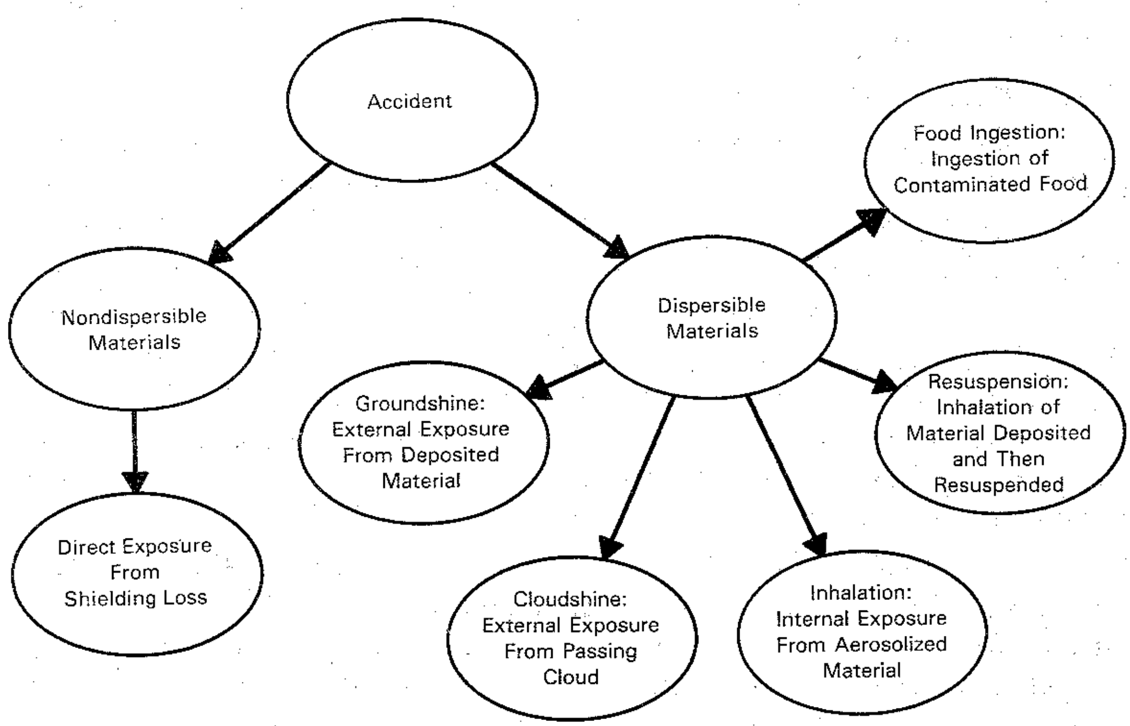


FIGURE I.4. Accident Dose Pathways Considered in RADTRAN III

and then ingested. RADTRAN III assumes that radioactive materials released from a package in an accident are dispersed according to standard Gaussian diffusion models. These models predict downwind airborne radionuclide concentrations and the amount of material deposited on the ground. Radiation doses to human organs are then determined using the calculated airborne radionuclide concentrations and standard dosimetric conversion factors. (See Appendix N for a discussion on converting radiation dose to health effects.) External radiation exposures from ground contamination (groundshine) are calculated using an infinite plane source model (Taylor and Daniel 1982). Radiation doses from groundshine include public exposures for 50 years to the radioactive material deposited on the ground. The model assumes that the contaminated area will be cleaned up to an acceptable residual level, if needed; or, if the contamination is too great, it is assumed that the area will be interdicted. Radiation doses to emergency response personnel and accident-cleanup crews are not included. Population doses from ingestion are estimated with the use of radionuclide transfer fractions which are the relationships between the amount of radioactive material ingested to the amount deposited on the ground.

The probability of an accident that involves radioactive materials is expressed in terms of the expected number of accidents per unit time. The response of the shipping container to the accident environment, and hence, the probability of release or loss of shielding, is related to the severity of the accident. Accidents with severities exceeding design standards for shipping packages (see 10 CFR 71 and 49 CFR 173) could potentially occur, but their probability is extremely small. Thus there is a slight possibility that an accident could occur accompanied by a release or loss of package shielding. Accident rates and probabilities are discussed further in Section I.4.2.

I.3.2 Nonradiological Impacts

The nonradiological impacts of transportation are calculated for both normal and accident conditions. Only the method for calculating impacts from normal, incident-free transportation is described here. The method for calculating impacts from transport accidents is based on accident statistics and is discussed in Subsection I.6.2.

The nonradiological impacts from transportation of radioactive materials are the same as those resulting from transport of nonradioactive materials. That is, these impacts are not associated with the radiological characteristics of the cargo.

For this study it is assumed that Hanford defense wastes will be transported by trucks and trains powered by gasoline or diesel fuel. The assumptions vary, depending upon the particular waste disposal alternative considered.

Dust will be generated in the turbulent wake behind a shipment, and pollutants, including particulates, sulfur and nitrogen oxides, hydrocarbons, and carbon monoxide, will be emitted from combustion of gasoline or diesel fuel. Also, particulates will be generated by abrasion of tires on paved surfaces. Procedures used to estimate the concentrations of these materials due to an assumed amount of traffic are discussed in this subsection.

I.3.2.1 Fugitive Dust Source Term

Fugitive dust generated on roads is computed using the following equation, which was developed for paved roads (Rao et al. 1982).

$$EF = 0.029(I)(4/n)(S/10)(L/280)(W/2.7)^{0.7}$$

where EF = fugitive dust emissions, kg/km

I = industrial road augmentation factor (1)

n = number of traffic lanes (4)

S = silt content on highway (10%)

L = surface dust loading on traveled portion of road (42 kg/km)

W = weight of truck trailer (34 t)

The values listed in parentheses, obtained from Rao et al. (1982), are consistent with those found in NUREG-0170 (NRC 1977).

No empirical equation is available for calculating the fugitive dust entrained by a passing rail car. For this study, it is assumed that the quantity entrained is 10% of that entrained behind a truck. This assumption is consistent with Rao et al. (1982) and DOE (1982).

I.3.2.2 Vehicular Exhaust Emissions

Emission factors for particulates, sulfur and nitrogen oxides, carbon monoxide, and hydrocarbons from heavy-duty diesel-powered trucks and trains are calculated using Environmental Protection Agency recommendations (Rao et al. 1982).

I.3.2.3 Pollutant Concentrations

Pollutant concentrations are calculated using the classic line-source model of diffusion in which the wind is assumed to be blowing in a direction perpendicular to the roadway. The geometry is represented in Figure I.5, and the equation is given below (Rao et al. 1982).

$$\bar{X} = \frac{K}{(D_{max} - D_{min})\ u} (2/\pi)^{1/2} IQ$$

where $\bar{X}$ = average concentration

Q = unit conversion factor

$= 1.3\ [(\frac{km}{m})(\frac{\mu g}{g})(\frac{h}{s})]$

D_{max} = 805 m (see Fig. I.5)

D_{min} = 30 m (see Fig. I.5)

u = wind speed: 3 m/sec

x = downwind distance, m

K = source term, g/km-h

$$I = \int_{30}^{805} x^{-0.78}\, dx$$

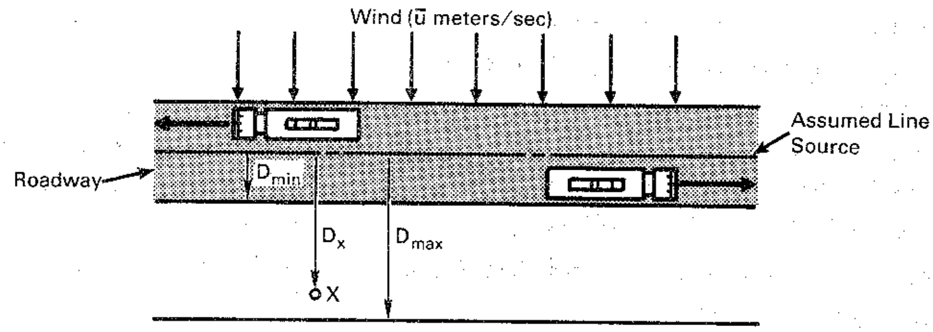


FIGURE I.5. Geometry Used To Calculate Nonradiological Impacts for Normal Transport

I.4 ACCIDENTS

This section discusses accident environments and releases that might occur under the most extreme, credible environments postulated. Also discussed is the expected frequency of these accident environments.

I.4.1 Accident Environments

About 500 billion packages of commodities of all kinds are transported within the United States each year (NRC 1977). Of those, about 2.8 million packages contain radioactive material (Javitz et al. 1985), i.e., only one in every 180,000 packages. Any transportation accidents involving hazardous materials must be reported to the U.S. Department of Transportation. On the average, about 6,000 accidents involving shipment of hazardous materials occur each year. Of those, fewer than 30 (~1/2%) contained radioactive materials, and none of those accidents entailed doses to the public exceeding applicable standards.

Use of DOT Type B packagings for offsite shipments is assumed for all scenarios because the radioactivity content of all shipments will exceed Type A packaging limits. As stated previously, all Type B packagings are certified to survive a series of hypothetical accident conditions as described in 10 CFR 71.73. These test environments are designed to simulate very severe transport accidents. The complete sequence consists of the following tests in this order:

1. Drop test: a 9-m drop onto an unyielding target
2. Puncture test: a 1-m drop onto a 15-cm-diameter probe
3. Thermal test: a 30-min-duration fire at 800°C
4. Water-immersion test: an 8-hr submersion in water.

Conditions equivalent to or more severe than these are not likely to be encountered. In fact, the percentage of accidents that do not exceed the test conditions is over 99.5% for both truck and rail transport (DOE 1980). This percentage was developed from a study of actual accidents (Dennis 1978) in which the cumulative probabilities of rail and truck accidents were estimated as a function of the change in velocity experienced by the package or the duration of a fire. The conditions produced by the hypothetical accident sequences listed above were then superimposed on graphs of cumulative probability of occurrence versus accident severity (i.e., velocity change or fire duration) to estimate the percentage of accidents that are less severe than the Type B regulatory test conditions.

Tests conducted at Sandia National Laboratories on spent fuel shipping casks have simulated very severe accident environments. Despite the severity of the conditions, only limited damage resulted to the casks (Jefferson and Yoshimura 1978). Actual accident data involving spent fuel casks is limited; however, no accident involving a Type B shipping container has resulted in a release of radioactive material.

I.4.2 Accident Rates and Probabilities

As discussed previously, the probability of a release of radioactive materials or loss of shielding is related to the severity of an accident. In general, a combination of mechanical (impact or puncture) and thermal (fire) environments are required to cause a loss-of-contents accident. The intensity of the accident environment is responsible for the degree of damage to the shipping container and for the quantity of radioactive material that is subsequently released. RADTRAN III categorizes accident severities by assigning accidents to a "severity category" based on the duration and temperature of a fire occurring during transport, and either crush forces (for truck transport) or puncture impact speed (for rail transport). Eight severity categories are considered in this analysis, with category 1 used to represent the regulatory conditions for Type A packages, category 2 to represent the hypothetical accident conditions, and higher categories for accident environments that exceed the regulatory conditions. Significant effort has been spent in defining these severity categories and their respective accident environments. The reader is referred to Dennis et al. (1978), Wilmot (1981), Wilmot et al. (1981), and McClure (1981) for further information regarding accident severities and probabilities.

The accident rates used in this appendix were derived from relatively large amounts of historical data (Dennis et al. 1978; McClure 1981). The error associated with these data is small, as indicated by Neuhauser and Reardon (1986). Although specific locations may have higher accident rates than others, these areas were included in the estimation of rural, suburban, and urban accident rates and helped determine the average rates that are incorporated in RADTRAN III. In addition, it was determined in a sensitivity analysis by Neuhauser and Reardon (1986) that a 100% increase in the accident rate values results in less than a 100% increase in the RADTRAN III-calculated accident risk value, and that calculated accident risk values are relatively insensitive to the specific accident rates used in the analysis. Furthermore, common sense indicates that accident rates are higher in urban areas than in rural or suburban areas. The accident rates used in this analysis (see Table I.8) are about 100 times higher in urban areas than in rural areas. Thus, areas where high accident rates and large populations coincide are considered in the analysis.

I.4.3 Release and Dispersibility

RADTRAN III uses four quantities to describe a release of radioactive material. These quantities are dependent upon severity category, the severity fraction (SEVFRC), the release fraction (RFRAC), the aerosolization fraction (AER), and the respirable fraction (RESP).

Severity fraction defines the fraction of accidents which occur that are of a particular severity. The values used in this analysis are shown in Table I.8.

The remaining parameters in Table I.8 determine the amount of material that could be released and subsequently inhaled by members of the public. "Release fraction" defines the amount of material of all sizes that could be released from the package (given as the fraction of the total quantity of material in the package). "Aerosolization fraction" defines the fraction of material released from the package that can be entrained in an aerosol (cloud of radioactive material), while the "respirable fraction" accounts for the fraction of aerosolized material that is also respirable. As indicated in Appendix F, only those particles less than 10 microns in size pose an inhalation hazard. As a result, only particles smaller than 10 microns are considered for the inhalation pathway.

The values used for the aerosolization and respirable fractions are set internally by the RADTRAN III code based on user-provided information regarding the physical characteristics of the waste form. Solidified HLW was assumed to be represented by "immobilized" material, cast slag by "sintered" material, and untreated TRU solid wastes by "loose powder-small" material (see Taylor and Daniel 1982). Cesium chloride is a molten salt and is poured into capsules where it solidifies. Strontium fluoride is a finely divided precipitate, which is fired and converted to sintered solid before encapsulation.

The release fraction values were obtained from various sources. For shipments of solidified HLW, the release fractions are consistent with those used by Wilmot et al. (1983). Release fractions for shipments of the cast slag material are assumed to be the values specified for drums by Shirley (1983). The design-basis release fractions for the TRUPACT system were used for retrievably stored and newly generated TRU wastes. Release fractions for shipments of strontium and cesium capsules were assumed to be a factor of 10 lower than the release fractions for HLW shipments. This factor-of-10 reduction accounts for the increased structural integrity of the triple-overpack configuration used for strontium and cesium materials. For additional protection the strontium and cesium capsules are placed within two overpack canisters before they are loaded into the shipping container. The increased structural integrity provided by the two overpacks was assumed to reduce the release fraction relative to HLW by a factor of 10. A review of the literature indicated that the strontium fluoride waste form consists of a hard, non-friable ceramic (Fullam 1981). Often, large chunks are encapsulated. Fullam (1981) indicated that heating of strontium fluoride, such as would be expected in a fire, will cause the material to sinter and agglomerate; i.e., not become airborne. For these reasons, the dispersible and respirable fractions were assumed to be 0.01 and 0.0005, respectively.

TABLE I.8. Values Used in RADTRAN III Analysis of Accident Impacts

Parameter, Fraction	Type of Waste: HLW	Sr/Cs	TRU-1[(a)]	TRU-2[(b)]
AERSOL	1.0×10^{-6}	1.0×10^{-2}	1.0×10^{-2}	1.0×10^{-1}
RESP	5.0×10^{-2}	5.0×10^{-2}	5.0×10^{-2}	5.0×10^{-2}

	Accident Rates (Accidents/km): Rural	Suburban	Urban
Truck	1.4×10^{-7}	2.7×10^{-6}	1.6×10^{-5}
Rail	1.5×10^{-7}	3.0×10^{-6}	1.3×10^{-5}

Severity Category	Release Fraction[(c)]: HLW	TRU-1[(a)]	TRU-2[(b)]	Sr/Cs
1	0.0	0.0	0.0	0.0
2	0.0	0.0	0.0	0.0
3	0.01	0.3	1×10^{-6}	0.001
4	0.1	0.5	1×10^{-5}	0.01
5	1.0	0.7	1×10^{-4}	0.1
6	1.0	0.9	1×10^{-3}	1.0
7	1.0	1.0	1×10^{-2}	1.0
8	1.0	1.0	1×10^{-1}	1.0

Severity Category	Severity Fraction[(d)] for Rail Shipments: Rural	Suburban	Urban	Severity Fraction[(d)] for Truck Shipments: Rural	Suburban	Urban
1	3.6×10^{-1}	3.1×10^{-1}	5.7×10^{-1}	4.6×10^{-1}	4.4×10^{-1}	5.8×10^{-1}
2	2.1×10^{-1}	1.9×10^{-1}	3.4×10^{-1}	3.0×10^{-1}	2.9×10^{-1}	3.8×10^{-1}
3	3.9×10^{-1}	4.5×10^{-1}	7.7×10^{-2}	1.8×10^{-1}	2.2×10^{-1}	2.8×10^{-2}
4	3.9×10^{-2}	4.5×10^{-2}	7.7×10^{-2}	4.0×10^{-2}	5.1×10^{-2}	6.4×10^{-3}
5	6.4×10^{-3}	3.4×10^{-3}	5.1×10^{-4}	1.2×10^{-2}	6.6×10^{-3}	7.4×10^{-4}
6	6.5×10^{-4}	1.6×10^{-4}	1.9×10^{-5}	6.5×10^{-3}	1.7×10^{-3}	1.5×10^{-4}
7	3.4×10^{-4}	3.8×10^{-5}	8.6×10^{-6}	5.7×10^{-4}	6.7×10^{-5}	1.1×10^{-5}
8	6.4×10^{-5}	3.1×10^{-6}	7.2×10^{-7}	1.1×10^{-4}	5.9×10^{-6}	9.9×10^{-7}

(a) TRU-1 designates the cast slag waste form from processing of TRU-contaminated soil and pre-1970 buried TRU wastes.
(b) TRU-2 designates retrievably stored and newly generated TRU waste. For conservatism, they are defined as a loose powder (RADTRAN III dispersibility category 5).
(c) Given as the fraction of the cask contents that are released as a result of an accident.
(d) Given as the fraction of accidents that occur that would be representative of the accident conditions described by each severity category. The overall accident frequency for each severity category can be obtained by multiplying the severity fraction by the overall accident rate.

I.5 RADIOLOGICAL IMPACTS

This section presents the results of the radiological impact analysis for transporting Hanford defense wastes. A discussion of why these results are considered to be conservative was presented in the Introduction to the Appendices.

I.5.1 Impacts During Normal Transport

The radiological impacts of routine transportation of Hanford defense wastes are described in this section. The input data used to calculate these impacts are also presented.

Some of the miscellaneous data used in this analysis are listed in Table I.9. Most of these data are available as default input to RADTRAN III and are consistent with data used in NUREG-0170 (NRC 1977).

TABLE I.9. Input Data for Impact Analysis of Routine Transport (DOE 1982; Wilmot et al. 1983)

Parameter	Truck	Rail
Number in Crew	2	5
Distance from Source to Crew, m	5	20
Population Densities, persons/km^2		
High-population zone (urban)	3,861	3,861
Medium-population zone (suburban)	719	719
Low-population zone (rural)	6	6
Average Speed, km/hr		
High-population zone (urban)	24	24
Medium-population zone (suburban)	40	40
Low-population zone (rural)	88	64
Number of Persons per Vehicle	2	3
Traffic Count, one-way vehicles/hr		
High-population zone (urban)	2,800	5
Medium-population zone (suburban)	780	5
Low-population zone (rural)	470	1
Average Exposure Distance While Vehicle Stopped, m	20	20
Stopover Time, hr/km	0.011	0.033
Number of Persons Exposed While Vehicle Stopped	50	100

The population densities of the regions across which shipments must be moved can influence the risks. The fractions of travel in the three population zones (urban, suburban, and rural) were taken from NUREG-0170 for shipments to an offsite repository 4,800 km away. The values are 5% of the travel in urban and suburban areas and 90% of the travel in rural areas. These values tend to overestimate exposures for shipments in the predominantly rural western

United States. Fractions of travel for shipments to WIPP were taken from the Joint Integration Office (JIO 1985). These values are 82% rural, 16% suburban, and 2% urban.

The calculated radiation doses received by the five population groups are presented in Table I.10. The cumulative impacts of transporting defense TRU wastes to the WIPP from all major federal sites were discussed in Section 5.1.4. The population doses for the routine transport of all Hanford defense wastes considered in this analysis are given in units of man-rem. The doses presented in Table I.10 would not result in any health effects.

The population doses shown in the table can be put in perspective by comparing them with the natural background doses received by the population along the hypothetical route between Hanford and the offsite repository. For this comparison, the affected population is assumed to be within 0.5 km on either side of the route. Thus the total affected area is 4,800 km long and 1 km wide or 4,800 km^2. It was assumed that 5% of this route traverses urban and suburban population zones and 90% traverses a rural region. After the appropriate population densities (from Table I.9) are multiplied by the affected area in each population zone and the three zones are summed, the total affected population becomes about 1.1 million persons. Assuming that each member of the general public receives an annual dose of 0.1 rem, a population dose of 110,000 man-rem is obtained. Applying the health effects conversion factor (100 to 1,000 cancer fatalities and genetic effects per million man-rem) results in an average annual number of health effects attributable to background radiation sources (cosmic and terrestrial) of about 11 to 110, or about 200 to 2,000 health effects over the assumed 18-year operating period of the repository. This can be compared with the expected number of health effects in the geologic disposal alternative for the option in which all wastes are shipped to an offsite repository. There the population dose amounted to 85 man-rem from which no health effects would be expected.

I.5.2 Impacts During Accident Conditions

The radiological impacts due to transport accidents involving Hanford defense wastes are described in this section. The impacts are presented in terms of expected health effects.

Some of the data needed to calculate impacts due to accidents were described previously in Section I.4. Data such as atmospheric stability, average wind speed, and population densities are default values for RADTRAN III. Fractions of travel in the three population zones are the same as those assumed for normal exposure calculations.

Each waste type contains a number of different radioisotopes. The curie inventories in shipments of the various waste types were presented in Tables I.2 through I.7. Additional information that is required in RADTRAN III to calculate the effects on the public of a release of radioactive materials is given in Madsen et al. (1983).

The results of the radiological risk calculations are presented in Table I.11. Separate risk values are presented for the onsite and offsite shipment alternatives for each waste type. In general, due to shorter travel distance, the onsite shipments are expected to result in lower impacts. Risk values are low for all waste types. As shown, shipment of strontium/cesium capsules to onsite or offsite disposal facilities dominates the risks due to

TABLE I.10. Cumulative Whole-Body Radiation Dose from Routine Transport of Hanford Defense Wastes, man-rem[(a)]

Waste Type	Dose to Crew	Dose to Surrounding Population	Dose to Persons on Road or Rail	Dose to Bystanders at Stop or Switchyards	Total
Existing Tank Wastes[(b)]					
Geologic Disposal					
Onsite	3.2×10^{-2}	6.7×10^{-3}	4.0×10^{-4}	1.7×10^{-2}	5.7×10^{-2}
Offsite	16	8.3	2.1×10^{-1}	8.4	32
Reference Alternative					
Onsite	7.8×10^{-4}	1.6×10^{-4}	9.6×10^{-6}	4.2×10^{-4}	1.4×10^{-3}
Offsite	3.7×10^{-1}	2.0×10^{-1}	4.9×10^{-3}	2.0×10^{-1}	7.8×10^{-1}
Future Tank Wastes[(b)]					
Geologic Disposal					
Onsite	7.6×10^{-3}	1.6×10^{-3}	9.3×10^{-5}	4.1×10^{-3}	1.3×10^{-2}
Offsite	4.5	2.0	4.8×10^{-2}	2.7	9.2
Reference Alternative					
Onsite	1.4×10^{-3}	2.8×10^{-4}	1.7×10^{-6}	7.4×10^{-4}	2.4×10^{-3}
Offsite	8.1×10^{-1}	3.5×10^{-1}	8.8×10^{-3}	4.9×10^{-1}	1.7
Sr/Cs Capsules[(c)]					
Onsite	6.3×10^{-2}	1.8×10^{-3}	5.8×10^{-3}	7.9×10^{-2}	1.5×10^{-1}
Offsite	3.9×10^{-1}	1.7×10^{-1}	4.2×10^{-3}	2.4×10^{-1}	8.0×10^{-1}
Pre-1970 Buried TRU[(d)]					
Offsite	1.9	1.5	3.6×10^{-2}	2.6	6.0
TRU-Contaminated Soil[(d)]					
Offsite	5.0×10^{-1}	3.9×10^{-1}	9.7×10^{-3}	6.9×10^{-1}	1.6
Retrievably Stored TRU[(e)]					
Offsite	7.5	2.2×10^{-1}	6.8×10^{-1}	9.5	18
Newly Generated TRU[(e)]					
Geologic Disposal	8.0	2.3×10^{-1}	7.2×10^{-1}	1.0	19
Reference Alternative	9.2	2.7×10^{-1}	8.3×10^{-1}	1.2	22
Totals					
Geologic Disposal					
HLW onsite; TRU to WIPP	18	2.4	1.5	23	45
HLW offsite; TRU to WIPP	38	13	1.7	33	85
Reference Alternative					
For HLW onsite, TRU to WIPP	17	4.9×10^{-1}	1.4	22	40
For HLW offsite, TRU to WIPP	18	1.2	1.5	22	43

(a) Exposures can be converted to health effects by multiplying by health effects conversion factor; 100 to 1,000 health effects (latent cancer fatalities plus genetic effects) per million man-rem.
(b) Onsite rail transport to a basalt repository; offsite rail transport to a hypothetical repository.
(c) Onsite truck transport to a basalt repository; offsite rail transport to a hypothetical repository.
(d) Offsite rail transport to WIPP.
(e) Truck transport is assumed for offsite (WIPP) shipments.

TABLE I.11. Total Radiological Risk Due to Accidents Involving Hanford Defense Wastes

Waste Type	Risk of Health Effects[a] Onsite Transport	Risk of Health Effects[a] Offsite Transport
Existing Tank Wastes[b]		
Geologic Disposal	2.2×10^{-7} x 2.2×10^{-6}	0.7×10^{-4} - 0.7×10^{-3}
Reference Alternative	1.7×10^{-8} x 1.7×10^{-7}	0.6×10^{-5} - 0.6×10^{-4}
Future Tank Wastes[b]		
Geologic Disposal	1.0×10^{-7} - 1.0×10^{-6}	0.9×10^{-4} - 0.9×10^{-3}
Reference Alternative	4.4×10^{-7} - 4.4×10^{-6}	1.4×10^{-4} - 1.4×10^{-3}
Sr/Cs Capsules[c]	1.4×10^{-4} - 1.4×10^{-3}	0.6×10^{-2} - 0.6×10^{-1}
Pre-1970 Buried TRU Solid Wastes[d]	NA	4.3×10^{-6} - 4.3×10^{-5}
TRU Contaminated Soil Sites[d]	NA	0.6×10^{-6} - 0.6×10^{-5}
Retrievably Stored TRU Wastes[e]	NA	2.0×10^{-6} - 2.0×10^{-5}
Newly Generated TRU Wastes[e]		
Geologic Disposal	NA	0.8×10^{-5} - 0.8×10^{-4}
Reference Alternative	NA	0.9×10^{-5} - 0.9×10^{-4}
Totals		
Geologic Disposal		
HLW onsite; TRU to WIPP	0	
HLW offsite; TRU to WIPP	0	
Reference Alternative		
For HLW onsite, TRU to WIPP	0	
For HLW offsite, TRU to WIPP	0	

(a) Health effects - 100-1,000 latent cancer fatalities plus genetic effects per million man-rem.
(b) Onsite rail transport to basalt repository; offsite rail transport to hypothetical repository.
(c) Offsite rail transport to hypothetical repository.
(d) Offsite rail transport to WIPP.
(e) Truck transport is assumed for offsite (WIPP) shipments.

transportation accidents. However, in no cases were any fatalities calculated. The cumulative impacts of transporting defense TRU wastes to the WIPP from all major federal sites were discussed in Section 5.1.4.

I.6 NONRADIOLOGICAL IMPACTS

This section presents the nonradiological impacts from transportation of Hanford defense wastes.

I.6.1 Nonradiological Impacts of Normal Transport

Pollutants are emitted during normal transport by the combustion of diesel fuel, by the passage of a shipment over a dusty road surface, and by tire abrasion. Combustion of diesel fuel generates sulfur dioxide, carbon monoxide, hydrocarbons, nitrogen dioxide, and particulates. The passage of a shipment over a roadbed or highway produces fugitive dust, and tire particulates are generated from the abrasion of tires on the pavement. Each pollutant has a unique character, and each may affect health. Each pollutant is described briefly here, and the health implications of each are discussed.

The category of pollutants called particulates can include a wide variety of particles with differing sizes, compositions, and origins. The size of the particulate is important when determining its dispersibility and potential health effects. For example, very large particles (>100 μm diameter) will settle within a few meters of their source. Particles less than 10 μm in diameter are considered respirable; that is, they can be inhaled and be deposited in the lungs. The larger particles, if inhaled, will be deposited in the nose and throat upon entry into the respiratory tract and will then be eliminated from the body. Therefore, the smaller particulates present a greater hazard than large particulates.

Particulates also act as scavengers for other pollutants; i.e., they often will contain or carry other absorbed toxic materials such as lead or other heavy metals. The composition of particulates depends on their origin (e.g., dust particles differ from particles in exhaust emissions) and thus is affected by specific locations. In general, sources of particulates can be industry, agriculture, or transportation. Since the mix of these sources varies among specific locations, the composition of the particles will differ in each community and thus may not have the same health effect (Rao et al. 1982).

Sulfur dioxide is a nonflammable, nonexplosive, colorless gas. The gas is detected first by taste and, at higher concentrations, can be detected by odor. In the atmosphere, it is partially converted to more hazardous products by photochemical or catalytic processes. Sulfur dioxide and its products irritate the lining of the respiratory tract. The irritation, which may be temporary or permanent, is more severe for the compounds of sulfur dioxide (such as sulfuric acid) than for sulfur dioxide itself, and may result in breathing difficulties.

Nitrogen dioxide is known to be toxic at relatively high concentrations and is a strong irritant to the eyes, nose and throat. Acute and chronic injury of the lungs, causing irreversible damage, has been observed at high concentrations. It is also involved in many complex chemical reactions. In the presence of sunlight, it may be converted to even more toxic intermediates.

Carbon monoxide has an affinity for hemoglobin, with which it combines, reducing the capability of the blood to carry oxygen. From a physiological viewpoint, symptoms of CO inhalation are similar to symptoms of anemia.

Because of the large variety of possible hydrocarbon pollutants, a discussion of each is restricted. We simply note that some are definitely carcinogenic and that many produce adverse health effects. Little information is available, however, from studies involving long-term exposure of humans to hydrocarbons.

The character of each pollutant can be described as a result of having been isolated during detailed laboratory experiments. However, since pollutants can also interact and form new and intermediate toxic pollutants, their effects can very rarely be isolated.

Determining the health effects produced by atmospheric pollutants is a difficult and complex problem. It is generally believed that air pollution can cause increased mortality and that pollutant levels at the relatively low ambient concentrations associated with transportation can result in increased respiratory symptoms. It is not possible, however, using present analytical techniques, to state that specific health effects are a result of a particular pollutant. Quantitative estimates of health effects have been prepared (Rao et al. 1982) but must be qualified extensively, which is beyond the scope of this study.

To compare emissions somewhat quantitatively to current pollution standards, the emissions resulting from the hourly passing of one diesel-powered truck or locomotive hauling a shipment of Hanford defense wastes was used to calculate an average air pollutant concentration. Estimates of the emissions from transportation, based on Environmental Protection Agency documentation (Rao et al. 1982) and equations in Section I.3.2, are listed in Table I.12. These data are source terms for calculating average concentrations which, in turn, can be compared to national primary air-quality standards.

TABLE I.12. Nonradiological Emissions from Transportation, g/km[a]

Pollutant	Truck	Rail
Particulates	0.81	4.5
Sulfur Dioxide	5.1	10
Nitrogen Dioxide	13	65
Hydrocarbons	3.3	19
Carbon Monoxide	22	24
Tire Particulates	0.54	(b)
Fugitive Dust	23	14

(a) Assumes 24 km/hr average speed in urban areas.
(b) Not applicable.

The values in Table I.12 were substituted into the atmospheric dispersion equation discussed in Section I.3.2 to obtain average concentrations of the pollutants. The resultant concentrations and the primary air-quality standards are shown in Table I.13. For each pollutant, the calculated concentration is much lower than the standard, even when one truck or train per hour is considered. Since the average number of Hanford defense waste shipments would more likely be two to three per day, the nonradiological impacts from them would probably be even smaller. The calculated pollutant levels are also significantly lower than measured levels in urban areas. The calculated concentrations are actually incremental increases in the levels due to shipments of Hanford defense wastes.

TABLE I.13. Comparison of Calculated Pollutant Concentrations With Air-Quality Standards and Monitored Mean Pollutant Levels, $\mu g/m^3$

Pollutant	Calculated Concentration Truck	Calculated Concentration Rail	Primary Air Quality Standard	Monitored Mean Pollutant Levels[a]
Particulates	0.031	0.014	75[b]	103
SO_x	0.012	0.024	80[c]	33
NO_x	0.031	0.15	100[c]	49.6
Hydrocarbons	0.0078	0.045	[d]	(e)
CO	0.052	0.056	10,000[f]	2.6

(a) Rao et al. 1982.
(b) Annual geometric mean.
(c) Annual arithmetic mean.
(d) No longer a primary standard.
(e) Hydrocarbon levels not measured.
(f) 8-hr maximum.

The expected health effects from these emissions were calculated using unit-risk factors obtained from Neuhauser et al. (1984). Unit-risk factors are a measure of the expected health effects (here, latent cancer fatalities--LCF) per km of travel. These factors are multiplied by the total distance traveled by the various types of Hanford defense wastes to estimate the number of health effects. The unit-risk factors are:

$$\text{Rail} = 1.3 \times 10^{-7} \text{ LCF/km (urban areas only)}$$
$$\text{Truck} = 1.0 \times 10^{-7} \text{ LCF/km (urban areas only).}$$

The results of the calculations are shown in Table I.14.

I.6.2 Nonradiological Impact of Transportation During Accident Conditions

Injuries and fatalities would be the nonradiological impact expected from accidents during transport of Hanford defense wastes to assumed repository locations. These injuries and fatalities are not directly related to the radioactive cargo being transported; however, they

TABLE I.14. **Nonradiological Impacts from Routine Transport of Hanford Defense Wastes**

Waste Type[a]	Health Effects[b]
Existing Tank Wastes	
Geologic Disposal	3.5×10^{-1}
Reference Alternative	8.5×10^{-3}
Future Tank Wastes	
Geologic Disposal	8.3×10^{-2}
Reference Alternative	1.5×10^{-2}
Sr/Cs Canisters	7.1×10^{-3}
Pre-1970 Buried TRU Solid Wastes	1.8×10^{-1}
TRU-Contaminated Soil Sites	5.6×10^{-2}
Retrievably Stored TRU Wastes	6.5×10^{-2}
Newly Generated TRU Wastes	
Geologic Disposal	9.7×10^{-2}
Reference Alternative	1.1×10^{-1}
Totals	
Geologic Disposal	
HLW onsite; TRU to WIPP	0
All to offsite repository[c]	1
Reference Alternative	
HLW onsite; TRU to WIPP	0
All to offsite repository	0

(a) See Chapter 3 of this EIS for descriptions of the alternatives.
(b) Unit risk factors are 0 for suburban and rural zones. Thus, there are 0 health effects for onsite shipments because the suburban zone was assumed for onsite shipments. Health effects equal latent cancer fatalities. These values include both public and occupational health effects.
(c) Assumes HLW is shipped to an offsite repository and TRU waste is shipped to WIPP.

would not be incurred if the cargo were not being transported. Thus the number of estimated injuries and fatalities would be the same even if the cargo was not radioactive material. This section uses unit-risk factors (injuries or fatalities per kilometer of travel) derived from published data on vehicular accidents to calculate these impacts.

The potential for accidents involving shipments of radioactive wastes is assumed comparable to that of general truck and rail transport in the United States. Rao et al. (1982) used statistics compiled by the Department of Transportation to develop the unit-risk factors shown in Table I.15. These factors are multiplied by the total distance traveled by each

TABLE I.15. Unit-Risk Factors for Vehicular Accidents[a]

Affected Persons/ Transport Mode	Population Zone Urban	Suburban	Rural
Nonoccupational			
Truck			
Fatalities/km	7.5×10^{-9}	1.3×10^{-8}	5.3×10^{-8}
Injuries/km	3.7×10^{-7}	3.8×10^{-7}	8.0×10^{-7}
Rail			
Fatalities/km	1.7×10^{-8}	1.7×10^{-8}	1.7×10^{-8}
Injuries/km	3.3×10^{-8}	3.3×10^{-8}	3.3×10^{-8}
Occupational			
Truck			
Fatalities/km	2.1×10^{-9}	3.7×10^{-9}	1.5×10^{-8}
Injuries/km	1.3×10^{-8}	1.3×10^{-8}	2.8×10^{-8}
Rail			
Fatalities/km	1.4×10^{-9}	1.4×10^{-9}	1.4×10^{-9}
Injuries/km	1.9×10^{-7}	1.9×10^{-7}	1.9×10^{-7}

(a) Neuhauser et al. (1984).

type of waste shipment to calculate the expected number of injuries and fatalities due to transportation of Hanford defense wastes. These impacts, averaged over all population zones, are shown in Tables I.16, and a summary of these impacts is shown in Table I.17.

I.7 TRANSPORTATION COSTS

Estimates of the transportation costs for shipment of various types of Hanford defense wastes are presented in this section. The bases, assumptions, and methods used to calculate these costs are also discussed.

I.7.1 Bases, Assumptions, and Methods

Two methods are used to calculate transportation costs: one for onsite shipments and one for offsite shipments. Both methods require calculation of a set of transportation cost elements for each waste class. Unit transportation costs are defined as the sum of the following three costs: 1) capital, 2) maintenance, and 3) shipping. Costs used in this analysis are based on 1987 dollars and should be used only for comparison among destination sites.

Transportation costs for offsite shipments are calculated using the following information. The capital cost element is the total capital cost for purchase of the required number of transport packagings. Packaging requirements are calculated using the total number of shipments shown in Table I.1 and assuming average truck speeds of 56 km/hr and rail speeds of 4.8 km/hr for short hauls and 18 km/hr for cross-country shipments (Wilmot et al. 1983). In addition, total loading plus unloading times for truck and rail shipments are assumed to be 3 and 5 days, respectively (Wilmot et al., 1983). Transport packages are assumed to be

TABLE I.16. **Nonradiological Impacts from Accidents During Shipments of Hanford Defense Wastes**

Waste Type[a]	Onsite Shipments[b]		Offsite Shipments[b]	
	Injuries	Fatalities	Injuries	Fatalities
Existing Tank Wastes				
Geologic Disposal	3.0×10^{-3}	1.0×10^{-3}	6.0	4.9×10^{-1}
Reference Alternative	7.1×10^{-5}	2.5×10^{-5}	1.4×10^{-1}	1.2×10^{-2}
Future Tank Wastes				
Geologic Disposal	6.9×10^{-4}	2.5×10^{-4}	1.4	1.2×10^{-1}
Reference Alternative	1.2×10^{-4}	4.3×10^{-5}	2.5×10^{-1}	2.1×10^{-2}
Sr/Cs Canisters	4.1×10^{-3}	1.7×10^{-4}	1.2×10^{-1}	9.8×10^{-3}
Pre-1970 Buried TRU Solid Wastes	NA[c]	NA	3.1	2.5×10^{-1}
TRU-Contaminated Soil Sites	NA	NA	9.5×10^{-1}	7.9×10^{-2}
Retrievably Stored TRU Wastes	NA	NA	3.6	2.8×10^{-1}
Newly Generated TRU Wastes				
Geologic Disposal	NA	NA	5.5	4.2×10^{-1}
Reference Alternative	NA	NA	6.3	4.9×10^{-1}

(a) See Chapter 3 of this EIS for descriptions of alternatives.
(b) Onsite refers to shipments to a basalt repository; offsite refers to shipments to a hypothetical repository 4,800 km away (for tank wastes and Sr/Cs canisters) or to WIPP (TRU wastes).
(c) NA: Not applicable because TRU waste is either disposed of in place or shipped to WIPP.

TABLE I.17. **Summary of Nonradiological Impacts from Accidents During Shipment of Hanford Defense Wastes[a]**

Waste Type[b]	Injuries	Fatalities
Geologic Disposal		
HLW onsite; TRU to WIPP	13	1
HLW offsite; TRU to WIPP	21	2
Reference Alternative		
HLW onsite; TRU to WIPP	10	1
HLW offsite; TRU to WIPP	10	1

(a) Includes both impacts of routine transport and accidents.
(b) See Chapter 3 of this EIS for descriptions of the alternatives.

available 300 days per year and have operational lifetimes of 15 years. Capital costs for each transport package (including trailer or rail car) were extracted from Wilmot et al. (1983), and are presented in Table I.18 in 1987 dollars.

TABLE I.18. Transport Package Capital Costs, $ Million[a]

Waste Type	Transport Mode Truck	Transport Mode Rail
Solidified HLW	NA[b]	2.8
Sr/Cs Canisters	1.8	2.8
TRU Wastes (offsite)[c]	1.1	2.0
TRU Wastes (onsite)[d]	0.13	NA

(a) Wilmot et al. 1983. Includes cost of trailer or railcar. Costs are escalated to 1987 dollars.
(b) NA = not applicable. No truck shipments of solidified HLW or onsite rail shipments of TRU wastes are assumed.
(c) TRUPACT transport packaging.
(d) Enclosed cargo van; cost is estimated.

The capital cost for the required transport packages per year is multiplied by a maintenance factor to obtain the total maintenance costs for each waste type. This factor is assumed to be 5% of the initial capital cost per year for HLW and strontium/cesium waste packagings and 10% per year for TRU waste packagings (Wilmot et al. 1983). Since this cost varies with the number of packagings required, the total maintenance costs will be different for each waste disposal alternative.

Shipping costs are defined as costs charged by commercial carrier companies for moving waste shipments from their origin to a destination facility and returning the empty shipping container to its origin. Shipping costs were determined for the various waste classes and shipment distances in this analysis. Data used to determine shipping costs were obtained from McNair et al. (1986) and are consistent with shipping costs used by Wilmot et al. (1983). These data are based on published tariffs and include such items as freight rates, demurrage,[a] security, and special equipment costs. Due to deregulation of the transportation industry, actual shipping costs cannot be determined until a contract is negotiated between the shippers and carrier companies.

The method used to calculate transport costs for onsite shipments does not assume use of commercial carrier companies; therefore, the shipping costs were calculated differently. Onsite shipping costs comprise the labor costs required to drive packages to/from the onsite repository. Shipping costs are calculated assuming that two drivers accompany each truck

(a) Detention of drivers/vehicles while shipping container is loaded or unloaded.

shipment and three train workers accompany each rail shipment. A single transport packaging is assumed for each trip. It is further assumed that labor costs amount to $60/hr.

Total transportation costs are the sum of capital, maintenance, and shipping costs, as discussed above. Data employed in the calculations include the total number of shipments, the average annual number of shipments, empty and loaded weights for the packages, and transit times. These costs do not account for such factors as road and rail extensions to facilities or handling costs. All costs are given in constant 1987 dollars; no discounting is assumed.

I.7.2 Results of Transportation Cost Calculations

Results of the transportation cost calculations for shipment of the various types of Hanford defense wastes are shown in Table I.19. The total cost, number of shipments, and package requirements for each waste class are shown in the table. Package requirements are rounded to the next largest whole number for conservative estimates. Although total costs are based on the best available information and are believed to be representative approximations, they are intended for comparison purposes only.

A summary of the transportation costs for each alternative is shown in Table I.20.

TABLE I.19. Total Packaging Requirements and Transportation Costs

Waste Type	Total Shipments[a]		Total Packagings		Total Costs,[b] $ Million		
	Onsite	Offsite	Onsite	Offsite	Onsite	Offsite	WIPP
Existing Tank Wastes							
Geologic Disposal	2,829	2,829	6	30	29	380	NA[c]
Reference Alternative	68	68	2	2	8.4	14	NA
Future Tank Wastes							
Geologic Disposal	662	662	2	8	9.2	67	NA
Reference Alternative	119	119	2	2	8.5	19	NA
Sr/Cs Canisters	509	57	2	2	5.4	13	NA
Pre-1970 Buried TRU Solid Wastes	NA	570	NA	18	NA	NA	130
TRU-Contaminated Soil Sites	NA	178	NA	6	NA	NA	46
Retrievably Stored TRU Wastes	NA	1,040	NA	4	NA	NA	16
Newly Generated TRU Wastes							
Geologic Disposal	NA	1,560	NA	4	NA	NA	21
Reference Alternative	NA	1,800	NA	6	NA	NA	27

(a) See Table I.1.
(b) Costs are given in 1987 dollars.
(c) NA = not applicable.

TABLE I.20. Summary of Transportation Costs

Waste Type	Total Cost,[a] $ Million
Geologic Disposal	
HLW onsite; TRU to WIPP	260
All to offsite repository	670
Reference Alternative[b]	
HLW onsite; TRU to WIPP	65
All to offsite repository	89

(a) Based on 1987 dollars.
(b) See Section 3.4 of this EIS for a description of the Reference Alternative.

I.8 EMERGENCY RESPONSE

Many agencies share the responsibilities for dealing with nonroutine events such as radioactive material transportation accidents. A national radiological assistance plan exists for dealing with a real or suspected release of radioactive material from a shipment in transit. For example, under this plan, the Federal Emergency Management Agency (FEMA) has the primary responsibility for emergency response planning for transportation accidents involving radioactive materials. Also, at the federal level, the DOE will make available from its resources radiological advice and assistance to protect the public health and safety and to cope with radiological hazards. Federal support is also available from the Environmental Protection Agency (EPA), the Department of Health and Human Services through the Food and Drug Administration (FDA), the DOT-Materials Transportation Bureau, and the NRC.

The ultimate responsibility for emergency-response planning generally lies with state and local governments. Most state and many local governments have established emergency-response plans. Local jurisdictions assume primary responsibility for emergency-response planning because a member of a local law enforcement agency or fire department is likely to be the first responder to a transportation accident.

The FEMA has published "Guidance for Development of State and Local Radiological Emergency Response Plans and Preparedness" (FEMA 1983). This document details necessary components of emergency-response plans, including organizational responsibilities and jurisdictions; accident characteristics and assessment; radiological exposure control; required emergency equipment, resources, and communications; medical support; notification methods and procedures; emergency-response training activities; and post-accident operations.

I.9 REFERENCES

Dennis, A. W., et al. 1978. Severities of Transportation Accidents Involving Large Packages. SAND77-0001, Sandia National Laboratories, Albuquerque, New Mexico.

Department of Energy (DOE). 1980. Final Environmental Impact Statement--Waste Isolation Pilot Plant. DOE/EIS-0026, Assistant Secretary for Defense Programs, Washington, D.C.

Department of Energy (DOE). 1982. Final Environmental Impact Statement - Defense Waste Processing Facility, Savannah River Plant, Aiken S.C. DOE/EIS-0082-Final, Assistant Secretary for Defense Programs, Washington, D.C.

Department of Energy (DOE). 1985. Safety Requirements for the Packaging and Transportation of Hazardous Materials, Hazardous Substances, and Hazardous Wastes. DOE Order 5480.3, Washington, D.C.

Federal Emergency Management Agency (FEMA). 1983. Guidance for Developing State and Local Radiological Emergency Response Plans and Preparedness for Transportation Accidents. FEMA-REP-5, Washington, D.C.

Federal Register. 1973. Memorandum of Understanding Between the U.S. Department of Transportation and the Atomic Energy Commission on Certification of DOE-Owned Packagings. Volume 38, p. 8466.

Fullam, H. T. 1981. Strontium-90 Fluoride Data Sheet. PNL-3846, Pacific Northwest Laboratory, Richland, Washington.

Javitz, H. S., et al. 1985. Transport of Radioactive Material in the United States: Results of a Survey to Determine the Magnitude and Characteristics of Domestic, Unclassified Shipments of Radioactive Material. SAND84-7174, SRI International, Menlo Park, California.

Jefferson, R. M., and H. R. Yoshimura. 1978. Crash Testing of Nuclear Fuel Shipping Containers. SAND77-1462, Sandia National Laboratories, Albuquerque, New Mexico.

Joint Integration Office (JIO). 1985. Transuranic Waste Transportation Assessment and Guidance Report. DOE-JIO 002, Albuquerque, New Mexico.

Madsen, M. M., E. L. Wilmot and J. M. Taylor. 1983. RADTRAN II Users Guide. SAND82-2681, Sandia National Laboratories, Albuquerque, New Mexico.

Madsen, M. M., et al. 1986. RADTRAN III. SAND84-0036, Sandia National Laboratories, Albuquerque, New Mexico.

McClure, J. D. 1981. The Probability of Spent Fuel Transportation Accidents. SAND80-1721, Sandia National Laboratories, Albuquerque, New Mexico.

McNair, G. W., et al. 1986. Truck and Rail Charges for Shipping Spent Fuel and Nuclear Wastes. PNL-5797, Pacific Northwest Laboratory, Richland, Washington.

Neuhauser, K. S., et al. 1984. A Preliminary Cost and Risk Analysis for Transporting Spent Fuel and High-Level Waste to Candidate Repository Sites. SAND84-1795, Sandia National Laboratories, Albuquerque, New Mexico.

Neuhauser, K. S., and P. C. Reardon. 1986. A Demonstration Sensitivity Analysis Using RADTRAN III. SAND85-1001, Sandia National Laboratories, Albuquerque, New Mexico.

Nuclear Regulatory Commission (NRC). 1977. Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes. NUREG-0170, Washington, D.C.

Rao, R. K., et al. 1982. Non-Radiological Impacts of Transporting Radioactive Material. SAND81-1703, Sandia National Laboratories, Albuquerque, New Mexico.

Rockwell Hanford Operations. 1985. Hanford Defense Waste Disposal Alternatives: Engineering Support Data for the HDW-EIS. RHO-RE-ST-30 P, Richland, Washington.

Shirley, C. G. 1983. "A Simple Risk Analysis Tool for Estimating Release Fractions from Shipments of 55-gallon Drums Involved in Transportation Accidents." In Proceedings of the 7th International Symposium on Packaging and Transportation of Radioactive Material (PATRAM '83), pp. 160-167. CONF-830528, May 15-20, 1983, New Orleans, Louisiana.

Taylor, J. M., and S. L. Daniel. 1982. RADTRAN II: A Revised Computer Code to Analyze Transportation of Radioactive Material. SAND80-1943, Sandia National Laboratories, Albuquerque, New Mexico.

Wilmot, E. L. 1981. Transportation Accident Scenarios for Commercial Spent Fuel. SAND80-2124, Sandia National Laboratories, Albuquerque, New Mexico.

Wilmot, E. L., J. D. McClure and R. E. Luna. 1981. Report on a Workshop on Transportation Accident Scenarios Involving Spent Fuel, May 6-8, 1980. SAND80-2012, Sandia National Laboratories, Albuquerque, New Mexico.

Wilmot, E. L., et al. 1983. A Preliminary Analysis of the Cost and Risk of Transporting Nuclear Waste to Potential Candidate Commercial Repository Sites. SAND83-0867, Sandia National Laboratories, Albuquerque, New Mexico.

APPENDIX J

METHOD FOR CALCULATING REPOSITORY COSTS USED IN THE HANFORD DEFENSE WASTE ENVIRONMENTAL IMPACT STATEMENT

THIS PAGE INTENTIONALLY
LEFT BLANK

APPENDIX J

METHOD FOR CALCULATING REPOSITORY COSTS USED IN THE HANFORD DEFENSE WASTE ENVIRONMENTAL IMPACT STATEMENT

Total costs for repository disposal of Hanford defense wastes are the sum of costs for three activities: retrieval and processing, transportation, and repository emplacement. Retrieval and processing costs are from Rockwell Hanford Operations (Rockwell 1985, 1987). Transportation costs are estimated on the basis of cost per kilometer traveled (Appendix I, Section I.7).

This appendix describes the method of estimating costs for repository emplacement. Cost estimates on a common basis have been generated for disposal of Hanford defense waste (HDW) in commercial waste repositories. Incremental mining and waste-handling costs in a commercial salt repository design were used to estimate the Waste Isolation Pilot Plant (WIPP) costs for contact-handled transuranic waste. The results of these analyses are contained in Table J.1, which breaks down the costs for disposal of the individual waste classes.

The numbers used in the draft EIS for calculating the required fee per volume of high-level waste disposed of in a deep geologic repository were based on very preliminary repository costs. Since then, there has been an effort underway to establish defense high-level waste disposal fees. This has resulted in a report that presents a perspective on methods to calculate a fee and provides a tentative cost range of $75,000 to $200,000 per canister depending on the approach used (DOE 1986b). In addition, a notice in the Federal Register went out on December 2, 1986, requesting public comment on the proposed fee to be paid (DOE 1986a). The total cost presented in the Federal Register notice, divided by the number of canisters presented in the same notice, calculates to a range of $165,000 to $214,000 per canister. The draft EIS costs were based on $35,000 to $45,000 per canister.

The retrieval and processing, transportation and disposal costs are summarized in Appendix L, Sections L.2.4 and L.4.4. These sections explain the source of the summary costs discussed in Section 3.4.1 and Chapter 5 of Volume 1. The costs, shown in Table J.1, were developed using the value of $214,000 per canister for high-level waste and the RECON model for TRU waste.

J.1 APPLICATION OF RECON MODEL FOR TRU WASTES

The primary tool used in developing cost estimates is the computer model RECON (Clark et al. 1983). The RECON computer model is a program for calculating life-cycle construction and operating costs for a geologic repository based on user-selected design characteristics and related cost inputs. Using the model, total repository cost estimates can be generated. More importantly, however, the cost impacts of repository design and waste scenario changes

TABLE J.1. **Incremental Repository Costs Associated with Emplacement of Hanford Defense Wastes**

Waste Class	Disposal Alternative	Canisters/Drums	Millions of $1987[a] Onsite	Offsite	WIPP
Existing Tank Waste	Geologic	19,800 canisters	4200	4300	NA[b]
	Reference	473 canisters[d]	100	110	NA
Future Tank Waste	Geologic	3,310 canisters	710	720	NA
	Reference	595 canisters[d]	130	130	NA
Sr/Cs Capsules	Geologic	509 canisters	110	110	NA
	Reference	509 canisters[d]	110	110	NA
TRU-Contaminated Soil Sites[c]	Geologic	64,000 drums			12
Pre-1970 TRU[c]	Geologic	205,000 drums			42
Retrievably Stored and Newly Generated TRU	Geologic	93,400 drums			12
	Reference	102,000 drums			13

(a) Costs were revised from the draft EIS to reflect increased proposed repository fees. Since these costs were calculated, further, increased repository fees have been proposed. If put into effect, these additional increases would increase costs for the geologic alternative by 20%, for the reference alternative by 5%, and for the preferred alternative by 5 to 20%. Although these changes do not affect the relative comparison of alternatives, they do widen the cost difference between the geologic and preferred alternatives and the other alternatives. However, the increase has not changed DOE's choice of a preferred alternative. Additional changes in estimated repository fees can be expected in the future.
(b) Not applicable.
(c) Waste not emplaced in geologic repository in the reference disposal alternative.
(d) Current estimates of total numbers of high-level waste canisters are available in Integrated Data Base for 1987: Spent Fuel and Radioactive Waste Inventories, Projections and Characteristics (DOE 1987).

can be estimated. The model has been validated against existing conceptual repository design and cost estimates. Although the model is less precise and provides less detailed cost information than would be generated by an architect-engineer, in validation runs all costs have fallen within 10% of the estimates, and the model can generate these estimates much more quickly and less expensively.

Use of RECON requires parametric data input to describe the repository. Basic model input parameters describe facilities, construction times, shafts, mine design, emplacement limitations, waste quantities available for disposal, waste processing parameters (labor, materials, utility, and equipment requirements), facility construction cost and unit labor, materials, utility and equipment costs. For the salt repository used in the WIPP estimates, design and economic data were obtained from recent draft studies of a commercial repository in salt.

Using the above information, RECON was used to determine the facility requirements for receiving, packaging, transporting, and emplacing the wastes. Based on the facility requirements, the model was used to calculate labor, materials, and equipment requirements. Requirements for replacing equipment were calculated based on equipment life (stated in years or units processed) and processing rates. All of the above requirements were calculated year by year for each waste class, thereby simulating actual repository operations.

J.2 REFERENCES

Clark L. L., B. M. Cole, G. W. McNair and M. E. Schutz. 1983. RECON: A Computer Program for Analyzing Repository Economics. PNL-4465, Pacific Northwest Laboratory, Richland, Washington.

Department of Energy (DOE). 1986a. "Calculating Nuclear Waste Fund Disposal Fees for DOE Defense Program Waste; Inquiry and Request for Public Comment." 51 FR 231, Washington, D.C.

Department of Energy (DOE). 1986b. Perspective on Methods to Calculate a Fee for Disposal of Defense High-Level Waste in Combined (Civilian/Defense) Repositories. DOE RL-86-10, U.S. DOE Richland Operations Office, Richland, Washington.

Department of Energy (DOE). 1987. Integrated Data Base for 1987: Spent Fuel and Radioactive Waste Inventories, Projections and Characteristics. DOE/RW-0006 Rev. 3, Washington, D.C.

Rockwell Hanford Operations. 1985. Hanford Defense Waste Disposal Alternatives: Engineering Support Data for the HDW-EIS. RHO-RE-ST-30 P, Richland, Washington.

Rockwell Hanford Operations. 1987. Engineering Support Data Update for the Hanford Defense Waste-Environmental Impact Statement. RHO-RE-ST-30 ADD P, Richland, Washington.

THIS PAGE INTENTIONALLY
LEFT BLANK

APPENDIX K

SOCIOECONOMIC IMPACTS

THIS PAGE INTENTIONALLY
LEFT BLANK

APPENDIX K

SOCIOECONOMIC IMPACTS

This appendix compares the alternatives analyzed in this EIS in terms of their socioeconomic impacts and provides details in support of Sections 4.8 and 5.7 of Vol. 1. The main determinants of socioeconomic impact are the pressures (demographic, fiscal, services, and social) a project places upon a community and the community's ability to meet those pressures in a planned, orderly, and cost-effective way. Socioeconomic impacts of a facility involving potentially hazardous materials include both growth-related effects (e.g., demand for housing and schooling, traffic congestion) and social, cultural or psychological effects related to the hazardous nature of the materials or technology involved (e.g., apprehension about the nuclear industry in general, concern for the risks involved in safely managing such materials, and stress resulting from perceived adverse consequences).

Since the size and scheduling of the work force are major factors affecting socioeconomic impacts, special attention was given to the manpower needed for construction and operation of each of the four alternatives. On the other hand, because the objective of this EIS is to provide a choice among alternative strategies for substantially improving the safety of defense waste management and reducing the potential for adverse environmental impacts, social and psychological impacts should be reduced on balance, and less attention has been given to assessing the socioeconomic impacts resulting from the hazardous nature of the materials and technologies involved.

The largest work force requirements are expected to occur between 1985 and 2020, the study period used in this analysis. The manpower data used in this analysis are preliminary and subject to refinement. The study area for the socioeconomic analysis is defined as Benton and Franklin counties, and numerous prior studies of facility development at the Hanford Site have demonstrated that the great majority of standard, growth-related socioeconomic impacts are experienced within these two counties. Given the recent economic history of this area, new workers are unlikely to residentially concentrate elsewhere. By concentrating all projected population and employment effects in this area, the analysis assures identification of potential adverse effects on public services and facilities in the Tri-Cities area due to the proposed alternatives.

This Appendix is not intended to provide a comprehensive socioeconomic baseline study of the Hanford Site. The approach taken here for the assessment of potential socioeconomic impacts is to recognize that the alternative disposal strategies reduce radiological risk to the environment rather than increase it. The objective is to determine which specific strategy will result in the least impact, socioeconomic and otherwise, so that the best choice can be made. The socioeconomic assessment draws on existing studies that report baseline conditions in the study area, and presents those data that are considered pertinent to evaluating the range of impacts presented by the "bounding analysis" in this EIS. Since the magnitude of the growth-related changes that could be caused by waste disposal activities is believed to be small relative to recent experience in the area and its capacity to absorb such growth effects, socioeconomic impacts are expected to be small and not significant. Also, because the projected radiological exposures calculated for each of the alternatives, including postulated upper-bound accidents, are expected to be small relative to both background and to the no action alternative, social impacts, including effects of perceived risk, are also expected to be small and not significant. Given that the absolute level of impact is expected to be very low, the socioeconomic analysis will focus primarily on differences in effects due to each alternative in order to facilitate the decision process.

K.1 WORK FORCE REQUIREMENTS

Construction and operation activities for the disposal alternatives (geologic, in-place stabilization, reference and preferred) are, for purposes of this analysis, postulated to begin between 1985 and 1990 and to last at least 50 years. In constructing a work force profile for each alternative, the work force estimates for each waste class (Rockwell 1985) were matched with their respective construction and operation schedules. The total expected worker-years were distributed throughout the specified activities associated with each alternative for each waste class. The overall work force profile was determined by adding the components for each activity for each of the six waste classes. A detailed distribution of work force data is shown in Tables K.1 through K.4 and in Figures K.1 through K.4. Only the first 45 years of construction and operation are shown in these tables because it is clear that the work force requirements are small and decline rapidly after that point.

It is important to note that, unlike most large construction projects, the activities of the construction and operation work forces associated with these alternatives overlap substantially, both in terms of scheduling and actual work requirements. Although socioeconomic impacts normally vary due to differences in temporary construction workers and permanent operations workers, those distinctions are blurred in this kind of activity. Therefore, the construction and operations workers are treated the same way in this socioeconomic analysis.

As seen in these tables, existing and future tank waste disposal activities on site account for about 90% of the expected work force requirements under each of the disposal alternatives. Several additional points need to be made about these work force data. First, the worker-years represented in the 35-year time span covered in this analysis are less than the total work force requirements because some workers will be needed in subsequent years for surveillance and maintenance. Also, workers employed in offsite disposal [i.e., retrievably stored and newly generated transuranic (TRU) waste sent to the Waste Isolation Pilot Plant] do not contribute to socioeconomic consequences in this study and are, therefore, not included. Second, the work force characteristics and distribution are subject to revision; future refinement of the construction and operation activities could result in somewhat modified work force requirements. Third, although the skill requirements of the work force influence socioeconomic impacts, no information is currently available about the skill requirements of any alternative. Generally, however, the labor force in the study area is considered well suited to this type of work, particularly that required during the construction period.

TABLE K.1. Estimated Work Force Requirements for the Geologic Disposal Alternative by Waste Class, 1985 to 2020, Worker-Year[a]

Year	Existing Tank Waste	Future Tank Waste	Sr/Cs Capsules	TRU-Contaminated Soil Sites	Pre-1970 Buried TRU Solid Waste	Retrievably Stored & Newly Generated TRU	Total
1985							
1986							
1987							
1988							
1989							
1990	1,987	257					2,244
1991	1,988	257	25	134	241	52	2,697
1992	1,988	257	25	134	241	52	2,697
1993	1,988	257	25	177	983	17	3,447
1994	1,987	257	25	177	983	16	3,445
1995	1,322	257	67	43	742	16	2,332
1996	1,322	257	67	20	266	15	1,832
1997	1,322	257	67	20	266	15	1,832
1998	1,322	257		20	266	15	1,765
1999	1,322	257		20	266	15	1,765
2000	1,322	257		20	266	15	1,765
2001	1,322	257		20	266	15	1,765
2002	1,322	257		20	266	15	1,765
2003	1,322	257		20	266	15	1,765
2004	1,322	257		20	266	15	1,765
2005	1,322	257		20	266	15	1,765
2006	1,322	257		20	266	15	1,765
2007	1,322	257		20	266	15	1,765
2008	1,322	257		20	266	15	1,765
2009	1,322	257		20	266	15	1,765
2010	1,322	257		20	266	15	1,765
2011	1,322	257		20	266	15	1,765
2012	1,322	257		20	266	15	1,765
2013				20	266	15	301
2014				20	266	15	301
2015				20	266	15	301
2016							
2017							
2018							
2019							
2020							

(a) Rockwell 1985.

TABLE K.2. **Estimated Work Force Requirements for the In-Place Stabilization and Disposal Alternative by Waste Class, 1985 to 2020, Worker-Year[a]**

Year	Existing Tank Waste	Future Tank Waste	Sr/Cs Capsules	TRU-Contaminated Soil Sites	Pre-1970 Buried TRU Solid Waste	Retrievably Stored & Newly Generated TRU	Total
1985							
1986							
1987	80	51					131
1988	101	117					218
1989	101	117					218
1990	130	26		24	87		267
1991	130	26					156
1992	484	26					510
1993	485	26				15	526
1994	485	26				15	526
1995	303	280				15	598
1996	298	274					572
1997	298	274					572
1998	298	274					572
1999	298	274					572
2000	50						50
2001	50						50
2002	50						50
2003	50						50
2004	50						50
2005	50						50
2006	50		54				104
2007	50		54				104
2008	50		54				104
2009	50		54				104
2010	50	304	138				492
2011	50		138				188
2012	50		138				188
2013	50		138				188
2014	50		3				53
2015	50		3				53
2016	50						50
2017	50						50
2018	50						50
2019	50						50
2020	50						50

(a) Rockwell 1985.

TABLE K.3. Estimated Work Force Requirements for the Reference (combination) Alternative by Waste Class, 1985 to 2020, Worker-Year[a]

Year	Existing Tank Waste	Future Tank Waste	Sr/Cs Capsules	TRU-Contaminated Soil Sites	Pre-1970 Buried TRU Solid Waste	Retrievably Stored & Newly Generated TRU	Total
1985	121	128					249
1986	121						121
1987	9	6					15
1988	29	26					55
1989	189	213					402
1990	300	205		24	87	43	659
1991	300	205	25			43	573
1992	482	47	25			43	597
1993	572	143	25			1	741
1994	572	143	25			1	741
1995	293	319	67			1	680
1996	293	319	67			17	696
1997	293	319	67			17	696
1998	292	318				17	627
1999	292	318				18	628
2000	164	138				18	320
2001	164	138				18	320
2002	164	138				18	320
2003	66	37				18	121
2004	66	37				18	121
2005	66	37				18	121
2006	66	37				18	121
2007	66	37				18	121
2008	66	37				18	121
2009	66	37				18	121
2010	66	37				18	121
2011	66	37				18	121
2012	66	37				18	121
2013	66	37				17	120
2014	66	37				17	120
2015	66	37				17	120
2016	66	37					103
2017	66	37					103
2018	66	37					103
2019	66	37					103
2020	66	37					103

(a) Rockwell 1985.

TABLE K.4. **Estimated Work Force Requirements for the No Disposal Action** (continued storage) **Alternative by Waste Class, 1985 to 2020, Worker-Year**[a]

Year	Existing Tank Waste	Future Tank Waste	Sr/Cs Capsules	TRU-Contaminated Soil Sites	Pre-1970 Buried TRU Solid Waste	Retrievably Stored & Newly Generated TRU	Total
1985	108	9		1	1	1	120
1986	108	9		1	1	1	120
1987	108	9		1	1	1	120
1988	108	9		1	1	1	120
1989	108	9		1	1	1	120
1990	108	9		1	1	1	120
1991	108	9		1	1	1	120
1992	108	9		1	1	1	120
1993	108	9		1	1	1	120
1994	108	9		1	1	1	120
1995	108	9		1	1	1	120
1996	108	9		1	1	1	120
1997	108	9		1	1	1	120
1998	108	9		1	1	1	120
1999	108	9		1	1	1	120
2000	108	9		1	1	1	120
2001	108	9		1	1	1	120
2002	108	9		1	1	1	120
2003	108	9		1	1	1	120
2004	108	9		1	1	1	120
2005	108	9		1	1	1	120
2006	108	9	52	1	1	1	172
2007	109	9	52	1	1	1	173
2008	108	9	53	1	1	1	173
2009	109	9	52	1	1	1	172
2010	108	9	282	1	1	1	402
2011	108	9	282	1	1	1	402
2012	108	9	282	1	1	1	402
2013	108	9	282	1	1	1	402
2014	108	9	7	1	1	1	127
2015	108	9	7	1	1	1	127
2016	108	9	7	1	1	1	127
2017	108	9	7	1	1	1	127
2018	108	9	7	1	1	1	127
2019	108	9	7	1	1	1	127
2020	108	9	7	1	1	1	127

(a) Rockwell 1985.

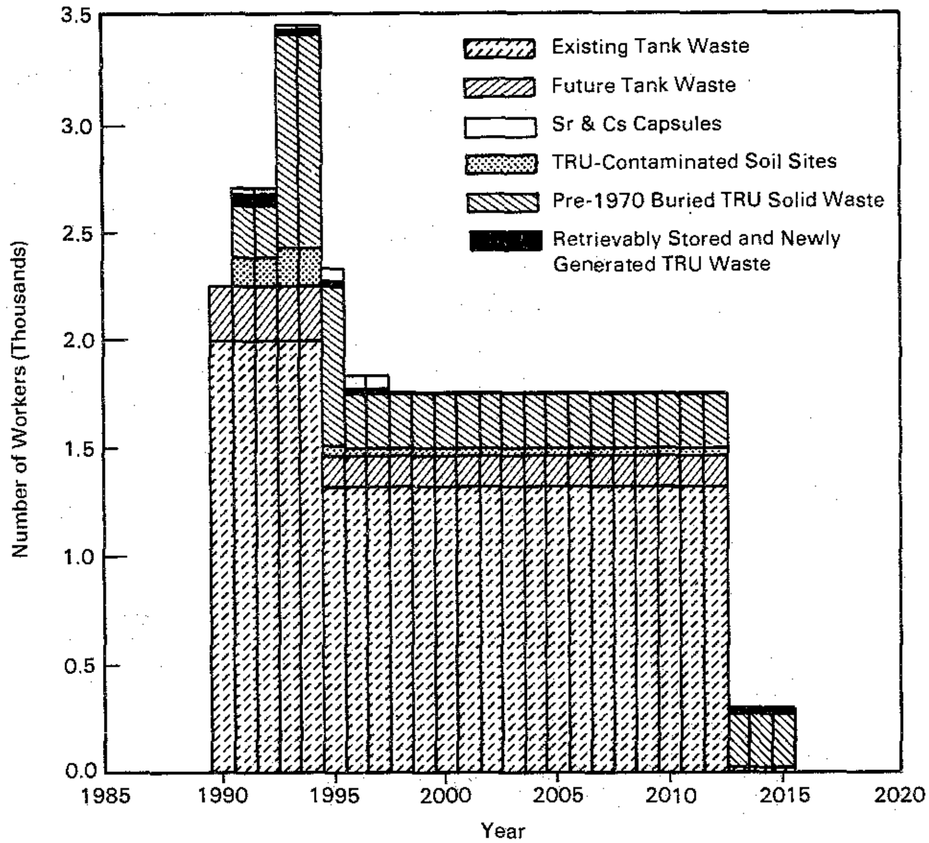


FIGURE K.1. Work Force Requirements for the Geologic Disposal Alternative

Figures K.1 through K.4 show that work force requirements for the no disposal action (continued storage) alternative, in-place stabilization and disposal, and the reference (combined) alternatives are relatively low compared with those of the geologic disposal alternative. Between 1990 and 2015, the average number of workers required per year year for the geologic disposal alternative is 5 to 11 times the requirement for each of the other three alternatives; its peak work force requirement is far more pronounced. Since the potential for socioeconomic impacts tends to be directly related to the size and geographic concentration of the work force, it is apparent that the geologic disposal alternative would have the greatest potential to cause socioeconomic impact. Impacts of the preferred alternative will depend upon the final disposal decision for the classes of waste for which no disposal decision is to be made at this time. However, the impacts would be bounded by the geologic and reference alternatives.

K.2 EMPLOYMENT AND POPULATION IMPACTS

Increased work force requirements may induce population growth in an area and put pressure on community services and social conditions. The extent of this impact depends largely on the availability of unemployed or underemployed workers already in the area who are qualified and available to work on these jobs. The availability of local workers can be greatly

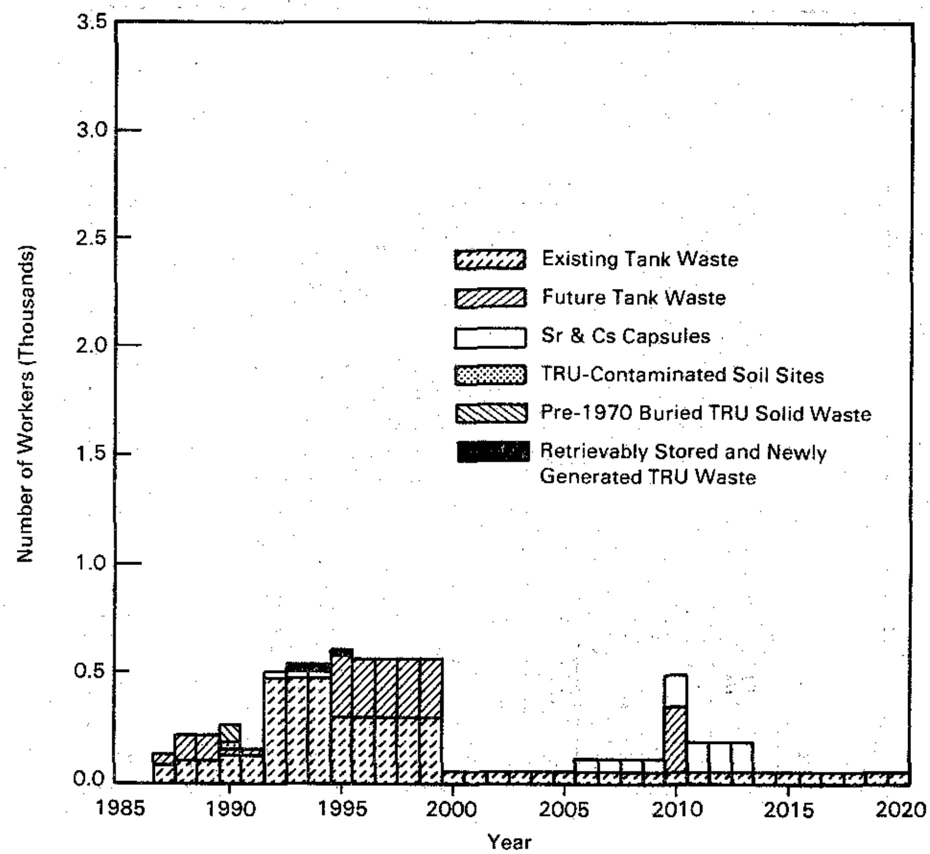


FIGURE K.2. Work Force Requirements for the In-Place Stabilization and Disposal Alternative

affected by the timing and work force requirements of other major projects that also place demands on the local labor supply. Multiple activities that draw from a common labor pool essentially compete for scarce labor resources. When the supply of local workers having the needed skills is less than the demand, workers will either in-migrate or commute over long distances. Because of the importance of this effect and the uncertainty about the schedule and work force requirements of other potential major projects in this area, two baseline projections have been prepared. One assumes limited growth in employment in the study area between 1984 and 2020, while the other assumes much more rapid growth. Historical and projected levels of total employment and population for the baseline (without waste management activities) are shown in Table K.5 and Figure K.5.

The economic and demographic growth experienced in the study area between 1973 and 1981 was caused primarily by growth in employment due to the construction of the Washington Public Power Supply System nuclear power reactors. During this period, Supply System-related employment increased at an average rate of about 39% per year. This growth was supplemented by an annual growth rate of 4.2% in the agricultural sector, 5.6% in DOE-related activities, 6.6% in manufacturing, 7.3% in service-based and retail/wholesale industries, and 6.0% in the government sector. The overall rate of employment growth during the 1973 to 1981 period was

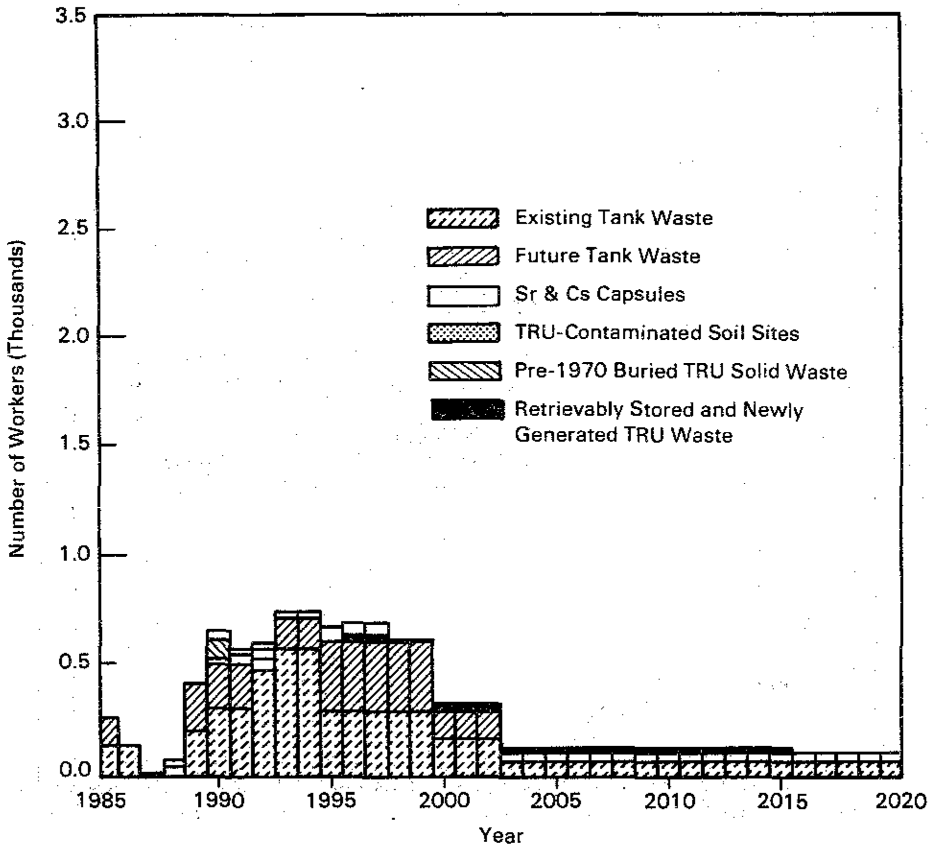


FIGURE K.3. **Work Force Requirements for the Reference** (combination) **Alternative**

8.3% per year. During this rapid growth period, the population of the study area expanded rapidly, and many new families settled in the area. After 1981, however, employment in the study area declined and out-migration of population occurred, due primarily to the mothballing of the Washington Public Power Supply System Unit 1 nuclear plant (WNP-1) and termination of the Supply System Unit 4 (WNP-4).

For the low baseline condition, the study area is assumed to undergo a gradual economic recovery after 1985, with employment growing at a steady 0.6% per year through 2020. Under the low baseline, 1981 employment and population levels are not reached until about 2019.

For the high baseline condition, work on WNP-1 is assumed to restart in 1988 and be completed in 1993. After 1995, employment in the study area is assumed to grow at about 1.9% per year, with population growing at 1.3% per year. These rates are consistent with the pre-Supply System period. Under the high baseline condition, the study area would reach 1981 employment and population levels by 1989. While it is recognized that a restart of WNP-1 and its timing is highly uncertain, such a large additional potential project activity is included here to account for the cumulative effects of multiple Hanford Site developments that could lead to greater socioeconomic consequences than implementation of the HDW alternative alone.

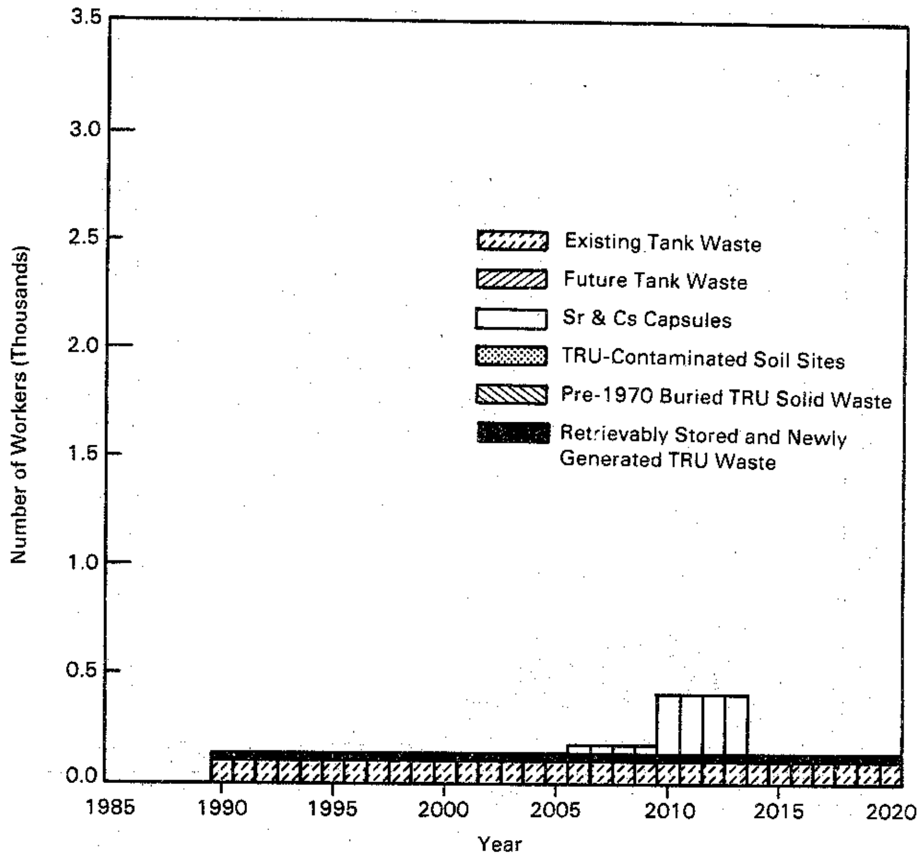


FIGURE K.4. Work Force Requirements for the No Disposal Action (continued storage) **Alternative**

Tables K.6 through K.9 show the total employment effects projected for each of the defense waste alternatives. Total employment effects include both the direct employment of workers by the alternative (primary employment) and the secondary employment created by project-related purchases and worker expenditures in the study area. Secondary employment was estimated for each alternative by multiplying direct employment by the ratio (the total employment multiplier) of total employment to primary sector employment (composed of Supply System, DOE, and agriculture) averaged over the period from 1973 to 1981. During this period, this ratio averaged 2.2; that is, for every primary sector job an additional 1.2 secondary jobs were created in the area.

As shown in these tables, the projected average total (primary and secondary) employment effect of the geologic disposal alternative between 1990 and 2000 is about 5,164 workers per year, with a peak of about 7,600 workers in 1993 and 1994 (Table K.6). This is almost 20 times the average total employment effect projected for the no disposal action (continued storage) alternative over this period (264 workers). As a percentage of the projected high baseline employment, the employment effects of the alternatives range from less than 1% to slightly less than 10% (in 1993 and 1994 for the geologic alternative). For comparison,

TABLE K.5. Projected Baseline Employment and Population

Year	High Baseline Employment[a,b]	High Baseline Population[c]	Low Baseline Employment[a,b]	Low Baseline Population[c]
1981	75,636	148,056	75,636	148,056
1982	69,736	143,631	69,736	143,631
1983	67,336	141,831	67,336	141,831
1984	62,936	138,531	62,936	138,531
1985	63,171	138,707	62,389	138,121
1986	64,757	139,897	62,716	138,366
1987	66,430	141,151	63,046	138,614
1988	68,989	143,071	63,381	138,864
1989	81,289	157,830	63,719	139,118
1990	86,807	164,452	64,061	139,375
1991	86,763	164,400	64,407	139,634
1992	86,151	163,665	64,757	139,897
1993	80,732	157,162	65,111	140,162
1994	82,166	158,883	65,470	140,431
1995	83,636	160,647	65,832	140,703
1996	85,141	162,452	66,199	140,978
1997	86,682	164,302	66,569	141,256
1998	88,261	166,197	66,945	141,537
1999	89,878	168,138	67,324	141,822
2000	91,535	170,126	67,708	142,110
2001	93,232	172,162	68,096	142,401
2002	94,970	174,248	68,489	142,695
2003	96,751	176,385	68,886	142,993
2004	98,575	178,574	69,288	143,295
2005	100,444	180,817	69,695	143,600
2006	102,245	182,977	69,992	143,822
2007	104,204	185,329	70,405	144,133
2008	106,214	187,740	70,826	144,448
2009	108,275	190,214	71,254	144,769
2010	109,879	192,139	71,179	144,713
2011	112,041	194,733	71,614	145,040
2012	114,256	197,391	72,055	145,370
2013	116,525	200,114	72,501	145,704
2014	119,456	203,631	73,557	146,496
2015	121,839	206,491	74,013	146,838
2016	124,281	209,421	74,474	147,185
2017	126,784	212,424	74,941	147,535
2018	129,348	215,501	75,413	147,889
2019	131,976	218,655	75,891	148,462
2020	134,670	221,887	76,374	149,042

(a) 1981-1984 figures from Washington State Employment Security Department, 1984. Primary and secondary employment required for the no disposal action (continued storage) alternative (see Table K.9) have been subtracted out of the employment figures to provide the baseline.

(b) Projections for 1985 and following are based on these assumptions (see Cluett et al. 1984 for detailed discussion of procedures used): 1) decline of Supply System employment to 920 in 1985 followed by the restart of WNP-1 in 1988 with completion in 1993; 2) growth in DOE and contractor employment by 5% per year from 1984 on; 3) growth in agricultural employment by 1% per year until 1985, increasing to 6% per year from 1986 to 1991, and an annual rate of 3% thereafter; 4) because of current excess labor supply, the multiplier effect of Supply System, DOE, and agriculture sector employment is assumed to be reduced to 1.5 through 1988; after 1988 the multiplier is assumed to be 2.2 for the high baseline. For the low baseline condition, the multiplier is assumed to be 1.5 through 2020.

(c) Because of current excess labor supply in the area, the population multiplier is assumed to be 0.75 until 1981 employment levels are reached and 1.2 thereafter.

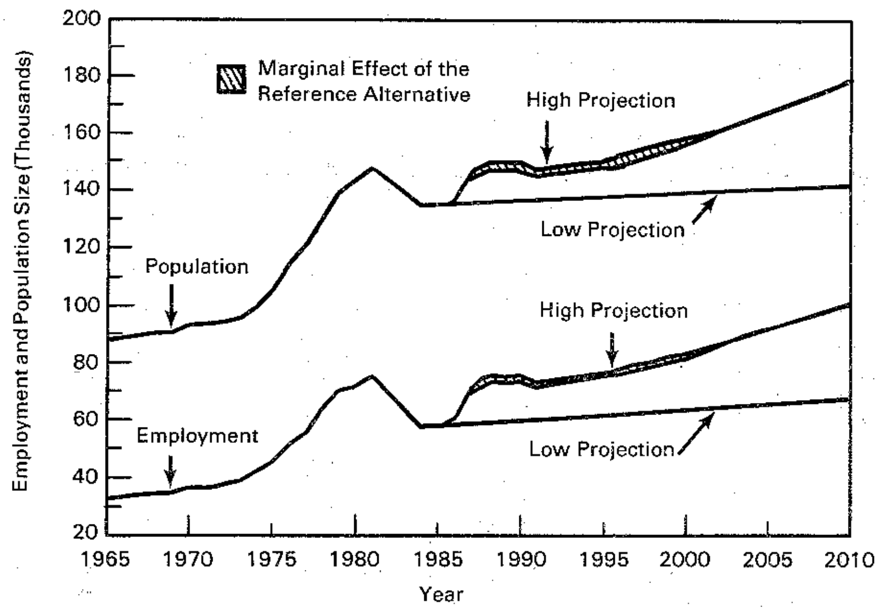


FIGURE K.5. **Historical and Projected Employment and Population for Benton/Franklin Counties**

total primary and secondary employment from the Supply System is estimated to have accounted for about 33% of study area employment in 1981.

The population effects of each alternative were projected using a similar analytic technique. For the pre-Supply System period of 1965 to 1973, the ratio of population to employment (the population multiplier) was calculated to provide a basis for estimating the increase in population that would result from the employment created by the proposed alternatives. For this period, the average employment multiplier was 1.2. That is, every new job resulted, on average, in a population increase of 1.2 persons. To account for the large number of available workers in the study area due to current depressed economic conditions, this population multiplier was adjusted downward to 0.75 until total study area employment reached 1981 levels.

Tables K.10 through K.13 show the projected population effects of each alternative under high and low baseline conditions. The projected peak population effects of the alternatives under the high baseline condition range from a high of about 9,100 people (5.9% of baseline population) in 1993 for the geologic disposal alternative to about 1,600 people (1% of baseline) in 1995 for the in-place stabilization and disposal alternative and about 2,000 people (1.3% of baseline) in 1993 and 1994 for the combined alternative, to a low of about 1,100 people (0.6% of baseline) in 2010 to 2013 for the no disposal action (continued storage) alternative. The population effects for all alternatives are considerably smaller under the low baseline alternative (peak for the geologic disposal alternative is 5,700). However, because of the lower baseline population, the population effect as a percent of baseline population is similar for both high and low baseline conditions.

TABLE K.6. **Projected Primary and Secondary Employment Under High and Low Baseline Conditions for the Geologic Disposal Alternative**

Year	Number of Primary and Secondary Workers[a]	Percent of High Baseline Employment[b]	Percent of Low Baseline Employment[b]
1990	4,937	5.7	7.7
1991	5,933	6.8	9.2
1992	5,933	6.9	9.2
1993	7,583	9.4	11.6
1994	7,579	9.2	11.6
1995	5,138	6.1	7.8
1996	4,030	4.7	6.1
1997	4,030	4.6	6.1
1998	3,883	4.4	5.8
1999	3,883	4.3	5.8
2000	3,883	4.2	5.7
2001	3,883	4.2	5.7
2002	3,883	4.1	5.7
2003	3,883	4.0	5.6
2004	3,883	3.9	5.6
2005	3,883	3.9	5.6
2006	3,883	3.8	5.5
2007	3,883	3.7	5.5
2008	3,883	3.7	5.5
2009	3,883	3.6	5.4
2010	3,883	3.5	5.5
2011	3,883	3.5	5.4
2012	3,883	3.4	5.4
2013	662	0.6	0.9
2014	662	0.6	0.9
2015	662	0.5	0.9

(a) Primary workers are those shown in Tables K.1 through K.4. Secondary workers are estimated to equal 1.2 times primary workers; total workers equals primary plus secondary. (See text for discussion.)
(b) See Table K.5 for baseline employment figures and an explanation of methods.

TABLE K.7. **Projected Primary and Secondary Employment Under High and Low Baseline Conditions for the In-Place Stabilization and Disposal Alternative**

Year	Number of Primary and Secondary Workers[a]	Percent of High Baseline Employment[b]	Percent of Low Baseline Employment[b]
1987	288	0.4	0.5
1988	480	0.7	0.8
1989	480	0.6	0.8
1990	587	0.7	0.9
1991	343	0.4	0.5
1992	1,122	1.3	1.7
1993	1,157	1.4	1.8
1994	1,157	1.4	1.8
1995	1,316	1.6	2.0
1996	1,258	1.5	1.9
1997	1,258	1.5	1.9
1998	1,258	1.4	1.9
1999	1,258	1.4	1.9
2000	110	0.1	0.2
2001	110	0.1	0.2
2002	110	0.1	0.2
2003	110	0.1	0.2
2004	110	0.1	0.2
2005	110	0.1	0.2
2006	229	0.2	0.3
2007	229	0.2	0.3
2008	229	0.2	0.3
2009	229	0.2	0.3
2010	1,082	1.0	1.5
2011	414	0.4	0.6
2012	414	0.4	0.6
2013	414	0.4	0.6
2014	117	0.1	0.2
2015	117	0.1	0.2
2016	110	0.1	0.1
2017	110	0.1	0.1
2018	110	0.1	0.1
2019	110	0.1	0.1
2020	110	0.1	0.1

(a) Primary workers are those shown in Tables K.1 through K.4. Secondary workers are estimated to equal 1.2 times primary workers; total workers equals primary plus secondary. (See text for discussion.)
(b) See Table K.5 for baseline employment figures and an explanation of methods.

TABLE K.8. Projected Primary and Secondary Employment Under High and Low Baseline Conditions for the Reference (combination) Alternative

Year	Number of Primary and Secondary Workers[(a)]	Percent of High Baseline Employment[(b)]	Percent of Low Baseline Employment[(b)]
1985	548	0.9	0.9
1986	266	0.4	0.4
1987	33	0.0	0.1
1988	121	0.2	0.2
1989	884	1.1	1.4
1990	1,450	1.7	2.3
1991	1,261	1.5	2.0
1992	1,313	1.5	2.0
1993	1,630	2.0	2.5
1994	1,630	2.0	2.5
1995	1,496	1.8	2.3
1996	1,531	1.8	2.3
1997	1,531	1.8	2.3
1998	1,379	1.6	2.1
1999	1,382	1.5	2.1
2000	704	0.8	1.0
2001	704	0.8	1.0
2002	704	0.7	1.0
2003	266	0.3	0.4
2004	266	0.3	0.4
2005	266	0.3	0.4
2006	266	0.3	0.4
2007	266	0.3	0.4
2008	266	0.3	0.4
2009	266	0.2	0.4
2010	266	0.2	0.4
2011	266	0.2	0.4
2012	266	0.2	0.4
2013	264	0.2	0.4
2014	264	0.2	0.4
2015	264	0.2	0.4
2016	227	0.2	0.3
2017	227	0.2	0.3
2018	227	0.2	0.3
2019	227	0.2	0.3
2020	227	0.2	0.3

(a) Primary workers are those shown in Tables K.1 through K.4. Secondary workers are estimated to equal 1.2 times primary workers; total workers equals primary plus secondary. (See text for discussion.)
(b) See Table K.5 for baseline employment figures and an explanation of methods.

TABLE K.9. **Projected Primary and Secondary Employment Under High and Low Baseline Conditions for the No Disposal Action** (continued storage) **Alternative**

Year	Number of Primary and Secondary Workers[a]	Percent of High Baseline Employment[b]	Percent of Low Baseline Employment[b]
1985	264	0.4	0.4
1986	264	0.4	0.4
1987	264	0.4	0.4
1988	264	0.4	0.4
1989	264	0.3	0.4
1990	264	0.3	0.4
1991	264	0.3	0.4
1992	264	0.3	0.4
1993	264	0.3	0.4
1994	264	0.3	0.4
1995	264	0.3	0.4
1996	264	0.3	0.4
1997	264	0.3	0.4
1998	264	0.3	0.4
1999	264	0.3	0.4
2000	264	0.3	0.4
2001	264	0.3	0.4
2002	264	0.3	0.4
2003	264	0.3	0.4
2004	264	0.3	0.4
2005	264	0.3	0.4
2006	378	0.4	0.5
2007	381	0.4	0.5
2008	381	0.4	0.5
2009	378	0.3	0.5
2010	884	0.8	1.2
2011	884	0.8	1.2
2012	884	0.8	1.2
2013	884	0.8	1.2
2014	279	0.2	0.4
2015	279	0.2	0.4
2016	279	0.2	0.4
2017	279	0.2	0.4
2018	279	0.2	0.4
2019	279	0.2	0.4
2020	279	0.2	0.4

(a) Primary workers are those shown in Tables K.1 through K.4. Secondary workers are estimated to equal 1.2 times primary workers; total workers equals primary plus secondary. (See text for discussion.)
(b) See Table K.5 for baseline employment figures and an explanation of methods.

TABLE K.10. **Projected New Population Attracted to the Study Area Under High and Low Baseline Conditions for the Geologic Disposal Alternative**

	High Baseline Conditions[a]		Low Baseline Conditions[a]	
Year	In-Migrant Population[b]	Percent of Baseline	In-Migrant Population[b]	Percent of Baseline
1990	5,924	3.6	3,703	3.5
1991	7,120	4.3	4,450	4.2
1992	7,120	4.4	4,450	4.2
1993	9,100	5.8	5,688	5.4
1994	9,095	5.7	5,684	5.4
1995	6,156	3.8	3,848	3.6
1996	4,836	3.0	3,023	2.9
1997	4,836	2.9	3,023	2.9
1998	4,660	2.8	2,912	2.7
1999	4,660	2.8	2,912	2.7
2000	4,660	2.7	2,912	2.7
2001	4,660	2.7	2,912	2.7
2002	4,660	2.7	2,912	2.7
2003	4,660	2.6	2,912	2.7
2004	4,660	2.6	2,912	2.7
2005	4,660	2.6	2,912	2.7
2006	4,660	2.5	2,912	2.7
2007	4,660	2.5	2,912	2.7
2008	4,660	2.5	2,912	2.7
2009	4,660	2.4	2,912	2.7
2010	4,660	2.4	2,912	2.7
2011	4,660	2.4	2,912	2.7
2012	4,660	2.4	2,912	2.7
2013	795	0.4	497	0.5
2014	795	0.4	497	0.5
2015	795	0.4	497	0.5

(a) For baseline population figures and method of derivation, see Table K.5.
(b) In-migrant population was projected from total employment (Tables K.6 through K.9) using a population multiplier of 0.75 until 1981 employment levels were reached (1988 for the high baseline and 2019 for the low baseline) and 1.2 thereafter.

TABLE K.11. **Projected New Population Attracted to the Study Area Under High and Low Baseline Conditions for the In-Place Stabilization and Disposal Alternative**

	High Baseline Conditions[a]		Low Baseline Conditions[a]	
Year	In-Migrant Population[b]	Percent of Baseline	In-Migrant Population[b]	Percent of Baseline
1987	216	0.2	216	0.2
1988	576	0.4	360	0.3
1989	576	0.4	360	0.3
1990	705	0.4	441	0.4
1991	412	0.3	257	0.2
1992	1,346	0.8	842	0.8
1993	1,389	0.9	868	0.8
1994	1,389	0.9	868	0.8
1995	1,579	1.0	987	0.9
1996	1,510	0.9	944	0.9
1997	1,510	0.9	944	0.9
1998	1,510	0.9	944	0.9
1999	1,510	0.9	944	0.9
2000	132	0.1	83	0.1
2001	132	0.1	83	0.1
2002	132	0.1	83	0.1
2003	132	0.1	83	0.1
2004	132	0.1	83	0.1
2005	132	0.1	83	0.1
2006	275	0.2	172	0.2
2007	275	0.1	172	0.2
2008	275	0.1	172	0.2
2009	275	0.1	172	0.2
2010	1,299	0.7	812	0.7
2011	496	0.3	310	0.3
2012	496	0.3	310	0.3
2013	496	0.2	310	0.3
2014	140	0.1	87	0.1
2015	140	0.1	87	0.1
2016	132	0.1	83	0.1
2017	132	0.1	83	0.1
2018	132	0.1	83	0.1
2019	132	0.1	132	0.1
2020	132	0.1	132	0.1

(a) See Table K.5 for baseline population figures and method of derivation.
(b) In-migrant population was projected from total employment (Tables K.6 through K.9) using a population multiplier of 0.75 until 1981 employment levels were reached and 1.2 thereafter.

TABLE K.12. Projected New Population Attracted to the Study Area Under High and Low Baseline Conditions for the Reference (combination) **Alternative**

	High Baseline Conditions[a]		Low Baseline Conditions[a]	
Year	In-Migrant Population[b]	Percent of Baseline	In-Migrant Population[b]	Percent of Baseline
1985	411	0.3	411	0.4
1986	200	0.1	200	0.2
1987	25	0.0	25	0.0
1988	145	0.1	91	0.1
1989	1,061	0.7	663	0.6
1990	1,740	1.1	1,087	1.0
1991	1,513	0.9	945	0.9
1992	1,576	1.0	985	0.9
1993	1,956	1.2	1,223	1.2
1994	1,956	1.2	1,223	1.2
1995	1,795	1.1	1,122	1.1
1996	1,837	1.1	1,148	1.1
1997	1,837	1.1	1,148	1.1
1998	1,655	1.0	1,035	1.0
1999	1,658	1.0	1,036	1.0
2000	845	0.5	528	0.5
2001	845	0.5	528	0.5
2002	845	0.5	528	0.5
2003	319	0.2	200	0.2
2004	319	0.2	200	0.2
2005	319	0.2	200	0.2
2006	319	0.2	200	0.2
2007	319	0.2	200	0.2
2008	319	0.2	200	0.2
2009	319	0.2	200	0.2
2010	319	0.2	200	0.2
2011	319	0.2	200	0.2
2012	319	0.2	200	0.2
2013	317	0.2	198	0.2
2014	317	0.2	198	0.2
2015	317	0.2	198	0.2
2016	272	0.1	170	0.2
2017	272	0.1	170	0.2
2018	272	0.1	170	0.2
2019	272	0.1	272	0.2
2020	272	0.1	272	0.2

(a) For baseline population figures and method of derivation, see Table K.5.
(b) In-migrant population was projected from total employment (Tables K.6 through K.9) using a population multiplier of 0.75 until 1981 employment levels were reached and 1.2 thereafter.

TABLE K.13. **Projected New Population Attracted to the Study Area Under High and Low Baseline Conditions for the No Disposal Action** (continued storage) **Alternative**

	High Baseline Conditions[a]		Low Baseline Conditions[a]	
Year	In-Migrant Population[b]	Percent of Baseline	In-Migrant Population[b]	Percent of Baseline
1985	198	0.1	198	0.2
1986	198	0.1	198	0.2
1987	198	0.1	198	0.2
1988	317	0.2	198	0.2
1989	317	0.2	198	0.2
1990	317	0.2	198	0.2
1991	317	0.2	198	0.2
1992	317	0.2	198	0.2
1993	317	0.2	198	0.2
1994	317	0.2	198	0.2
1995	317	0.2	198	0.2
1996	317	0.2	198	0.2
1997	317	0.2	198	0.2
1998	317	0.2	198	0.2
1999	317	0.2	198	0.2
2000	317	0.2	198	0.2
2001	317	0.2	198	0.2
2002	317	0.2	198	0.2
2003	317	0.2	198	0.2
2004	317	0.2	198	0.2
2005	317	0.2	198	0.2
2006	454	0.2	284	0.3
2007	457	0.2	285	0.3
2008	457	0.2	285	0.3
2009	454	0.2	284	0.3
2010	1,061	0.6	663	0.6
2011	1,061	0.6	663	0.6
2012	1,061	0.5	663	0.6
2013	1,061	0.5	663	0.6
2014	335	0.2	210	0.2
2015	335	0.2	210	0.2
2016	335	0.2	210	0.2
2017	335	0.2	210	0.2
2018	335	0.2	210	0.2
2019	335	0.2	335	0.2
2020	335	0.2	335	0.2

(a) For baseline population figures and method of derivation, see Table K.5.
(b) In-migrant population was projected from total employment (Tables K.6 through K.9) using a population multiplier of 0.75 until 1981 employment levels were reached and 1.2 thereafter.

Compared to the area's recent experience, the magnitude of population growth caused by any of the proposed alternatives is moderate and, from an economic and demographic standpoint, especially under the low baseline condition, can be seen as alleviating the depressed conditions in the study area by reducing the levels of unemployment and underemployment among area residents. Because of the uncertainty about future baseline employment and population conditions in the study area, it will be essential to monitor the labor force, employment requirements, and migration patterns in the study area throughout the study period.

K.3 IMPACTS ON COMMUNITY SERVICES

New work force and population moving into Benton and Franklin counties in response to employment opportunities associated with the proposed alternatives will require housing and a range of community support services, including transportation, health care, schools, police and fire, water and sewer, and recreation facilities. The potential socioeconomic effects of each of the four alternatives can be estimated by comparing the likely demand for these services with estimates of their availability. Based on previous expectations of continued rapid growth during the late 1970s, in many areas community services facilities were expanded beyond current needs. Because of this, and since the study area will be in the process of recovering from the significant employment and population losses of the early 1980s at the time of the heaviest manpower requirements projected for the various alternatives, most of the services mentioned above have sufficient capacity to meet projected demand during the early portion of the study period. Previous experience in responding to population growth is expected to facilitate the development of any additional services that might be needed.

K.3.1 Housing

The general magnitude of community service impacts can be estimated by examining the need for additional housing. Under the low baseline condition, the population in the bicounty area is expected to be about 10,000 people below the peak population level of 1981 at the beginning of construction activities in the late 1980s (see Table K.5). In 1981, the Tri-Cities had an estimated 2,000 vacant dwelling units, not including mobile homes and trailers. In addition, the momentum of planned housing construction led to further additions to housing stock in 1982, even though employment and population had begun to decline. Consequently, even if there were no additional growth in housing stock beyond 1982 and if some of the existing excess housing stock were lost because of dilapidation, at least 5,000 vacant units would be available in the Tri-Cities in the late 1980s, plus additional housing in surrounding communities and potential additions to housing by mobile homes. Under these conditions, none of the proposed alternatives would require the construction of additional housing to accommodate new population.

Under the high baseline condition, population growth in the study area is substantially greater. By 1990, the year construction of the geologic disposal alternative would start, baseline population is projected to be about 164,000 people, compared to the 1981 population of 148,056. By 1993, the year of peak population effect from the geologic disposal alternative, the total population in the study area is projected to be 166,262, of which 9,100 are due to the proposed alternative. In this case, the incremental population in the study area (approximately 18,000 people) between 1981 and 1993 would require approximately 6,000 housing units, assuming an average of three persons per household (Malhotra and Manninen 1980). In 1993, about 3,000 of these housing units would be needed by population associated with the geologic disposal activities. However, about 1,000 of these units would be needed for only the two peak years of 1993 and 1994, and thus would most probably be provided by mobile homes. Given the size of the existing housing stock in the study area, building capacity, and the fact that housing construction is likely to resume as the local economy improves, it does not appear that any shortage of housing would be created by this demand.

Since housing demands for the other alternatives are projected to be significantly lower than those of the geologic disposal alternative, adverse housing impacts due to any of these alternatives seem unlikely, particularly under the low baseline condition.

K.3.2 Traffic

Traffic congestion in the study area was aggravated by Hanford Site employment and population increases in the past. However, since 1981, traffic volume has decreased in conjunction with the decrease in activities at the Hanford Site. In addition, an increased emphasis on transportation planning in the Tri-Cities has resulted in decreased traffic congestion. These improvements, such as better intersection design, the completion of the I-182 bridge across the Columbia River, and the implementation of a Tri-Cities mass transit bus system are expected to alleviate certain aspects of congestion that would otherwise have been anticipated as construction activity at the Hanford Site and population in the bi-county area increased. The linear arrangement of the communities along the Columbia River will continue to contribute to some degree of traffic congestion. However, this congestion would be most directly related to access to the Hanford Site and limited to times of peak commuting to and from the Site. Staggered shift hours and use of mass transportation to the Site were used in the past to try to reduce commuter congestion. Such mechanisms undoubtedly will be used in the future. The total amount of increased traffic to the Hanford Site associated with any of the waste alternatives will be substantially lower than the amount of traffic related to the Supply System peak construction period.

K.3.3 Education

During the fall of 1974, many Tri-Cities schools were near or over capacity. Several suggestions were made to alleviate problems of overcrowding, including temporary portable classrooms, new construction, double shifts, and year-round school sessions (Woodward-Clyde 1975). As a result, there has been a considerable amount of new construction and additions to facilities since 1977. Some of the projects were planned before the downturn in the economy and were actually carried out in 1982 and 1983, creating excess capacity even at the peak population levels experienced in 1981. In 1982, the total excess capacity in these schools was estimated to be around 4,700 student positions. Since the school districts have continued to lose students in the downturn, the schools in this study area will be able to absorb student population growth caused by the construction and operation of any of the four waste alternatives. Therefore, no negative capital cost impacts with respect to schools are anticipated.

K.3.4 Utilities and Other Services

Although the study area's extraordinarily rapid growth between 1973 and 1981 put pressure on utilities and other services, gaps in the services to the population do not appear to have been substantial. Because of the largely unanticipated Supply System cutbacks in 1981, the planning and development of increased capacity in the region's community services were expanded beyond the immediate needs of the residents at that time. With declining population and economic activity, revenues dropped and budgets were readjusted. During the decline, all community services were affected, including staffing levels and space utilization requirements of health, education, public safety, and social services. Given adequate lead time and notification of future development activities, these affected departments and agencies can be expected to adjust to the projected economic and population conditions without undue difficulty. The high baseline condition would require resumption of growth management and expansion activities, even without any of the proposed alternatives.

K.4 FISCAL CONDITIONS

It is not yet possible to predict the total result of the current economic downturn associated with the Supply System rampdown because data outlining the fiscal condition of the region during this period of economic decline are not yet available. In view of the record of fiscal adaptability in the study area during the period of high growth in the 1970s, it seems likely that the less steep growth curves projected to be associated with the construction and operation of each of the waste disposal alternatives will not create serious problems in management or financing for the area.

As was the case during the high-growth period of the 1970s, it seems probable that the fiscal benefits that would accompany any of the four alternatives would primarily affect Richland, West Richland, Kennewick, and Benton City. However, the increased accessibility of Pasco, owing to the I-182 bridge, is likely to increase the share of new development activity and fiscal benefits occurring in Pasco.

K.5 SOCIAL CONDITIONS

Social conditions refer to both individual and community well-being, and in the case of the Hanford Site, include the "cultural community" of neighboring Indian tribes. The defense waste program is designed and intended to improve existing conditions. Because the implementation of any defense waste disposal alternative is projected to result in reduced impacts on the environment and reduced adverse health and safety consequences, compared with the "no action alternative," adverse social impacts are also expected to be insignificant. The prospect of improved environmental and radiological conditions is expected to have positive social consequences. The defense waste program should have no adverse effects on industrial and economic development decisions in the region, on the marketability of Washington agricultural products, on perceptions of the Tri-Cities as a good place to live and raise a family, on the attractiveness of the area for recreation or tourism, or on beliefs about the general quality of life of the local area, the region or the state.

The standard growth-related socioeconomic impacts are not sufficiently large, relative to the capacity of this area to absorb or manage growth and relative to its recent history of economic decline, to be expected to cause measurable adverse social problems, such as increased alcoholism, crime and other socially disruptive behaviors, or psychological stress responses related to excessively rapid social and community change.

Radiological impacts to offsite populations, including the possibility of adverse impacts on the Columbia River and its fisheries, are expected to be much smaller than the effects of background radiation, and therefore no significant associated social or cultural impacts to Indian or other populations are expected. The potential for disturbance of Indian lands or areas of religious significance, or restricted access to Indian lands, could result in sociocultural impacts. DOE and its contractors will adhere strictly to their compliance guidelines in order to ensure that cultural impacts of this sort are minimized.

During the last decade, a highly skilled labor force (from construction workers to professionals) has settled around the Hanford Site in anticipation of continued growth and employment opportunity. The early timing of the rampdown of the two major Supply System construction projects in mid-1981 was unexpected. While it is clear that a significant decline in employment and population has taken place in the interim, it also appears that many residents have not yet decided whether to stay or leave. This decision will depend on a number of factors, including opportunities elsewhere and the likelihood that this area will experience an employment upturn (or possibly another downturn) and the perceptions of local employment opportunities for the future. Given the uncertain future course of the local and regional economy, people living in and around the Tri-Cities are concerned about the well-being of their families and friends. Developments that contribute to economic stability and that reduce radiological health and safety risks associated with Hanford Site activities will be viewed as having beneficial impacts.

The implementation of any of the disposal alternatives would generally be viewed as a positive contributor to the area's recovery from the decline of the early 1980s. Since the geologic disposal alternative has the largest work force requirements of the disposal alternatives, its impacts in this area would be substantially greater than those from the other three alternatives.

K.6 REFERENCES

Cluett, C., P. Bolton, S. Malhotra, J. McStay and J. Slingsby. 1984. Nuclear Waste Repository in Basalt: Preliminary Socioeconomic Assessment. RHO-BW-CR-142 P/BHARC-400/83/023, prepared for Rockwell Hanford Operations, Richland, Washington.

Malhotra, S., and D. Manninen. 1980. Migration and Residential Location of Workers at Nuclear Power Plant Construction Sites. Vol. 2, Chapter 7. Battelle Memorial Institute, Seattle, Washington.

Rockwell Hanford Operations. 1985. Hanford Defense Waste Disposal Alternatives: Engineering Support Data for the HDW-EIS. RHO-RE-ST-30 P, Richland, Washington.

Washington State Employment Security Department, Research and Analysis Branch. 1984. "Labor Force, Employment and Unemployment for Richland-Kennewick-Pasco MSA, 1980-1984." Kennewick, Washington.

Woodward-Clyde Consultants. 1975. Socioeconomic Study: WPPSS Nuclear Projects 1 and 4. Prepared for the Washington Public Power Supply System, Richland, Washington.

THIS PAGE INTENTIONALLY
LEFT BLANK

APPENDIX L

NONRADIOLOGICAL IMPACTS--CONSTRUCTION AND OPERATIONAL PERIOD

THIS PAGE INTENTIONALLY
LEFT BLANK

APPENDIX L

NONRADIOLOGICAL IMPACTS--CONSTRUCTION AND OPERATIONAL PERIOD

L.1 INTRODUCTION

Appendix L includes various data relating to nonradiological consequences associated with the disposal alternatives and the no disposal action (continued storage) alternative as applied to the six waste classes. These impacts are secondary ones (some are quite minor) and probably would not be useful in deciding among alternatives; consequently, they are not included in the main body of this EIS. An exception is cost information that is presented in this Appendix. The difference in cost between the other alternatives and the geologic disposal alternative (considerably more expensive) may be a significant factor in selection.

Included in Appendix L are data on the following nonradiological environmental consequences, for each of the alternatives and for each of the six waste classes:

- emissions of nonradiological pollutants
- estimated injuries and fatalities
- requirements for depletable resources
- costs.

The geologic disposal alternative (Section L.2) and the reference alternative (combination disposal, L.4) are composed of numerous subalternatives and therefore require detailed tables and discussion to present the data. The results for the remaining sections (L.3, In-Place Stabilization and Disposal, and L.5, No Disposal Action) lend themselves to concise summary tables. Nonradiological impacts of the preferred alternative are bounded by the geologic and reference alternatives.

L.2 GEOLOGIC DISPOSAL ALTERNATIVE

The geologic disposal alternative would have the largest nonradiological consequences, being the most complex as well as the most expensive of the alternatives analyzed in this EIS.

Three types of deep geologic repositories are possible candidates. The reference offsite geologic receptor for transuranic (TRU)-containing wastes is the Waste Isolation Pilot Plant (WIPP) repository in New Mexico, and would involve a round trip of 4,800 km. An onsite repository in basalt is considered; this option would involve a round trip of 20 km. In addition, a generic repository (in granite or salt) is assumed for an offsite repository. To place an upper bound on impacts, this generic repository is assumed to involve a round-trip distance of 9,600 km to and from a point somewhere in the southeastern United States.

The geologic disposal alternative analyzed in this EIS is a multifaceted option; in other words, not every candidate waste is disposed of in its entirety in a geologic repository. For example, the strontium/cesium currently in capsules would be sent to a repository;

however, plans for existing tank waste call for dividing the waste into high-activity and low-activity fractions. The high-activity fraction would be vitrified, packaged, and sent to a geologic repository. The remainder would be made into a grout suitable for near-surface disposal on site. Estimated costs for "geologic disposal" include the costs of both operations.

L.2.1 Nonradiological Emissions

Processing, transportation, and disposal of waste would result in the emission of nonradioactive pollutants. The most significant would be dust, with smaller amounts of the other EPA-controlled pollutants.

The processing and disposal emissions are the sum of emissions generated during retrieval, packaging, storage, and site stabilization (Rockwell 1985).

Transportation emissions, given in Appendix I, are extremely small and well below applicable standards.

Pollutant emissions for geologic disposal are summarized in Table L.1. "Particulates" includes dust, which is generated during earth-moving activities. These are the totals that would be emitted over the period of years required to implement this alternative. At any given time the emissions would be within the applicable air-quality standards. Estimates for hydrocarbon emissions are included, although federal and state ambient air-quality standards for hydrocarbons have been dropped. For details on transportation-related air-quality impacts, the reader is referred to Appendix I.

L.2.2 Nonradiological Injuries, Illnesses, and Fatalities

The estimated number of postulated nonradiological injuries, illnesses, and fatalities associated with the geologic disposal alternative is based on accident experience for similar activities and on estimated manpower requirements. Manpower requirements for waste retrieval and processing are estimated in Rockwell (1985). Manpower estimates for repository mining, construction, and operation are based on information available in DOE (1980a) for 800-ha repositories and in DOE (1980b), prorated to that portion of the repository that would be occupied by each waste type. Methods of calculating nonradiological injuries, illnesses, and fatalities are detailed in Appendix G.[a] Results for the geologic disposal option are summarized in Table L.2, reported in integers. One fourth to one half of the postulated injuries, illnesses and fatalities are from repository construction activities.

L.2.3 Resource Requirements

The geologic disposal alternative will require the largest expenditure of depletable resources, partly because of the large underground repository that must be constructed to contain the wastes, and partly because of the extensive processing (e.g., vitrification) which precedes the actual disposal.

(a) Appendix G also contains definitions of terms used here such as occupational injury and illness and lost workdays, as given by the Occupational Safety and Health Administration.

TABLE L.1. **Summary of Onsite Nonradiological Emissions for the Geologic Disposal Alternative, t**

Existing Tank Waste	
Particulates	54,000
SO_x	2,400
CO	4,300
NO_x	2,000
HC	500
Future Tank Waste	
Particulates	2,600
SO_x	150
CO	200
NO_x	170
HC	30
Sr/Cs Capsules	
Particulates	10
SO_x	1,100
CO	30
NO_x	50
HC	4
TRU-Contaminated Soil Sites	
Particulates	110
SO_x	50
CO	70
NO_x	300
HC	20
Pre-1970 Buried TRU Solid Waste	
Particulates	1,500
SO_x	140
CO	200
NO_x	900
HC	30
Retrievably Stored and Newly Generated TRU Waste	
Particulates	1
SO_x	1
CO	4
NO_x	15
HC	2
Totals	
Particulates	58,000
SO_x	3,800
CO	4,800
NO_x	3,400
HC	590

TABLE L.2. Nonradiological Injuries, Illnesses, and Fatalities Postulated for the Geologic Disposal Alternative by Activity and Waste Class

Activity	Onsite Repository: Injuries and Illnesses[(a)]	Onsite Repository: Fatalities	Offsite Repository: Injuries and Illnesses[(a)]	Offsite Repository: Fatalities
Existing Tank Waste				
Retrieval and Processing	370	2	370	2
Transportation	0	0	6	1
Repository Construction and Operation	190	2	170	2
Total	560	4	550	5
Future Tank Waste				
Retrieval and Processing	42	0	42	0
Transportation	0	0	1	0
Repository Construction and Operation	35	0	25	0
Total	77	0	68	0
Sr/Cs Capsules				
Retrieval and Processing	4	0	4	0
Transportation	0	0	0	0
Repository Construction and Operation	15	0	13	0
Total	19	0	17	0

Waste Isolation Pilot Plant (WIPP)

Activity	Injuries and Illnesses[(a)]	Fatalities
TRU-Contaminated Soil Sites		
Retrieval and Processing	24	0
Transportation	1	0
Repository Construction and Operation	20	0
Total	45	0
Pre-1970 Buried TRU Solid Waste		
Retrieval and Processing	77	0
Transportation	3	0
Repository Construction and Operation	73	0
Total	150	0
Retrievably Stored and Newly Generated TRU Waste		
Retrieval and Processing	4	0
Transportation	10	1
Repository Construction and Operation	42	0
Total	56	1

(a) Lost workday cases.

Resource requirements estimated for the geologic disposal alternative include energy and materials.

Resources are those expended during retrieval, processing, and transportation (Rockwell 1985) combined with those required for construction and operation of a geologic repository (DOE 1980a,b). The repository resource values are prorated to that portion of the repository occupied by each waste type.

Resource requirements for each of the Hanford defense wastes are summarized in Table L.3. Annual U.S. production of some of these resources (DOE 1980a) is shown in Table L.4. The requirements shown in Table L.3 may be divided by factors of 15 to 30 to place these on an annual basis. The requirements are then seen to be small fractions of U.S. annual production. About 7.1 million m^3 of fill material will be required. Twenty-five percent of the fill material is soil, 65% is riprap, and 10% is gravel.

In addition to these material requirements, the geologic disposal alternative will require the use of manpower, as shown in Table L.5. About 90% of the manpower required for geologic disposal of tank waste is for onsite activities (retrieval and processing).

L.2.4 Costs

A summary of estimated costs for the geologic disposal alternative waste is presented in Table L.6. Retrieval and processing costs are taken from Rockwell (1985). Transportation costs are taken from Appendix I. Repository disposal cost estimates for the onsite and offsite repositories are taken from Appendix J. The estimates for an offsite repository (granite) are higher than for the onsite repository (basalt) because of vertical borehole emplacement methods currently assumed for HLW disposal in granite. WIPP cost estimates are based in part on recent preliminary studies of salt repositories in Texas. These cost estimates are lower than those for the hard rock media because of lower mining costs.

L.3 IN-PLACE STABILIZATION AND DISPOSAL ALTERNATIVE

In-place stabilization and disposal will produce moderate emissions of nonradiological pollutants. The principal pollutant will undoubtedly be particulate matter, most of it dust from earth-moving and other construction activities. Since these activities will take place centrally on the 1,500-km^2 Hanford Site, the dust will be only a localized onsite problem. Pollutant emissions for the in-place stabilization and disposal alternative are summarized in Table L.7. These are the totals that would be emitted over the years required to implement this alternative. At any given time the emissions would be within applicable air-quality standards. Estimates for hydrocarbon emissions are included, although federal and state ambient air-quality standards for hydrocarbons have been dropped.

Injuries, illnesses, and fatalities are summarized in Table L.8. Results are reported in integers. Calculation methods are detailed in Appendix G.[a] Resource requirements are

(a) Appendix G also contains definitions of terms used here such as occupational injury and illness and lost workdays, as given by the Occupational Safety and Health Administration.

TABLE L.3. **Resource Requirements for the Geologic Disposal Alternative**

	Onsite Repository	Offsite Repository	WIPP Repository
	Existing Tank Waste		TRU-Contaminated Soil Sites
Energy			
Propane, m^3	86,000	86,000	40
Diesel Fuel, m^3	86,000	88,000	3,600
Gasoline, m^3	8,600	8,900	520
Electricity, GWh	1,800	1,900	180
Coal, t	370,000	380,000	20,000
Materials			
Concrete, m^3	220,000	220,000	9,600
Steel, t	61,000	61,000	1,800
Stainless Steel, t	5,700	5,700	--
Copper, t	1,900	1,900	10
Lumber, m^3	44,000	44,000	90
	Future Tank Waste		Pre-1970 Buried TRU Solid Waste
Energy			
Propane, m^3	11,000	11,000	130
Diesel Fuel, m^3	11,000	11,000	12,000
Gasoline, m^3	1,000	1,100	1,700
Electricity, GWh	270	280	600
Coal, t	37,000	38,000	68,000
Materials			
Concrete, m^3	26,000	26,000	27,000
Steel, t	8,400	8,400	6,000
Stainless Steel, t	850	850	--
Copper, t	220	220	33
Lumber, m^3	2,500	2,500	300
	Sr/Cs Capsules		Retrievably Stored and Newly Generated TRU Waste
Energy			
Propane, m^3	170	170	80
Diesel Fuel, m^3	1,900	2,200	4,700
Gasoline, m^3	270	290	1,000
Electricity, GWh	60	65	2,100
Coal, t	27,000	28,000	12
Materials			
Concrete, m^3	3,700	3,800	12,000
Steel, t	560	580	2,200
Stainless Steel, t	20	20	--
Copper, t	7	7	20
Lumber, m^3	320	330	180

TABLE L.4. **Annual U.S. Production of the Key Resources Required for Implementation of Disposal** (DOE 1980a)

Resource	Annual U.S. Production
Propane, m^3	1×10^6
Diesel Fuel, m^3	4×10^8
Gasoline, m^3	6×10^8
Electricity, GWh	2×10^6
Steel, t	1×10^8
Lumber, m^3	3×10^9

TABLE L.5. **Manpower Requirements the for Geologic Disposal Alternative**

	Manpower, man-yr		
Waste Class	Onsite Repository	Offsite Repository	WIPP Repository
Existing Tank Waste	38,000	39,000	
Future Tank Waste	4,500	4,700	
Sr/Cs Capsules	640	640	
TRU-Contaminated Soil			3,300
Pre-1970 Buried TRU Solid Waste			8,100
Retrievably Stored and Newly Generated TRU Waste			2,600

summarized in Table L.9. In addition to the resources listed, about 9.2 million m^3 of fill material will be required. The fill material listed consists of 17% soil, 74% riprap, and 9% gravel. Costs are given in Table L.10.

L.4 REFERENCE ALTERNATIVE (COMBINATION DISPOSAL)

This alternative combines elements of geologic disposal and in-place stabilization and disposal and is intended to provide cost-effective, long-term disposal of wastes of varying character.

For existing tank wastes, the reference alternative would employ in-place stabilization and disposal of single-shell tank waste. Double-shell tank waste would be divided into a high-volume, low-activity fraction, suitable for grout stabilization and near-surface disposal, and a low-volume, high-activity fraction which would be vitrified and sent to geologic disposal either on site or off site.

TABLE L.6. **Summary of Estimated Costs for the Geologic Disposal Alternative, millions of $1987**

Activity	Tank and Capsules: Onsite Repository	Tank and Capsules: Offsite Repository	TRU Wastes (to WIPP)
	Existing Tank Waste		TRU-Contaminated Soil Sites
Retrieval and Processing	8,500	8,500	410
Transportation	29	380	46
Repository Emplacement	4,200	4,300	12
Totals (rounded)	12,700	13,200	470
	Future Tank Waste		Pre-1970 TRU Buried Solid Waste
Retrieval and Processing	1,000	1,000	1,400
Transportation	9.2	67	130
Repository Emplacement	710	720	42
Totals (rounded)	1,700	1,800	1,600
	Sr/Cs Capsules		Retrievably Stored and Newly Generated TRU Waste
Retrieval and Processing	92	92	130
Transportation	5.4	13	38
Repository Emplacement	110	110	12
Totals (rounded)	210	220	180

TABLE L.7. **Nonradiological Emissions for the In-Place Stabilization and Disposal Alternative, t**

Pollutant	Existing Tank Waste	Future Tank Waste	Sr/Cs Capsules	TRU Soil Sites	Pre-1970 TRU	Retrievably Stored and Newly Generated TRU	Totals
Particulates	11,000	2,600	140	3,300	2,500	2,800	22,000
SO_x	80	280	340	29	23	40	790
CO	1,000	200	160	260	200	340	2,200
NO_x	350	300	130	130	100	180	1,200
HC	110	35	15	32	24	42	260

Future tank wastes (double-shell tanks) would be fractionated in much the same way. The lower-activity fraction would be disposed of as grout on site, and the high-activity fraction would be vitrified before disposal in a geologic repository, again either on site or off site.

Strontium and cesium currently in capsules would be sent to a geologic repository either on site or off site.

TABLE L.8. Nonradiological Injuries, Illnesses, and Fatalities Postulated for the In-Place Stabilization and Disposal Alternative

Waste Class	Injuries and Illnesses[a]	Fatalities
Existing Tank Waste	70	0
Future Tank Waste	23	0
Sr and Cs Capsules	10	0
TRU-Contaminated Soil Sites	1	0
Pre-1970 Buried TRU Solid Waste	2	0
Retrievably Stored and Newly Generated TRU Waste	1	0
Totals	110	0

(a) Lost workday cases.

In the reference alternative, TRU-contaminated soil and pre-1970 buried TRU solid waste are considered to have been disposed of in place and require no further action except the filling of voids with grout and covering with the protective barrier and marker system.

Newly generated TRU waste, which is retrievably stored, would be handled differently; the largest fraction, contact-handled TRU waste, would be processed and packaged for disposal in WIPP. The remote-handled TRU waste fraction is so small that, rather than having its own processing facility, it would probably be handled by processing in a special campaign in the Waste Receiving and Processing facility (Appendix E), and also disposed of in a geologic repository.

L.4.1 Nonradiological Emissions

Although the reference alternative also utilizes geologic disposal for both existing and future double-shell tank wastes (except that technetium and strontium are not removed), the nonradiological emissions are significantly less than for the geologic disposal alternative because the single-shell tank wastes are disposed of in place. For tank waste, emissions are those generated during processing and repository activities. Processing emissions are generated during retrieval, packaging, storage and onsite stabilization (as estimated in Rockwell 1985). Emissions from repository activities (DOE 1980a) are prorated to the portion of a repository that existing and future double-shell tank wastes would occupy.

In the offsite case, vitrified high-activity waste is assumed to go (by rail) to a repository (9,600 km round trip). Pollutant concentrations resulting from transport of waste to an offsite repository are extremely small and well below applicable standards. Estimates for hydrocarbon emissions are included, although federal and state ambient air quality standards for hydrocarbons have been dropped. Emissions from transportation to an onsite

TABLE L.9. Resource Requirements for the In-Place Stabilization and Disposal Alternative

Resource	Existing Tank Waste	Future Tank Waste	Sr/Cs Capsules	TRU-Contaminated Soil Sites	Pre-1970 Buried TRU Solid Waste	Retrievably Stored and Newly Generated TRU Waste	Totals
Energy							
Propane, m^3	2,100	430	580	--	--	--	3,100
Diesel Fuel, m^3	31,000	8,100	1,500	8,400	18,000	11,000	78,000
Gasoline, m^3	1,000	190	220	540	290	250	2,500
Electricity, GWh	1,300	100	110	--	--	--	1,500
Coal, t	--	30,000	43,000	--	--	--	73,000
Manpower, man-yr	6,300	2,100	770	90	170	120	9,500
Materials							
Concrete, m^3	14,000	1,600	2,300	--	--	--	18,000
Steel, t	3,000	470	1,000	--	0	6,500	11,000
Stainless Steel, t	0	10	20	--	--	--	30
Copper, t	26	2	4	--	--	--	32
Lumber, m^3	4,000	200	290	--	--	--	4,500
Manpower, worker-yr	6,300	2,100	770	90	170	120	10,000

TABLE L.10. **Summary of Estimated Costs for the In-Place Stabilization and Disposal Alternative, millions of $1987** (Rockwell 1985)

	Processing and Stabilization	Protective Barrier and Marker System	Total (rounded)
Existing Tank Waste	1,250	190	1,400
Future Tank Waste	450	30	500
Sr/Cs Capsules	200	11	210
TRU-Contaminated Soil Sites	1.2	67	68
Pre-1970 Buried TRU Solid Waste	0.35	140	140
Retrievably Stored and Newly Generated TRU Waste	5.9	62	68
Totals (rounded)	1,900	500	2,400

repository are insignificant and are not listed separately. For details on transportation-related air-quality impacts, the reader is referred to Appendix I.

The strontium and cesium currently in capsules and retrievably stored TRU waste would be sent to geologic disposal in the reference alternative, and the estimated emissions are equivalent to those for geologic disposal (Table L.1). In this alternative, TRU-contaminated soil and pre-1970 buried TRU solid waste are disposed of by in-place stabilization and disposal, and the emissions are equivalent to those reported for that alternative (Table L.7).

Pollutant emission data are summarized in Table L.11. "Particulates" includes dust. These totals are emitted over the years required to implement this alternative. At any given time the emissions would be within applicable air-quality standards.

L.4.2 Nonradiological Injuries and Fatalities

The numbers of postulated nonradiological injuries and fatalities associated with the reference alternative are summarized in Table L.12. Results are reported in integers. Methods used to estimate nonradiological injuries and fatalities are detailed in Appendix G.[a]

L.4.3 Resource Requirements

Requirements for resources for the reference alternative are summarized in Table L.13. Resource requirements for strontium and cesium capsules are the same as for the geologic disposal alternative. Resource requirements for TRU-contaminated soil and pre-1970 TRU solid wastes are the same as for the in-place stabilization and disposal alternative. About 6 million m^3 of fill material will be required. Sixteen percent of the fill material is soil, 72% is riprap, and 12% is gravel.

(a) Appendix G also contains definitions of terms used here such as occupational injury and illness and lost workdays, as given by the Occupational Safety and Health Administration.

TABLE L.11. **Summary of Onsite Nonradiological Emissions for the Reference Alternative** (combination disposal), t

Existing Tank Waste	
Particulates	11,000
SO_x	180
CO	1,100
NO_x	430
HC	120
Future Tank Waste	
Particulates	2,600
SO_x	150
CO	270
NO_x	180
HC	30
Sr/Cs Capsules	
Particulates	7
SO_x	1,100
CO	30
NO_x	40
HC	4
TRU-Contaminated Soil Sites	
Particulates	3,300
SO_x	30
CO	260
NO_x	130
HC	30
Pre-1970 Buried TRU Solid Waste	
Particulates	2,000
SO_x	20
CO	200
NO_x	100
HC	20
Retrievably Stored and Newly Generated TRU Waste	
Particulates	1
SO_x	1
CO	5
NO_x	20
HC	2
Totals	
Particulates	19,000
SO_x	1,500
CO	1,900
NO_x	900
HC	210

TABLE L.12. Nonradiological Injuries, Illnesses, and Fatalities Postulated for the Reference Alternative (combination disposal)

Activity	Onsite Repository: Injuries and Illnesses[(a)]	Onsite Repository: Fatalities	Offsite Repository: Injuries and Illnesses[(a)]	Offsite Repository: Fatalities
		Existing Tank Waste		
Retrieval and Processing	84	0	84	0
Transportation	0	0	0	0
Repository Construction and Operation	5	0	4	0
Total	89	0	88	0
		Future Tank Waste		
Retrieval and Processing	48	0	48	0
Transportation	0	0	0	0
Repository Construction and Operation	6	0	5	0
Total	54	0	53	0
		Sr/Cs Capsules		
Retrieval and Processing	4	0	4	0
Transportation	0	0	0	0
Repository Construction and Operation	15	0	13	0
Total	19	0	17	0
		Onsite Activities		
		TRU-Contaminated Soil Sites		
In-Place Stabilization and Disposal	1	0	NA[(b)]	
		Pre-1970 Buried TRU Solid Wastes		
In-Place Stabilization and Disposal	4	0	NA[(b)]	

Waste Isolation Pilot Plant (WIPP)

Activity	Injuries and Illnesses[(a)]	Fatalities
	Retrievably Stored and Newly Generated TRU Wastes	
Retrieval and Processing	5	0
Transportation	10	1
Repository Construction and Operation	46	0
Total	61	1

(a) Lost workday cases.
(b) NA--not applicable.

TABLE L.13. **Resource Requirements for the Reference Alternative (combination disposal)**

Resource	Onsite Repository	Offsite Repository
	Existing Tank Waste	
Energy		
Propane, m^3	7,400	7,400
Diesel Fuel, m^3	33,000	33,000
Gasoline, m^3	1,500	1,500
Electricity, GWh	1,500	1,500
Coal, t	15,000	15,000
Materials		
Concrete, m^3	30,000	30,000
Steel, m^3	6,800	6,800
Stainless Steel, t	730	730
Copper, t	180	180
Lumber, m^3	7,700	7,700
Manpower, worker-yr	7,800	7,800
	Future Tank Waste	
Energy		
Propane, m^3	6,100	6,100
Diesel Fuel, m^3	9,200	9,300
Gasoline, m^3	570	580
Electricity, GWh	100	100
Coal, t	3,500	3,800
Materials		
Concrete, m^3	18,000	18,000
Steel, t	4,000	4,000
Stainless Steel, t	620	620
Copper, t	150	150
Lumber, m^3	1,800	1,800
Manpower, worker-yr	4,400	4,400
	Sr/Cs Capsules	
Energy		
Propane, m^3	170	170
Diesel Fuel, m^3	1,900	2,200
Gasoline, m^3	270	290
Electricity, GWh	60	62
Coal, t	27,000	28,000
Materials		
Concrete, m^3	3,700	3,800
Steel, t	560	580
Stainless Steel, t	20	20
Copper, t	7	7
Lumber, m^3	300	300
Manpower, worker-yr	640	630

TABLE L.13. (contd)

Resource	In-Place Stabilization and Disposal TRU-Contaminated Soil Sites
Energy	
Diesel Fuel, m^3	8,400
Gasoline, m^3	540
Manpower, worker-yr	90
	Pre-1970 Buried TRU Solid Waste
Energy	
Diesel Fuel, m^3	16,000
Gasoline, m^3	230
Electricity, GWh	2
Materials	
Steel, t	310
Manpower, worker-yr	300
	Disposal in WIPP — Retrievably Stored and Newly Generated TRU Waste
Energy	
Propane, m^3	85
Diesel Fuel, m^3	5,700
Gasoline, m^3	1,100
Electricity, GWh	2,100
Materials	
Concrete, m^3	13,000
Steel, t	2,700
Copper, t	22
Lumber, m^3	190
Manpower, worker-yr	2,700

L.4.4 Costs

Estimated costs for the reference alternative are summarized in Table L.14. Costs for strontium and cesium capsules are the same as for the geologic disposal alternative. Costs for TRU-contaminated soil and pre-1970 TRU solid wastes are the same as for the in-place stabilization and disposal alternative.

L.5 NO DISPOSAL ACTION (CONTINUED STORAGE)

The no disposal action (continued storage) alternative, basically a continuation of present practices will also have some minor nonradiological consequences.

A summary of pollutant emissions is presented in Table L.15; all of these emissions are very minor. Injuries, illnesses and fatalities (Table L.16), based on historical operational

TABLE L.14. **Summary of Estimated Costs for Disposal, the Reference Alternative** (combination disposal), **millions of $1987**

Activity	Existing Tank Waste Onsite	Existing Tank Waste Offsite	Future Tank Waste Onsite	Future Tank Waste Offsite	Sr/Cs Capsules Onsite	Sr/Cs Capsules Offsite
Retrieval and Processing	1,900	1,900	1,200	1,200	92	92
Transportation	8	14	8	19	6	13
Repository Emplacement	100	110	130	130	110	110
Totals (rounded)	2,000	2,000	1,300	1,300	210	220

Activity	TRU-Contaminated Soil Sites	Pre-1970 Buried TRU Solid Wastes
Processing and Stabilization	1.2	47
Barrier and Marker System	67	118
Totals (rounded)	68	170

Activity	Retrievably Stored and Newly Generated TRU Waste
Retrieval and Processing	130
Transportation to WIPP	44
Repository Emplacement	13
Totals (rounded)	190

TABLE L.15. **Nonradiological Emissions for the No Disposal Action** (continued storage) **Alternative, t**

Pollutant	Emissions
Particulates	100
SO_x	330
CO	170
HC	120
NO_x	18

TABLE L.16. **Nonradiological Injuries, Illnesses, and Fatalities Postulated for the No Disposal Action** (continued storage) **Alternative**

	Injuries and Illnesses[a]	Fatalities
Existing Tank Waste	90	0
Future Tank Waste	24	0
Sr and Cs Capsules	15	0
TRU-Contaminated Soil Sites	1	0
Pre-1970 Buried TRU Solid Waste	0	0
Retrievably Stored and Newly Generated TRU Waste	0	0
Totals	130	0

(a) Lost workday cases.

data, are also quite low.[a] Results are reported in integers. Requirements for resources are summarized for each waste class in Table L.17. About 0.7 million m^3 of soil will also be required. In addition, land requirements vary from 2 to 14 ha, all of which is land already dedicated to nuclear activities. Costs are summarized for the six waste classes in Table L.18.

Each of these impacts is estimated for continued storage for 100 years.

(a) Appendix G contains definitions of terms such as occupational injury and illness and lost workdays, as given by the Occupational Safety and Health Administration.

TABLE L.17. Resource Requirements for the No Disposal Action (continued storage) Alternative

Resource	First 100 yr
Existing Tank Waste[a]	
Propane, m^3	8,000
Diesel Fuel, m^3	22
Gasoline, m^3	200
Electricity, GWh	130
Coal, t	31,000
Concrete, m^3	22,000
Steel, t	13,000
Lumber, m^3	5,200
Manpower, worker-yr	8,200
Future Tank Waste[a]	
Propane, m^3	8,000
Diesel Fuel, m^3	3
Gasoline, m^3	200
Electricity, GWh	50
Coal, t	32,000
Concrete, m^3	22,000
Steel, t	12,000
Stainless Steel, t	23
Lumber, m^3	2,500
Manpower, worker-yr	2,200
Sr/Cs Capsules	
Propane, m^3	580
Diesel Fuel, m^3	3
Gasoline, m^3	200
Electricity, GWh	120
Coal, t	43,000
Concrete, m^3	2,300
Steel, t	1,000
Stainless Steel, t	20
Copper, t	4
Lumber, m^3	290
Manpower, worker-yr	1,300
TRU-Contaminated Soil Sites	
Diesel Fuel, m^3	41
Gasoline, m^3	400
Manpower, worker-yr	70
Pre-1970 Buried TRU Solid Waste	
Diesel Fuel, m^3	19
Gasoline, m^3	190
Manpower, worker-yr	40
Retrievably Stored and Newly Generated TRU Waste	
Diesel Fuel, m^3	26
Gasoline, m^3	470
Manpower, worker-yr	40

(a) Based on retanking every 50 years.

TABLE L.18. **Summary of Estimated Costs for the No Disposal Action** (continued storage) **Alternative, millions of $1987**

Activity	Cost for First 100 yr	Cost for Each 100 yr Thereafter
Existing Tank Waste		
Retrieval and Retanking	330	320
Surveillance	690	460
Subtotal	1,000	780
Future Tank Waste		
Retrieval and Retanking	390	380
Surveillance	59	45
Subtotal	450	430
Sr/Cs Capsules		
Overpacking and Maintenance	250	0
Surveillance	47	64
Subtotal	300	64
TRU-Contaminated Soil Sites		
Monitoring and Surveillance	10	10
Vegetation Control	0.19	0.19
Subsidence Maintenance	0.80	0.80
Subtotal	11	11
Pre-1970 Buried TRU Solid Waste		
Monitoring and Surveillance	4.7	4.7
Vegetation Control	0.35	0.35
Subsidence Maintenance	0.35	0.35
Subtotal	5.4	5.4
Retrievably Stored and Newly Generated TRU Waste		
Maintenance and Surveillance	9.4	9.4
Total (rounded)	1,800	1,300

L.6 REFERENCES

Department of Energy (DOE). 1980a. Final Environmental Impact Statement--Management of Commercially Generated Radioactive Waste. 3 vols. DOE/EIS-0046F, Washington, D.C.

Department of Energy (DOE). 1980b. Final Environmental Impact Statement--Waste Isolation Pilot Plant. DOE/EIS-0026, Washington, D.C.

Rockwell Hanford Operations. 1985. Hanford Defense Waste Disposal Alternatives: Engineering Support Data for the HDW-EIS. RHO-RE-ST-30 P, Richland, Washington.

THIS PAGE INTENTIONALLY LEFT BLANK

INDEX

THIS PAGE INTENTIONALLY
LEFT BLANK

INDEX

9 7 0 1 1 4 7 1 1 9